Experimental Breeder Reactor II (EBR-II) Level 1 Probabilistic Risk Assessment

Nuclear Science and Engineering Division

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Experimental Breeder Reactor II (EBR-II) Level 1 Probabilistic Risk Assessment

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APPENDIX B: FAULT TREES

ACRONYMS AND ABBREVIATIONS

I. General

ACRDS	Automatic Control Rod Drive System
ACS	Argon Cooling System
ANL	Argonne National Laboratory
ANL-E(ast)	Argonne National Laboratory, Illinois Site
ANL-W(est)	Argonne National Laboratory, Idaho Site
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ATR	Advanced Test Reactor
ATWS	Anticipated Transient Without Scram
BFTF	Breached Fuel Test Facility
BOP	Balance-of-Plant
BWR	Boiling Water Reactor
CCF	Common Cause Failure
CDF(cdf)	Cumulative Distribution Function
CDFR	Commercial Demonstration Fast Reactor
CRBR(P)	Clinch River Breeder Reactor (Plant)
CRDM	Control Rod Drive Mechanism
CREDO	Centralized Reliability Data Organization
CP	Control Power (Relays)
CPS	Constant (Continuous) Power System
DAS	Data (Digital) Acquisition System
DBW	Design Basis Wind
DCS	Digital Control Station
DHR	Decay Heat Removal
DN	Delayed Neutron
DSNP	Dynamic Simulator for Nuclear Power Plants
EBR-II	Experimental Breeder Reactor II
ECCS	Emergency Core Cooling System
EP	Emergency Procedures
EP	Engineering Physics Division (ANL)
EPRI	Electric Power Research Institute
ESF	Engineered Safety Feature
ESRP	Eastern (portion) of the Snake River Plain
FAA	Federal Aviation Agency
FCF	Fuel Cycle Facility (ANL)
FERD	Fuel Element Rupture Detection (System)
FFTF	Fast Flux Test Facility
FMEA	Failure Modes and Effects Analysis
FTAP	Fault Tree Analysis Program

ACRONYMS AND ABBREVIATIONS (Cont'd)

GLASS	Germanium-Lithium Argon Scanning System
GSCS	Graphite and Shield Cooling System
HEP	Human Error Probability
HEPA	High-Efficiency Particulate-Air
HFBR	High Flux Beam Reactor
HFEF	Hot Fuel Examination Facility (ANL)
HFIR	High Flux Isotope Reactor
HMLD	Hydrogen Meter Leak Detector
HPP	High Pressure Plenum
IDCOR	Industry Degraded Core Rulemaking Group
IE	Initiating Event
IEEE	Institute of Electrical and Electronic Engineers
IFR	Integral Fast Reactor
IFR	Instrument Flight Rules
IHX	Intermediate Heat Exchanger
INEL	Idaho National Engineering Laboratory
IOT	Inherent Operability Tests
IREP	Interim Reliability Evaluation Program
LCDL	Delayed Loss of Circulation
LCRA	Rapid Loss of Circulation
LLNL	Lawrence Livermore National Laboratory
LMFBR	Liquid Metal Fast Breeder Reactor
LMR	Liquid Metal Reactor
LOCA	Loss of Coolant Accident
LOCP	Loss of Constant Power
LOF(WS)	Loss of Flow (without Scram)
LOHS(WS)	Loss of Heat Sink (without Scram)
LONP	Loss of Normal Power
LOS(A)	Loss of Sodium (Accident)
LOPI	Loss of Piping Integrity
LPP	Low Pressure Plenum
LWR	Light Water Reactor
MG (M/G)	Motor-Generator
MSL	Mean Sea Level
MV/I	Millivolt to Current Converter
NAS	National Academy of Sciences
NCPS	Numerically Controlled Positioning System
NESC	National Energy Software Center
NITF	Nuclear Instrument Test Facility
NQA-1	Quality Assurance Program Requirements for Nuclear Facilities (ANSI/ASME Standard)
OC	OverCooling
PDF(pdf)	Probability Density Function
PICT	Plant Inherent Control Test
P&ID	Piping and Instrumentation Diagram
PFR	Prototype Fast Reactor

ACRONYMS AND ABBREVIATIONS (Cont'd)

PRA	Probabilistic Risk Assessment
PRD	Power Reactivity Decrement
PTCGSSS	Primary Tank Cover Gas Sampling Supply System
PWR	Pressurized Water Reactor
QA	Quality Assurance
QAE	Quality Assurance Element
QAPD	Quality Assurance Program Description
RA	Reactor Analysis Division (ANL)
RAS	Reactor Analysis and Safety Division (ANL)
RAW	Risk Achievement Worth
RBCB	Run-Beyond-Clad-Breach
RDF	Rod Drop Feedback Reactivity
RE	Reactor Engineering Division (ANL)
RIFL	Large Fast Reactivity Insertion
RRW	Risk Reduction Worth
RSS	Reactor Shutdown System
SA (S/A)	Subassembly
SAP6	Structural Analysis Program - Version 6
SASSYS	Safety Analysis Code for Plant Transient Analysis
SAS4A	Safety Analysis Code for Severe Accident Analysis
SDD	System Design Description
SGTR	Steam Generator Tube Rupture
SHRT	Shutdown Heat Removal Test
SOT	Subassembly Outlet Temperature
SRP	Savannah River Plant
SRP	Standard Review Plan
TAN	Test Area North
THTB	Transient Heat Transfer Code - Version B
TMI-2	Three Mile Island - Unit 2
TOP	Transient OverPower
TSOF	Total Sodium Flow
UOR	Unusual Occurrence Report
UPFB	Reactor Upper Plenum Flow - Channel B
UPS	Uninterruptible Power Supply
USDOE (DOE)	United States Department of Energy
USNRC (NRC)	United States Nuclear Regulatory Commission
VFR	Visual Flight Rules
WCC	Warning Communications Center
WR	Wide-Range

ACRONYMS AND ABBREVIATIONS (Cont'd)

II. Accident Sequence Analysis

1. Initiating Events

ANTC	Anticipatory Shutdown
DKHR	Decay Heat Removal Sub-tree
EVTR	Evaporator Tube Rupture
I	Initiating Event - Reloading Errors
IDAM	Fuel Damage Sub-tree
IE	Initiating Event
LCDL	Loss of Circulation - Delayed
LCRA	Loss of Circulation - Rapid
LF1A	Single Pump Loss of Flow - Group A
LF1B	Single Pump Loss of Flow - Group B
LF1C	Single Pump Loss of Flow - Group C
LF1D	Single Pump Loss of Flow - Group D
LF2A	Double Pump Loss of Flow - Group A
LF2B	Double Pump Loss of Flow - Group B
LF2C	Double Pump Loss of Flow - Group C
LF2D	Double Pump Loss of Flow - Group D
LF2E	Double Pump Loss of Flow - Group E
LF2F	Double Pump Loss of Flow - Group F
LMFR	Liquid Metal Fire
LOCP	Loss of Constant Power
LONP	Loss of Normal Power
LOPI	Loss of Piping Integrity
MDLF	Local Faults: Metal Driver Breach Initiator
MELF	Local Faults: Metal Experiment Breach Initiator
OCP	Overcooling by Primary Pump Overspeed
OCSL	Overcooling by Major Steam Leak
OCS	Overcooling by Secondary Pump Overspeed
OELF	Local Faults: Oxide Experiment Breach Initiator
RIFL	Fast Reactivity Insertion (> 1.00 \$)
RIFS	Fast Reactivity Insertion (< 0.35 \$)
RISA	Slow Reactivity Insertion - Group A (< 0.20 \$)
RISB	Slow Reactivity Insertion - Group B (0.20-0.65 \$)
RISC	Slow Reactivity Insertion - Group C (0.65-1.10 \$)
RISD	Slow Reactivity Insertion - Group D (> 1.10 \$)
SCRM	Scram
SDFR	Shutdown Cooler NaK Fire in Containment
SFLE	Subassembly Local Fault from Loading Error
SFPB	Subassembly Local Fault from Partial Blockage
SHDL	Long Shutdown
SHDS	Short Shutdown
SPTR	Superheater Tube Failure
SSFR	Secondary Sodium Fire in Containment
TAIR	Transient - Loss of Instrument Air
TBOP	Loss of Steam Plant Heat Removal

ACRONYMS AND ABBREVIATIONS (Cont'd)

TNDO	Tornado
TSDC	Transient - Unrecoverable Loss of a Shutdown Cooler
TSEC	Loss of Secondary System Heat Removal
TSHC	Transient - Unrecoverable Loss of Shield Cooling

2. Event Tree TOPs

BPHR	Balance of Plant Decay Heat Removal
DETP	Detect in Time to Shutdown and Prevent Propagation
DETS	Detect in Time to Shutdown and Prevent Propagation
DHR	Decay Heat Removal
DHRL	Decay Heat Removal Long Term
DHRS	Decay Heat Removal Short Term
DRLA	Decay Heat Removal Long Term (1 sd cooler + shield cooling)
DRLB	Decay Heat Removal Long Term (2 shutdown coolers)
DRSA	Decay Heat Removal Short Term (1 sd cooler + shield cooling)
DRSC	Decay Heat Removal Short Term (1 sd cooler)
DSRC	Decay Heat Removal Short Term
DUME	Water and Sodium Dump before Leak Escalation
DUMS	Water and Sodium Dump before Leak Escalation
FROD	Sufficient Control or Safety Rods Drop during LOF Event
FSDG	Less than 2 DEG Failures within First 0.2 Seconds
FSIG	Scram Signal during LOF Event
LKDT	Leak Detection by HMLDs
LLRF	Limited Leak
LMFD	Liquid Metal Fire Detection
LROD	Sufficient Control or Safety Rods Drop during LOF Event
MODS	Leak Escalates within First Minute
MRSS	Manual SCRAM Under LOHS Conditions
NBFM	No Boiling or Fuel Melting in Average Core Assembly
NPFB	No Pin Failure or Boiling in T.S. Limit Pins
NPPB	No Pin-to-Pin Propagation
NPPD	No Pin-to-Pin Propagation
NPPE	No Pin-to-Pin Propagation
NPPO	No Pin-to-Pin Propagation
NPPP	No Pin-to-Pin Propagation
NSSP	No Subassembly-to-Subassembly Propagation
NSSS	No Subassembly-to-Subassembly Propagation
PKRU	Pumps Keep Running for 1 Hour
PRRE	Pressure Relief by Secondary Boundary Failure
PRRS	Pressure Relief by Secondary Boundary Failure
PRUN	Primary Pumps keep Running
PS	Protective Surveillance
RFIR	Reactor Shutdown
RIHX	Reactor Shutdown before IHX Failure
RROD	Sufficient Control or Safety Rods Drop during TOP Event
RSIG	Scram Signal during TOP Event

ACRONYMS AND ABBREVIATIONS (Cont'd)

S	Safety Rod Signal
SCRM	Successful Scram
SHTP	Shutdown of Reactor
SHTS	Shutdown of Reactor
SLPD	Self-Limiting Pin Propagation
SLPE	Self-Limiting Pin Propagation
SLPP	Self-Limiting Pin Propagation
SMLK	Small or Medium Leak
SNPB	Subassembly Fault: No Pin Breach
SPRS	Pressure Relief System Activates
SPTR	Secondary Pump Trips
SSSD	Fire Suppression: No Secondary Sodium Dump
SYSO	Systems in Other Locations Operate as Required
T	Safety Rod Trip
TGTT	Temperature Between Eutectic Temp. and 1500°F for 60 s in Hot Pin
TLTT	Temp. < Eutectic Temp. in Hot Pin

3. Fuel Damage Categories (End States)

CD	Core Damage
CSD	Core Structural Damage
MCD	Minor Core Damage
ND	No Damage
PED	Potential Experiment Damage

EXECUTIVE SUMMARY

EXPERIMENTAL BREEDER REACTOR-II LEVEL 1 PRA

1.0 BACKGROUND

The National Academy of Sciences Study of the USDOE Category A Reactors recommended that a Probabilistic Risk Assessment (PRA) should be performed on the EBR-II Reactor. In January 1989, a Level 1 PRA was initiated. This report documents work performed in completing the Level 1 PRA except for seismic analysis; it describes the methodology used, assumptions made, the analysis performed, and the final results. The seismic risk assessment is in progress. It will be separately documented.

Probabilistic Risk Assessments are categorized at three levels. The product of a Level 1 PRA is the identification of the dominant component failure combinations that may lead to fuel damage and their frequencies. In a Level 2 PRA the physical processes occurring during an accident are analyzed and the response of the confinement system determined. The time and mode of confinement failure as well as the inventories of radionuclides released to the environment are predicted. A Level 3 PRA analyzes the transport of radionuclides through the environment and can assess the public health or economic consequences of an accident.

2.0 OBJECTIVES

The EBR-II Level 1 PRA successfully fulfilled four primary objectives:

- Identification of the plant features that are the most important contributors to the EBR-II fuel-damage frequency.
- Development of plant logic models for use in analyzing safety and reliability issues.

- Identification of dominant EBR-II accident sequences and their frequencies.
- Establishment of an Argonne National Laboratory team with demonstrated expertise in performing risk and reliability analyses.

3.0 SCOPE

The EBR-II Level 1 PRA examined internal events such as equipment failures, operator and maintenance errors, seismic events and internal fire and flooding. Initiating events were identified through a review of plant event reports and a functional analysis of the plant. An examination of plant records provided a plant-specific data base. Where required by lack of site-specific data, generic data bases were employed with careful consideration of EBR-II unique features and conditions. A thorough consideration of common-cause events was included, particularly in the reactor shutdown system.

4.0 VALIDITY AND REVIEW

To ensure a high degree of validity of the EBR-II models, there was extensive involvement of EBR-II operations, engineering analysis, and safety staff personnel in performing the study. This close cooperation of the EBR-II staff in the initial analyses will facilitate downstream use of the PRA for risk related design or operational decisions.

In addition, review cycles of the program were implemented. These included:

- Review and approval of the system descriptions, success criteria, and fault trees by the plant system and nuclear safety engineers.
- External (non-ANL) peer review of the program methodology and results by PRA consultants.
- Independent review of the methodology and results by EBR-II division, ANL-West.

- Independent review of the scope and methodology by a senior management review panel (ANL).

5.0 RESULTS

An important result of a Level 1 PRA is the estimation of the frequency of events resulting in damage. The EBR-II Level 1 PRA calculated a total accident sequence frequency from internal events of slightly less than 2×10^{-6} per year. Figure 1 compares the results of this study with the results from a recent study of commercial LWRs. It should be emphasized that the EBR-II is not directly comparable to LWRs as EBR-II has many unique design features and mitigating safety systems which act to limit the progression of reactor fuel damage. Also the definition of damage for the EBR-II PRA is cast in such a way as to include accidents that may lead only to minor or no core damage under specific circumstances, while Level 1 PRAs for commercial LWRs often assume that incipient core melt will always progress to full core melt. Therefore caution should be used in comparing the EBR-II results with any specific LWR. However, as shown in Fig. 1, the EBR-II risk profile is lower than commercial LWRs.

The risk from external events is also low. The dominant nonseismic external event is a liquid metal fire under the deck plates which is judged to lead to damage with an annual frequency of 3.6×10^{-6} per year. Table 1 compares the results of this study with other USDOE Category A reactors. Again it can be seen that EBR-II appears to represent a low risk when compared to these reactors.

The conclusions reached by the Level 1 study are:

- The frequency of damage at EBR-II is generally lower than that reported for commercial LWRs and the USDOE Category A reactors.
- There is no single dominant source or cause of the various accident sequences.
- The passive decay heat removal function and the ability of reactivity feedbacks to limit the temperature excursion in a transient lead to low risk.

- The passive decay heat removal function and ability of reactivity feedbacks to limit temperature excursion in a transient also lead to a sparse set of requirements for active safety systems or operator intervention.
- EBR-II is relatively insensitive to human error and only a few important human actions were identified.
- EBR-II has little or no requirement for electrical power to prevent fuel damage.

Fig. 1. Comparison of EBR-II Damage Frequency with Core Damage at Commercial LWRs
(LWR data from NUREG-1150)

Table 1. Mean Core Damage Frequencies for USDOE Facilities

Reference		Mean Core Damage Frequencies (Yr. ⁻¹)				Total External	Total
		Total Internal	Seismic	Fire	Wind/ Tornado		
	EBR-II	1.6 10 ⁻⁶		3.6 10 ⁻⁶	-		
[13.4]	ATR	1.8 10 ⁻⁴	-	-	-	-	-
[13.5]	HFBR	3.5 10 ⁻⁴	-	-	-	-	
[13.3]	HFIR	3.1 10 ⁻⁴	1.2 10 ⁻⁴	1.9 10 ⁻⁵	2.9 10 ⁻⁴	4.3 10 ⁻⁴	7.4 10 ⁻⁴
[13.6]	N-Reactor	6.7 10 ⁻⁵	1.7 10 ⁻⁴	1.7 10 ⁻⁵	-	1.9 10 ⁻⁴	2.5 10 ⁻⁴
[13.7] [13.8]	SRP K-Reactor	2.1 10 ⁻⁴	1.2 10 ⁻⁴	1.4 10 ⁻⁷	-	2.2 10 ⁻⁴	4.3 10 ⁻⁴

1.0 INTRODUCTION

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1.0 INTRODUCTION

1.1 Project Motivation and Initiation

The Committee to Assess Safety and Technical Issues at DOE Reactors of the National Academy of Sciences issued a report entitled "Safety Issues at the DOE Test and Research Reactors" in August 1988 [1.1]. Among the recommendations in this report is: "All of the DOE Class A reactors should have Level I PRAs. The need for Level II and Level III PRAs should be given careful consideration...". Specifically with respect to EBR-II, "The Committee recommends that a Level I PRA be performed for the EBR-II plant and that refueling accidents and the reliability of the reactor protection system and containment isolation be carefully examined in the context of the PRA." In response to this recommendation, a Level I Probabilistic Risk Assessment (PRA) for EBR-II which addresses identification and quantification of the frequencies of sequences which might lead to core damage was initiated.

A major benefit of performing a PRA for EBR-II is to explicitly evaluate the risks of the EBR-II plant by bringing together modern systems analysis and the existing analytical and experimental results pertaining to EBR-II safety in the integrated PRA framework. The PRA results include an integrated model of the plant that delineates the interfaces and relationships among the inherent characteristics of the reactor, the equipment, procedures, and human activities, and quantifies the contributions of these components to safety and to risk. The PRA models also allow assessment of the impacts of plant modifications, aging of plant equipment, procedure changes, and technical specifications changes on risk. In addition, the PRA models can be used to assess risks associated with both irradiation and plant experiments, and allow evaluation of experimental limits and controls intended to limit risk. Finally, the PRA provides a basis for operator training in identifying and responding to risk dominant accident sequences.

Realization of all of the potential benefits of an EBR-II PRA requires that there be an ongoing process of development and improvement of PRA models of the plant, the data base supporting quantification of the models, the evaluations of accident sequences, and applications

of the PRA in decision making processes. Thus, the EBR-II PRA should not be regarded as a closed-ended study, but as an on-going process of model development, analysis, and applications in performance of the EBR-II mission.

Upon completion and review of the Level I study, which is the subject of this document, effort will be applied to consideration of a limited scope containment study.

1.2 Project Scope

The scope of the EBR-II PRA is summarized in Table 1.1.

1.3 Topical Summaries of PRA Report Sections

1.3.1 Volume I

Section 2.0 of this report describes the formulation of EBR-II PRA project plans and the assessment and selection of PRA methodologies applicable to this study.

Section 3.0 provides a descriptive overview of the EBR-II plant, including an historical perspective of operations and some of the major plant improvements over 25 years. This section also provides an insight into the unique features of EBR-II and some of the integral plant experiments performed to demonstrate the relatively benign behavior of the reactor when subjected to severe loss-of-flow and loss-of-heat-sink transients.

Section 4.0 presents a discussion of special features of EBR-II, including the passive safety features of the reactor with respect to inherent shutdown from full power.

Section 5.0 provides a description of the accident sequence selections pertinent to EBR-II operations and the concomitant analyses that could lead to several categories of core damage.

Table 1.1. Scope of EBR-II Level I PRA

Sources of Radioactivity Considered

- Accumulated fission products and actinides in core and blanket subassemblies.
- Induced activity in structural materials and in sodium coolant.
- Accumulated fission products in subassemblies in the storage basket or undergoing fuel transfer between core and basket.

Plant States Considered

- Plant configuration is as of October 1, 1989 (except as noted).
- Core configuration is considered to be mostly driver fuel, with irradiation experiments operating within Technical Specification (Tech. Spec.) limits.
- Initial operating conditions are considered to be within Tech. Spec. limits, with allowable conditions assumed to occur.

Initiating Events Considered

- Generic initiating event categories usually examined for LMRs such as Loss-of-Flow (LOF), Transient Over Power (TOP), Loss-of-Heat-Sink (LOHS), Over Cooling (OC) and Local Faults.
- Initiating events with LWR analogues such as Loss-of-Coolant-Accident (LOCA) - for EBR-II, Loss-of-Sodium Accident.
- General Transients.
- Criticality (fuel handling).
- External events, particularly earthquakes and fires.

Section 6.0 summarizes all of the detailed EBR-II systems analyses contained in Appendix C of this document. The summary includes schematic system diagrams, a discussion of failure data and the system failure rates obtained from fault tree analysis.

Section 7.0 describes the criteria for selection of failure rates, aging effects, and event probabilities from generic PRA literature, LMR plant data, and EBR-II plant-specific data.

Section 8.0 provides the rationale and data selection for human factors analysis as applied to EBR-II operations and procedures.

Section 9.0 describes the basis for seismic analysis of the EBR-II site and plant components. An earthquake is considered to be the major external event that could affect plant integrity.

Section 10.0 considers the PRA effects of other external events, specifically fires, internal floods, and tornadoes.

Section 11.0 presents the process and results of sequence quantification, which combines the accident sequence event trees with the results of fault tree analyses of EBR-II systems to produce the core damage frequencies and the percentage contribution of each sequence to each core damage state.

Section 12.0 discusses the importance of each core damage contributor and the attendant uncertainties and sensitivities of the data input.

Section 13.0 presents an interpretation of all the qualitative and quantitative results of the EBR-II PRA.

Section 14.0 presents the conclusions and recommendations resulting from the PRA effort.

Appendix A describes the risk analysis of core refueling operations.

Appendix B contains all fault trees and input data used in the quantification of the EBR-II PRA.

1.3.2 Volume II (System Notebooks)

Appendix C contains the details of systems analyses for the EBR-II front-line and support systems including system descriptions, data selection, fault tree analyses, and quantification leading to system failure rates. This Appendix contains twelve system notebooks as follows:

- C.1 Reactor Shutdown System
- C.2 Primary Pumps
- C.3 Primary Pumps Motor/Generator Sets
- C.4 Shutdown Coolers
- C.5 Shield Cooling
- C.6 Thimble Cooling
- C.7 Stack Exhaust System
- C.8 Secondary Sodium Systems/Balance-of-Plant Systems
- C.9 Electrical Systems
- C.10 Instrument and Plant Air Systems
- C.11 Cooling Water Systems
- C.12 Argon Systems

1.3.3 Volume III

Appendix D contains documents written specifically to support the EBR-II PRA with respect to PRA methodology, passive safety features of EBR-II, local fuel element faults, loss-of-sodium accidents and seismic analysis.

1.4 Document Chronology

The first version of the system notebooks, Appendix C, was released as Revision 0 in February 1990. The main body of the text was issued for ANL internal review in October and November 1990 and the updated system notebooks in January and February 1991 and labelled Revision 1. After resolution of all the review comments, the entire document was released as Revision 2 in June 1991.

Section 1.0 References

- 1.1 Committee to Assess Safety and Technical Issues at DOE Reactors, "Safety Issues at the DOE Test and Research Reactors," National Academy Press, (1988).

**2.0 PROJECT PLANS
AND METHODS ASSESSMENT**

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2.0 PROJECT PLANS AND METHODS ASSESSMENT

2.1 Project Organization

In response to the recommendation in the report entitled "Safety Issues at the DOE Test and Research Reactors" [2.1] in August 1988, an effort was begun in mid-1989 at ANL to organize a project dedicated to the production of a Level I Probabilistic Risk Assessment (PRA) for EBR-II. At the direction of the Laboratory Director and the Associate Laboratory Director for Engineering Research, the principal responsibility for planning and performing the EBR-II PRA was vested in the Reactor Analysis and Safety Division (RAS) at ANL. Within the project duration, a reorganization of the Engineering Research divisions resulted in the dissolution of RAS and the subsequent creation of two new divisions, Reactor Analysis (RA) and Reactor Engineering (RE). The principal EBR-II PRA responsibility currently is delegated to RA, with staff support from RE, Engineering Physics (EP), and EBR-II. Because of the matrix management structure of the project, the project organization has remained intact (see Section 2.4, "Quality Assurance" and the details of the QA Program [2.2]).

The principal individual responsibility for production of the PRA is vested in the EBR-II PRA Project manager, who reports to RA division management along with periodic reports to the Senior Review Panel which consists of senior ANL technical management who are also responsible for review of the PRA final draft (internal review). The Project Manager is responsible for coordination of the technical tasks and advisory reviews performed by the Systems Analysts in conjunction with the EBR-II Technical Support Group. He is also responsible for ensuring that the methodology adopted is state-of-the-art; to this end, the USNRC PRA procedures guide [2.3] is used for initial guidance, complemented by the inclusion of current expertise from outside contractors in the areas of PRA methodology, Human Reliability, and Seismic Analysis.

The EBR-II PRA was initiated with a one week EBR-II familiarization course held at ANL-W, Idaho Falls, in January 1989. It was planned that a first draft of the final report be delivered to ANL management, the EBR-II Division, and the EBR-II PRA Senior Review Panel in October 1990, with submission of the report for DOE review in 1991.

2.2 Objectives and Scope

2.2.1 Objectives

Probabilistic risk assessments can be used to achieve many different objectives, ranging from evaluation of a plant's performance in comparison to a probabilistic safety goal, through evaluation of the effects of technical specification modifications, to support of an operational risk tracking system. The details of the PRA modeling required depend on the objectives. The objectives of the EBR-II PRA are summarized as follows:

1. To quantitatively assess the level of safety of EBR-II and the uncertainties in the assessment.
2. To identify changes in design or operations which could significantly improve safety.

Initially, these objectives are addressed by completion of a Level I PRA analysis focused on assessment of core damage frequency. The need for extension of the analysis to Level II and Level III will be evaluated when Level I results for core damage frequency are available.

3. To develop a method for continuing use by EBR-II in assessing effects of plant modifications, experiments, and operational changes.

This objective will be addressed by making the initial PRA models available in convenient form for use in on-going analysis of EBR-II. A continuing process of upgrading and extending the initial PRA models, adding new models, and improving the data base supporting the models is visualized. It is intended that the PRA will become a part of the EBR-II evaluation methods for

use as needed in the decision-making process. In order to fully implement the PRA in its intended role, a final objective is:

4. To develop staff expertise in performance and application of risk analysis.

Development of expertise at ANL in performance and application of PRA techniques, and in interpretation of results is essential to the continuing development and applications of the PRA. Table 2.1 presents the specific objectives of the PRA in relationship to the general objectives stated above.

It is noted that general objectives 1 and 2 and the specific objectives 1 through 7 represent the first order of business for the EBR-II PRA Project, and is addressed in this report. A quantitative assessment of the safety of EBR-II in PRA terms is considered to be the most important near term goal of the project. In the longer term, integration of the PRA into the overall battery of analysis methods available to EBR-II, and continuing utilization of the PRA as a decision-making tool is perhaps more important to the continued safe and efficient operation of the plant.

2.2.2 Scope

The EBR-II PRA Project has concentrated on completing a Level I PRA for the reactor system, which focuses on producing a comprehensive list of accident scenarios for EBR-II together with a measure of expected occurrence frequencies and associated uncertainties and an assessment in terms of core damage alone. At the conclusion of the Level I study, a limited study will be conducted to assess the need for more detailed analysis of containment loads, containment performance, and radiological consequences of releases from containment. The scope of the Level I analysis of the reactor system is summarized in Table 2.2.

Table 2.1. General and Specific PRA Objectives

<u>General Objectives</u>	<u>Related Specific Objectives</u>
1. To quantitatively assess the level of safety of EBR-II and the uncertainties in the assessment.	1. To identify those accident sequences which are the dominant contributions to core damage frequency.
2. To identify changes in design or operations which could significantly improve safety.	2. To identify and model those systems and components which are important to accident sequence initiation and/or mitigation.
	3. To identify those human actions which are important to accident sequence initiation and/or mitigation.
	4. To assess the effects of external events (in particular, seismic events) on core damage frequency.
	5. To assess the effects of EBR-II passive safety features in reducing both the frequency and consequences of the accident sequences.
	6. To assess important dependencies.
	7. To develop a database for use in quantifying system and component failure rates and external events frequencies.
3. To develop a method for continuing use in assessment of plant modifications, experiments, and operational changes in EBR-II.	8. To develop a computerized PRA database for continuing use in analysis of EBR-II.

Table 2.1 General and Specific PRA Objectives (Cont'd)

<u>General Objectives</u>	<u>Related Specific Objectives</u>
4. To develop staff expertise in performance and application of risk analysis.	9. To develop a computerized method for updating the database as additional experience is accumulated. 10. To extend existing models and add new models as needed. 11. To familiarize EBR-II analysis, engineering, and operations personnel with PRA concepts and their utilization. 12. To establish in-house expertise in all major areas of PRA analysis, to facilitate continuing development of the PRA and it's application in decision-making.

Table 2.2. Scope of EBR-II Level I PRA

Sources of Radioactivity Considered

- Accumulated fission products and actinides in core and blanket subassemblies.
- Induced activity in structural materials and in sodium coolant.
- Accumulated fission products in subassemblies in the storage basket or undergoing fuel transfer between core and basket.

Plant States Considered

- Plant configuration is as of October 1, 1989 (except as noted)
- Core configuration is considered to be mostly driver fuel, with irradiation experiments operating within Technical Specification (Tech. Spec.) limits.
- Initial operating conditions are considered to be within Tech. Spec. limits, with allowable conditions assumed to occur.

Initiating Events Considered

- Generic initiating event categories usually examined for LMRs such as Loss-of-Flow (LOF), Transient Over Power (TOP), Loss-of-Heat-Sink (LOHS), Over Cooling (OC) and Local Faults.
- General Transients.
- Criticality (fuel handling).
- External events, particularly earthquakes and fires.

Not included in the scope are reactor start-up events, partial power operation, plant experiments performed under special procedures, such as the automatic control rod drive system in high-speed mode, and human error-acts of commission. None of these categories is considered to introduce significant effects on the level of risk associated with EBR-II operations.

In order to assess the comparative level of risk associated with operation of EBR-II the core damage frequencies derived are compared to those for commercial LWRs such as given in NUREG-1150 [2.4], any relevant USDOE goals, the results of other PRAs being performed on USDOE Class A Facilities and the results for other LMRs. These benchmarks are summarized in Table 2.3.

2.3 Methodology

The purpose of this description is to provide an introduction to, and an overview of, the basic risk analysis methodology used in this study. The various review bodies, whose comments on DOE class A facilities provided the impetus for this study, have emphasized state-of-the-art methodology [2.1]. The individual components of the methodology will be developed in later sections.

The USNRC PRA procedures guide [2.3] has been used to define the scope and nature of the present work. This risk analysis therefore fits within a previously defined formal structure for such activities. However for each specific PRA there are still many decisions on methodology which must be made. In the first instance, it is intended that a comprehensive Level I [2.3] PRA be performed. Therefore, the first task of this risk analysis was to produce a comprehensive list of accident scenarios for EBR-II together with a measure of expected occurrence frequencies and associated uncertainties, and an assessment of the consequences in terms of core damage.

Table 2.3. Benchmarks for Comparative Risk

- Commercial LWRs (NUREG-1150) [2.4]
- USDOE goals [2.5]
- PRAs for other USDOE Category A facilities [2.1]
- PRAs for other LMRs [2.9, 2.11]

The accidents which are identified are intended to encompass all potential accident scenarios which might possibly lead to core damage in EBR-II. In particular this includes external events, with emphasis on fire, internal floods, and earthquake. At the initiation of the present analysis, it was anticipated that these three external events would contribute significantly to the core damage frequency. It should be noted that, in view of the life extension studies at EBR-II, careful note will be taken of scenarios which threaten availability as well as safety.

In this risk analysis, the event trees have been developed to the level of detail necessary to clearly define the accident sequence but with front-line and support systems treated using fault trees. That is, we have adopted the "small event tree-linked fault tree" approach to the analysis. In the quantification of the trees, uncertainty is built in from the outset, using standard methodology [2.3].

Figure 2.1 is a diagrammatic representation of the process which was followed in performing the Level I EBR-II PRA. The various activities are discussed in detail in later sections. However, it is important to recognize the first, and key, task of plant familiarization. Obviously a meaningful PRA cannot result unless those charged with performing the PRA become thoroughly familiar with the design and operation of EBR-II. This was attained by plant walkthroughs and by accessing available sources of information on these subjects. The acquisition of the basic information is crucial to the success of the project.

Two features of EBR-II introduce different considerations into this PRA. The first of these is the ability of EBR-II passive safety characteristics, such as reactivity feedbacks and decay heat removal, to mitigate the effects of several accident sequences. Indeed, an unprotected Loss-of-Flow (LOF) and an unprotected Loss-of-Heat-Sink (LOHS) have been performed as plant experiments in EBR-II [2.6]. One of the challenges in performing this PRA is to take account of these characteristics of EBR-II in an appropriate manner. Second, it is important to consider the mission of EBR-II as a research facility, which conducts experimental irradiations and plant transient experiments, within the PRA. These factors must be considered in characterizing the reactor and in assessing accident sequence initiation and outcome, so that

Fig. 2.1. Task Flow Diagram

the risk assessment adequately considers the impact of experiments on facility risk. In this way, the PRA can be used as a factor in decisions on experiment acceptability.

2.3.1 Plant Familiarization

In Fig. 2.1, the first activity identified is plant familiarization. As noted above, this activity is crucial to the successful performance of the entire PRA. Plant and system familiarizations were continuous activities which took place throughout the project in order that the PRA analysts were fully aware of how the plant is built and operated.

The assembly of the necessary information for analysis of EBR-II was a prodigious task. The first step in the process was a short course on EBR-II given by EBR-II staff to the PRA staff with the dual objective of providing a background for consideration of EBR-II and to open lines of communication between the largely Illinois-based PRA staff and the Idaho-based engineers whose understanding of EBR-II needed to be transferred. To further enhance this activity a list of principal engineering, operations, and maintenance contacts for each system, and subsystem, within EBR-II was established.

An integral part of the familiarization process is the plant walkthrough. A first general walkthrough formed part of the short course on EBR-II, thereafter individual analysts undertook system specific inspections with the cognizant EBR-II engineer(s). A great deal of emphasis was placed upon these personal contacts because, for a plant having the extensive operating experience of EBR-II, the personal insights of an experienced staff are of high value to systems analysts.

The systems analyses of the EBR-II systems operating characteristics are documented in a series of system notebooks (Appendix C of this document and a summary in Section 6.0). The system notebooks are an important product of the PRA, independently of the risk analysis implications, because they represent a consolidation of EBR-II documentation and operating data and experience which heretofore was not available. E.g., one important feature of the systems analyses is to identify and document all the systems interfaces and dependencies.

Documents which are important to the systems analysis procedure are the System Design Descriptions (SDDs), Operating Instructions and Maintenance Procedures, Piping and Instrumentation Diagrams (P&IDs) and detailed drawings. EBR-II documents used for developing the initiating events and event trees include existing safety documentation, the Unusual Occurrence Reports (UORs), and the Emergency Procedures. Excellent background material is found in the training manuals and the Life Extension study [2.7].

2.3.2 Accident Sequence Analysis

The first formal step in the risk analysis is to produce the list of initiating events and to layout the corresponding event trees. This is discussed further below.

2.3.2.1 Internal Initiating Events

The safety of LMRs with respect to large radiological releases has been traditionally discussed in terms of three generic whole-core accident sequences, broadly categorized as TOP (Transient Over Power), LOF (Loss-of-Flow) and LOHS (Loss of Heat Sink) [2.8], together with Local Faults. For PRA application, this categorization must be broadened to list all possible scenarios. Use of the methods in reference 2.3, shows that, beyond these three categories above, a number of other possibilities must be considered, including plant transients and a LOCA (Loss-of-Coolant Accident). The classic primary system LOCA is inappropriate for a pool-type LMR (i.e., EBR-II); for the EBR-II PRA, the appropriate analogue is a Loss-of-Sodium Accident, so designated throughout this document.

Having established the broad categories of accident scenarios, a list of initiating events was compiled. This process enumerated all possible events which could (conceptually) lead to core damage and or radionuclide release. This list of initiating events was generated by:

- (1) Determining how each of the categories from reference 2.3 can occur.

- (2) Reviewing generic lists of accident initiating events from other LMR studies ([2.9], [2.10]), [2.11]).
- (3) Reviewing the EBR-II UORs (Unusual Occurrence Reports) for actual events which have occurred at EBR-II.
- (4) Considering events for which emergency procedures are included in the EBR-II Emergency Procedures manual [2.12].

There are some special issues at EBR-II which are incorporated into the analysis. The National Academy of Sciences (NAS) review drew attention to refueling accidents [2.1] and these are the subject of a separate study within the PRA. Similarly, the NAS review identified containment isolation as requiring particular attention. Containment isolation will be discussed in follow-on studies.

Once the provisional list of initiating events was in place it was reviewed for completeness. This review process, which involved EBR-II operations staff, EBR-II management and PRA specialists, was essential to establishing the base list of initiating events which underpins the analysis. The initiating events are grouped in terms of the expected system response to reduce the number of event models. Finally, the frequency for each initiating event was derived from plant specific data, fault tree analysis or expert elicitation.

2.3.2.2 External Initiating Events

There is a special class of initiating event which are typically considered separately because of the unique effects members of this class can have on the behavior of the system. These are the external events, of which three were expected to be particularly important for EBR-II, fires, internal floods, and earthquakes. Other potential external events were screened for inclusion.

2.3.2.2.1 Earthquakes

Following guidelines laid down in earlier PRA studies, the seismic analysis involved:

- (1) Seismic Hazard: This is a determination of how frequently ground motions occur at the INEL site and what accelerations are associated with them. There was some information on site characterization already available from a study of DOE sites conducted by LLNL [2.13], and from actual earthquakes [2.14]. However, it was decided that a new hazard curve for the ANL-West site was needed to extend the frequency range and uncertainty bands and to remove any undue conservatism for PRA applications. In addition, seismological estimates of earthquake magnitude are needed to validate assumptions regarding vertical and horizontal motion.
- (2) Fragility: This requires the determination of the responses of the structures and plant equipment to any given earthquake.
- (3) Plant Logic: This is the exercise of determining the effects of the combinations of system failures on the plant behavior.

These three features are then assembled to provide a description of plant damage states and the associated probabilities.

The basic methodology for the seismic risk assessment is that referred to in the NRC procedures guides as the Zion Method, first used in the Oyster Creek Seismic Risk Assessment [2.15] and later improved and used in the Zion Probabilistic Safety Study [2.16]. In this method, the various fragilities are expressed in terms of a ground motion parameter (peak ground acceleration). Dynamic analyses of the structures or the equipment are required to determine the response of that system to an earthquake. For EBR-II, a limited amount of analysis was available which provided some of the required information. Additional dynamic

analyses were performed at ANL. Clearly it is not practical to perform a dynamic analysis of every part of EBR-II, so a priority list was developed on the basis of potential hazards of seismic failures. In particular, priority was given to systems in which failure could lead directly to core damage or which could disable passive safety functions.

2.3.2.2.2 Fires

The fire analysis follows the steps outlined in the NRC procedures guide [2.3]. A screening analysis was performed to identify critical fire impact locations and effort concentrated upon these locations. Event tree and fault tree logic models were constructed to identify any routes that lead from fires to potential core damage.

2.3.2.2.3 Other External Events

The usual lists of potential external events were reviewed for applicability to EBR-II [2.3]. For those external events which warranted further consideration, a two-tiered screening analysis was performed to identify the routes to potential core damage. If a particular external event was identified which has a significant impact on plant risk then that event was the subject of more detailed analysis. External events are discussed in detail in Section 10.

2.3.2.3 Event Trees

The process of event tree development was to some degree iterative, in that only when a complete understanding of the EBR-II accident responses was acquired could the definitive set of headings be produced, and yet this process provided the frame upon which the other activities are based.

For the EBR-II PRA, the top events and their success criteria were chosen to lead to the spectrum of possible accident sequences and their associated probabilities given a specific initiating event. It is not possible to immediately step from this result to estimates of core

damage, since the EBR-II core varies in composition from run to run. Any particular core is composed of assemblies of some or all of the following types, Mk II driver fuel (U-5Fs, 316SS), Mk III (U-10Zr, D9), Mk IIIA (U-10Zr, 316SS), metal fuel experiments (U-10Zr, U/Pu/Zr, HT-9, D9, 316SS) and oxide fuel experiments (UO₂, PuO₂, 316SS, D9, HT-9). There are also other types of fuel (e.g., UN) which conceivably could be irradiated in the future. Irradiation experiments are likely to be of high burnup and to be run-to-clad-breach (RTCB) in order to find burnup limits for a particular fuel type/clad combination, or even run-beyond-clad-breach (RBCB). These experiments are typical of those being used to qualify a new driver fuel for use in EBR-II. The essential point is that the probability of a given sequence can be derived from general consideration of the plant, assuming that the particular core configuration does not affect the plant system and passive response, whereas the extent of core damage in any particular sequence is heavily dependent on the core configuration. Therefore, the event trees are constructed in a manner which defines accident sequences, but these are not automatically combined into core damage, or not core damage, bins. The response of the reactor core and individual types of subassemblies to particular accident sequences is described in a separate event tree and the conditional probability of damage given an accident sequence is evaluated.

At the conclusion of the Accident Sequence Analysis task there was available a complete list of initiating events, internal and external, each of which was screened for its contribution to core damage frequency and grouped according to their system response. Event trees have been developed for each initiating event or initiating event group. The top events in the event trees provide the focus for systems analysis in terms of definitive success and failure criteria for those system responses. The event trees themselves define sets of accident sequences which can be modelled using coupled neutronic/thermal-hydraulic codes such as SASSYS [2.17].

2.3.3 Systems Analysis

Once the event trees are complete the Level I analysis requires systems analysis input to proceed to quantification. This task is lengthy and complex, requiring a full analysis of each front-line and support system to establish the probability of success for each safety function and associated uncertainties, together with the initiating event frequencies, in order to

provide the probabilities of each sequence. The probabilities associated with each top event can be different from sequence to sequence if there is dependency between system performance and the initiating event.

2.3.3.1 System Modeling

The systems analysis used standard fault tree methodology to describe the failure modes of each system analyzed. The design of EBR-II is such that there are very few mitigative functions, apart from scram and natural convection decay heat removal, required to protect the core from damage and therefore much of the emphasis in the systems analysis was upon identifying dependencies and searching for potential initiating events.

A system notebook was produced for each system analyzed. This is discussed further in Section 6; the notebooks constitute Appendix C.

2.3.3.2 Human Reliability

In a system such as EBR-II, humans are relied upon to perform many tasks in the event of off-normal occurrences at the plant, e.g., manual scram, manual activation of safety rods, etc. As a result, human reliability will be a major consideration in the possible causes of system failure. The importance of human factors is shown, for example, by the fact that, in the accident at TMI-2, a major contributing cause was operator error prior to and throughout the accident. Other areas where human error can be implicated in accident progression are inadequate procedures, training or accident response planning. The human reliability analysis involved screening the PRA model to identify human factors related initiating events and then investigating the role of the operator in accident sequences, particularly through recovery actions.

2.3.3.3 Common Cause

Common causes for failures can lead to unexpectedly high probabilities of redundant system failures where they are found to exist. In general terms, the more reliable a system from the viewpoint of random independent failure, the more likely it is that common cause will dominate the total failure frequency.

In the analysis the β factor method was used to treat common cause. The potential for common cause failure was assessed using a methodology similar to that expounded in a recent joint USNRC/EPRI report [2.18].

2.3.3.4 Data Base

Twenty-five years of operating history at EBR-II provides a potentially large data base for the evaluation of system reliability. This is particularly so for the more recent operating history where detailed records have been kept. However, these facility records were not kept with a PRA in mind. Nevertheless, the raw data from maintenance, testing, and operation of EBR-II was applied where possible. This plant specific data was augmented by using generic industry data. In addition, the EBR-II division has been contributing data to the Oak Ridge LMR data base, CREDO [2.19]. Maximum use was made of all the data held in CREDO, EBR-II, Fast Flux Test Facility (FFTF), and Japanese (JOYO) databases, to establish EBR-II or LMR specific data. Also, generic data sources for non-LMR specific components were utilized where appropriate (e.g. electronics). Figure 2.2 gives an outline of the total information flow required to establish an EBR-II data base and the corresponding PRA data base. It should be noted that all data relevant to the production of the PRA, including source data, event trees, and fault trees has been centralized in a single database, created using the relational database ORACLE [2.20].

Fig. 2.2. Data Flow in the EBR-II PRA

2.3.4 Accident Sequence Quantification

Accident sequence quantification is the process by which the results from the separate tasks are synthesized to give overall frequencies for a given accident sequence. For each initiating event, the various sequences are defined by linking the appropriate fault trees and this, combined with the initiating event frequency, leads to the overall frequency for any specific sequence. In order to provide a link for future risk estimates, some estimate of core damage for a given sequence is required. Core damage is not arbitrarily assumed. The technique applied is to perform thermal hydraulic analyses of the specific sequences to determine the level of damage to driver fuel and various experiments. This was done using the SASSYS code [2.17].

2.3.4.1 Core Damage

The definition of what constitutes core damage is a special issue in the EBR-II PRA for two reasons, firstly because the core varies in composition from run to run and secondly because EBR-II is a research reactor modified to routinely accommodate fuel experiments with breached cladding. As discussed earlier a particular core might contain metallic driver fuel and both oxide and metal fueled experimental assemblies as well as more exotic fuel types which might be irradiated as experiments. The top events were defined in terms of temperature limits upon driver fuel. Driver fuel in this context is anything which satisfies the current or anticipated Technical Specification limits in power and flow conditions and burnup. Therefore, it can be evaluated whether a particular sequence leads to damage in driver fuel and an assessment could be made as to the degree of damage from each experiment. It is highly time-consuming to assess the damage for each assembly on a run by run basis and continually update the assessment, (note that burnup affects the potential damage); therefore, for the purposes of damage assessment, the damage is defined solely in terms of driver fuel, but a special damage category for experiments is included.

2.3.4.2 Uncertainty and Importance in Accident Sequences

Uncertainty analysis is carried out in order to determine the uncertainty in the results for various identifiable uncertainties. These include model uncertainties, statistical uncertainties, and the uncertainty associated with completeness. In general, uncertainty becomes more important when accident sequence frequencies are low.

A related topic is that of importance. Importance analysis seeks to identify those systems whose existence is most important in maintaining the current level of risk and those systems whose improvement would most reduce the existing risk. There are various possible measures of importance, the more common measures of importance are Risk Achievement Worth (RAW), Risk Reduction Worth (RRW) and Fussell-Vesely. All three of these measures have been applied.

Sensitivity studies seek to determine if variation of model parameters and assumptions significantly affect the results of the PRA. This is carried out in a less systematic manner than the statistical treatment of uncertainty but is important in providing insight into the results of a PRA.

2.3.5 Mathematical and Computing Methods

Over the last fifteen years many computer codes have been written which are designed to facilitate the performance of PRAs. The selection of appropriate codes for application in this PRA depends as much upon availability as their appropriateness for the task. The final product of this PRA consists of the elements of a risk based management system, which EBR-II operations can use routinely to assess the effect of plant modifications upon risk. Therefore, it was a requirement of the code system adopted that it be portable, i.e. easily transferred from ANL-East to ANL-West. Ideally the code system would be self contained, i.e. it would not rely on any particular commercial product, and that ANL would control the source code so that the configuration could be monitored without being dependent upon any external organization.

Several commercial codes, USNRC or EPRI sponsored, were evaluated and adapted to EBR-II PRA requirements to perform cross-check calculations as verification of the ANL chosen codes and also to demonstrate that the code package chosen had been implemented successfully.

The chosen code suite (EBR2PRA) has been implemented for use on the SUN computer systems at ANL.

2.3.5.1 Event Trees

The event trees used for the Level I PRA are analyzed by resolving the accident sequences. Sequences are analyzed by an in-house program that uses some of the subroutines of MODULE [2.21] for quantification.

2.3.5.2 Fault Trees

Since the preferred methodology used in this study involves the study of large fault trees, the most important modeling decision to be made was the selection of the computer code (or suite of codes) used to handle the complex problems which arise from large fault trees.

Once the qualitative understanding of the system failure is acquired, the fault tree analysis code can be used to generate sequence probabilities, primarily based upon the minimal cutsets and the system data base. A version of a Korean-authored fault tree code MODULE [2.21], was installed upon the SUN workstation and used in this project. Several sample problems were run and compared to better known codes such as FTAP to provide verification for the code. MODULE was combined with other special purpose codes and TREDRA [2.22] to create a package termed EBR2PRA. The applicability and precision of this package has been benchtested against the PRA results from a commercial code using EBR-II input data.

2.3.5.3 Uncertainty and Importance

Uncertainty in both fault and event trees is handled within the codes, or code suites chosen. For fault trees it was a functional requirement of the code selected that it handle uncertainty and importance readily. MODULE has an integrated treatment of uncertainty and importance. This capability was extended into EBR2PRA.

2.3.5.4 Sequence Evaluation

Accident sequences in LMRs have received a great deal of attention and a number of codes have been written to follow the course of accidents up to and including fuel damage. The best example of this capability is the ANL-authored fast reactor accident analysis code SAS4A/SASSYS [2.17]. In the case of EBR-II, the EBR-II Division of ANL has two plant specific models which are used for routine thermohydraulic and safety analysis, NATDEMO [2.23] and DSNP [2.24]. Neither of the latter two codes treats fuel failure or events beyond fuel failure. However, any of the three can be used to estimate whether a given plant state will likely lead to core damage using thermal criteria. This allows cross-check calculations to be performed.

2.3.6 Interpretation and Application of Results

When the quantification portion of the PRA is complete the process of interpretation of the numerical results determines whether the final product meets the stated objectives. Identification of the risk dominant accident sequences at the end of the Level I PRA provides insight into those potential plant vulnerabilities that lead to core damage. Each dominant sequence is examined to determine the factors that are the major contributions to its frequency. The potential plant vulnerabilities thus identified, are then examined to determine whether improvements can be identified that might significantly reduce the likelihood of core damage.

2.4 Quality Assurance

The EBR-II PRA Project Quality Assurance Staff has established a Quality Assurance Program (QAPD) [2.2] in accordance with the ANL-East Quality Assurance Manual [2.25]. Specifically the QAPD was evolved by applying the standards of NQA-1 [2.26] to the various activities involved in the production of the EBR-II PRA. Of the eighteen (18) Quality Assurance Elements (QAEs) continued in NQA-1, ten (10) are applicable to the EBR-II PRA Project. Most important of these are QAE 3.0 - Review of Technical Work and QAE 6.0 - Document Control.

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3.0 OVERVIEW OF EBR-II

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3.0 OVERVIEW OF EBR-II

3.1 Site Location

The Experimental Breeder Reactor II (EBR-II) is located in Bingham County, Idaho, at latitude 43° 35' 10" N, longitude 112° 40' 10" W. The average altitude is 5125 ft (1562 m) above sea level. The EBR-II is approximately in the center of the Argonne National Laboratory-West (ANL-W) portion of the Idaho National Engineering Laboratory (INEL).

The INEL, (Fig. 3.1), in east-central Idaho, has been a nuclear research site since 1949. This large site is a federal government complex of facilities which was initially called the National Reactor Testing Station, but was renamed the INEL in 1974. The reservation is a 572,000-acre (231,500 ha) government-owned reservation under the control of the U. S. Department of Energy (DOE). The INEL is established on a large high-desert, sagebrush covered and treeless plain in the north Rocky Mountains of the United States that has few significant terrain features. All activities at the INEL are limited to those permitted by DOE, and the establishment of permanent private residence is prohibited. The INEL reservation covers parts of four Idaho counties, with the 810-acre (328 ha) area administered by ANL-W being entirely in Bingham County. Underneath a large portion of the INEL is the large Snake River Plain Aquifer.

ANL-W encompasses an area in the southeastern corner of INEL (Fig. 3.2), and contains a number of nuclear facilities (Fig. 3.3) including HFEF, FCF, and EBR-II. EBR-II lies 10 miles (16 km) from the eastern boundary of INEL, and 3 miles (5 km) north of east-west U. S. Highway 20.

In the vicinity of ANL-W, the area is mostly undeveloped, the climate semiarid, and the population density low. The Snake River and the city of Idaho Falls, both approximately 30 miles (48 km) to the east are, respectively, the most significant natural and manmade features

INEL 4 4605

Fig. 3.1. Location of INEL in Relation to Surrounding States

a. Excepting state highways, all INEL Site roads are access-controlled.

Note: All highways are two-way roads.

INEL 4 4593

Fig. 3.2. Location of Primary INEL Facilities and Relationship to INEL Site Boundary

Fig. 3.3. ANL-W Site Plan

of the area. Other prominent natural features are several nearby volcanic-origin buttes to the south and southwest, and mountains to the west and north which are visible on a clear day. Sagebrush and similar dry-country vegetation covers much of the INEL, and irrigated farms are adjacent on lands not within the INEL reservation.

The terrain of the government-owned reservation is characteristic of a cool high-elevation desert plain, and has climatological conditions typical of such regions. The climate is dry, with moderately hot summers and cold winters. Moderate winds from the southwest are frequent. The predominant native vegetative species are mainly sagebrush, bitterbrush, buffalo grass and similar desert plants which provide a sparsely distributed groundcover.

The overall topography of the ANL-W site is that of a shallow, slightly concave ground depression in the low-rolling rises and depressions that generally characterize the terrain of this area of the INEL. The ground surface consists generally of unvegetated composite layers of interbedded volcanic rock and water-permeable sedimentary materials. The latter include alluvium, alluvial-fan deposits, lake-bed (lacustrine) and playa deposits, and wind-blown (aeolian) deposits. This soil material is only a few meters deep, covering the volcanic bedrock below. The voids and cracks in and between the layers of basalt and sedimentary material are readily traversed by precipitated and ground-source waters. The sandy soils are characterized as having high ion-exchange characteristics. Beneath ANL-W, the uppermost portion of the ground water layer of the Snake River Plain aquifer is between 600 and 700 ft (183 and 214 m) below the surface. The principal sources of groundwater recharge of the aquifer are from the streams and canals of the Birch Creek Valley and the Big and Little Lost River Valleys to the north and northwest, and from the impoundments and irrigation canals from the Mud Lake area to the northeast. The level terrain, absorbent soil characteristics and low annual precipitation preclude significant soil erosion.

3.2 Function and History of EBR-II

Experimental Breeder Reactor II (EBR-II) is a complete power-producing liquid metal fast breeder reactor, with rated thermal power of 62.5 MW and electrical output of

approximately 20 MW to on-site and utility loads. It includes a pool-type sodium-cooled reactor, sodium-cooled secondary-heat-transport system, and conventional steam turbine-generator for power conversion. In addition, a fuel reprocessing facility was originally available to complete the closed fuel-cycle operation. After the fuel-cycle concept was demonstrated in 1965-1968, the fuel reprocessing was replaced by fuel-evaluation programs. The reactor building and power plant are shown in Fig. 3.4 and the reactor operating floor is shown in Fig. 3.5.

The primary system is a submerged pool-type design. The entire reactor and primary coolant system are contained in a single tank (primary tank) (Fig. 3.6). There is no outside piping containing primary sodium, except for small-diameter piping connections to the primary purification cell, the radioactive sodium chemistry loop, and the fuel element rupture detector.

About 90,000 gal (341 m^3) of sodium are contained in the primary tank, and it has a normal operating temperature of 700°F (371°C). The sodium is pumped through the reactor at 9000 gpm ($0.568 \text{ m}^3/\text{s}$) and is heated to 883°F (473°C) at 62.5 MWt. The primary sodium, which is radioactive, is confined to the primary tank and is never mixed with the secondary system.

The secondary system is an intermediate closed loop between the primary system and the steam system. About 14,000 gal (53.0 m^3) of sodium are contained in the secondary sodium loop. The secondary sodium is circulated through the tube side of the intermediate heat exchanger at a rate of 5580 gpm ($0.352 \text{ m}^3/\text{s}$) at 62.5 MWt and is heated to 865°F (463°C). The heated sodium flows through the shell side of the superheaters and evaporators and is cooled to about 587°F (308°C).

The steam system consists essentially of a steam generator, turbine, condenser, feedwater heaters, and pumps. Feedwater at a temperature of 550°F (288°C) flows to the steam drum and then through the tubes of seven parallel-connected evaporators and is heated to a water-steam mixture at about 580°F (304°C). After the moisture is removed from the steam in the steam drum, the saturated steam flows through the tubes of two parallel-connected superheaters where

Fig. 3.4. Reactor Building and Power Plant

Fig. 3.5. EBR-II Operating Floor

Fig. 3.6. EBR-II Primary-system Components

it is heated to 820°F (438°C). The superheated steam then flows either to the turbine or directly to the condenser.

EBR-II completed its 25th year of operation in August 1989. Since the early 1960s, the facility has been compiling an outstanding operating record and a growing list of accomplishments. A review of EBR-II's operating history reveals a plant that has demonstrated high availability, stable and safe operating characteristics, and excellent performance of its sodium components. The record of high capacity factor documents the reliability and maintainability that can be achieved in a pool-type sodium-cooled fast reactor.

A major reason for the high capacity factor of recent years and the reliable operation of EBR-II has been the ongoing simplification of the plant. What started as a relatively complex plant to operate has become an increasingly simple, safe, and reliable plant today, and efforts are constantly under way to further improve and simplify its operation. A case in point is the simplification of the reactor shutdown system. The number of input trip signals to the shutdown system and the number of startup interlocks have been reduced from the original design. This was possible because of the inherent safety of the design and the increasing evidence of the benign behavior of EBR-II even under upset conditions.

EBR-II has proved to be a very versatile facility in which to perform various programs reliably and safely. Its mission has changed several times to meet the needs of the LMFBR base program. In its initial operation, EBR-II successfully demonstrated the feasibility of a sodium-cooled fast breeder reactor operating as an integrated reactor, power plant, and fuel-processing facility. In 1965, the role of EBR-II was changed to that of a steady-state irradiation facility. Since that time, over 11,000 individual experiments have been irradiated. The experiments have consisted of various fuel types (oxides, carbides, nitrides, and metal), cladding and structural materials, and reactor control materials. Peak burnups have reached nearly 20 at. % for mixed-oxide and mixed-carbide fuels, and nearly 19 at. % for metal fuels.

Since early 1978, the reactor has been operating for extended periods with failed (breached) fuel elements as part of a national program to determine the lifetime and survivability

of failed fuel in fast reactors, and the effects of such fuel on reactor operation. In preparation for this mode of operation, a cover-gas cleanup system (CGCS) was installed in 1977 to remove fission gases from the primary cover gas by cryogenic distillation.

The performance of the reactor and its primary cooling system has been very good. Throughout its operating history, the reactor has been highly stable and readily controllable. An ever-present and predictable negative power coefficient ensures kinetics stability under all foreseeable operating conditions.

Mechanical stability of the core components has been excellent. No indication of vibration or coolant-flow blockage has been observed, nor has any indication of gas entrainment been detected in the coolant. Post-irradiation examinations conducted on discharged core components (subassemblies, control- and safety-rod thimbles, etc.) have uncovered no evidence of excessive wear, fretting, ratcheting, vibration, or abrasion.

In 1976, a program of annual nondestructive inspection was begun on the most critical areas of the secondary-sodium and high-pressure-steam systems. Items examined included pipe welds, sections of steam and secondary piping, hangers, and valve-body casings.

The programs for shutdown-heat-removal testing and use of EBR-II as a test bed for components such as steam generators and pumps, a man-machine-interface program, demonstration of automatic control of the reactor, and the initiative for an Integral Fast Reactor are all examples of the various research missions of EBR-II.

3.3 Plant Description

3.3.1 Major Structures

A plot plan of the ANL-W site is given in Fig. 3.3. The buildings of direct relevance are 767, EBR-II reactor building, 768, the power plant, and 769, the sodium boiler building. Also worthy of note are 757, the cooling tower, 758, the electrical substation and 765,

originally called the fuel cycle facility, changed to Hot Fuels Examination Facility/South (HFEF/S) and recently renamed Fuel Cycle Facility (FCF).

The EBR-II systems are shown in simplified form in Fig. 3.7. The reactor, intermediate heat exchanger, control rods and safety rods all reside in the reactor building. Secondary sodium is piped from the IHX to the sodium-boiler plant through the yard piping. The sodium-boiler plant houses the secondary electromagnetic (EM) pump, seven evaporators, two superheaters, surge tank, steam drum etc. The power plant adjoins the reactor building and houses the primary pump motor generator sets, feedwater pumps and feedwater heaters, the turbine generator, the reactor control room, electrical supply cabinets and all other equipment. In addition there are a number of offices in the power plant. Figures 3.8 and 3.9 give detailed layouts of the power plant and the sodium boiler building. Both these buildings are steel framed and not designed to contain or confine any energy release. In fact the north and east walls of the sodium-boiler plant are designed with blowout panels in the event of a large energy release in the sodium boiler plant.

Access to the reactor building is gained from the power plant via a personnel air lock and an emergency air lock. There is also an equipment air lock for transfer of fuel between FCF and EBR-II, as well as large pieces of equipment. The reactor containment building is a steel shell, one inch thick with 12 inches of concrete on the inside surface. There is a polar crane used for transferring fuel. The primary tank, containing the reactor, 90,000 gallons of liquid sodium and the pumps and heat exchanger, lies below grade as shown in Fig. 3.10. The control rod drives are situated on the reactor floor and enter the core through the small rotating plug.

3.3.2 Reactor System

The EBR-II reactor core (see Fig. 3.11) consists of a 7-row hexagonal central core containing enriched uranium which is completely surrounded by 3 rows of stainless steel reflections and a 6-row radial blanket of depleted uranium. The blanket regions contain either depleted uranium or stainless steel and nickel. The core- and blanket-region materials are

Fig. 3.7. EBR-II Systems (Simplified)

Fig. 3.8. Power Plant - Location of Components

NOTE: ALL ELEVATIONS REFERENCED TO SEA LEVEL - 5000 FT.
 EXAMPLE: BOTTOM OF REACTOR BUILDING - 5077' -6"
 TOP OF REACTOR BUILDING - 5216' -3"

T/P-C-10059-A

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Fig. 3.10. Reactor Building

Fig. 3.11. EBR-II Core

contained in 637 subassemblies, which are contained in and supported by the reactor vessel. The reactor vessel, consisting of a grid plenum assembly, inner and outer reactor-vessel shells, and a reactor-vessel cover, is surrounded by a radial neutron shield and is submerged beneath approximately 10 ft of sodium. The reactor-vessel cover, which also contains neutron shielding, raises to permit fuel handling. It contains penetrations for control-rod drive shafts to provide reactor control and for the holddown and gripper shafts used for fuel handling.

The reactor vessel is constructed of an outer shell, an inner shell, and thermal baffles, all of Type 304 stainless steel. It is well shielded by the inner radial neutron shield and well protected thermally. The neutron shield is composed of a seven-layer radial shield and an axial shield in the reactor-vessel cover. Two layers of the seven-layer radial shield are between the two reactor-vessel shells. The other five layers cover the outside of the reactor vessel from below the grid plenum to the top of the vessel cover. The shield is composed of plain graphite blocks and borated graphite blocks, in stainless steel cans, that moderate and absorb neutrons passing through the reactor blanket. The blanket subassemblies also act as neutron and gamma shields to reduce radiation damage to the reactor vessel and the components in or directly outside the reactor vessel.

The grid plenum assembly, which supports the hexagonal subassemblies in the reactor vessel, has 637 apertures arranged to position the subassemblies in a tightly packed triangular lattice. The subassemblies are all outwardly identical: each has an upper adapter to permit handling (removal and insertion) by the same gripper and transfer devices; each has a hexagonal support body that contains the fuel or blanket material; and each has a lower adapter. The lower adapters have different bottom configurations to prevent positioning in the wrong region and have different coolant orifices to ensure the appropriate amount of sodium flow.

Operation of the reactor is controlled by vertical movement of fueled control rods with B_4C followers in thimbles in row 5 of the core region. Control-rod drive shafts, operating through penetrations in the reactor-vessel cover, are connected to the control rods. The control-rod drive shafts are connected to independently operated electromechanical drives mounted above

the primary-tank assembly on the small rotating plug. Twelve rods were originally in the core; currently two original control-rod positions contain test facilities. The control-rod drives are operated through 3 in. of travel or less at least once every 24 hours to check for the possibility of control-rod binding.

Because control rods are made of fuel with a poison follower the core is made critical by raising the control rods into the core, and scram is effected by dropping the control rods out of the core.

Safety rods located in thimbles in row 3 of the core region are connected to safety-rod connectors that extend up through the reactor-vessel grid plenum assembly. The connectors are connected through a safety-rod drive beam and safety-rod shafts outside the reactor vessel (but within the primary tank) to the safety-rod drive mechanism at the top of the primary-tank cover. The safety-rod drive system is a part of the reactivity control system; it provides negative reactivity when the reactor is operating, or when it is shutdown and the control rods are disconnected. In addition, use of the safety rods as shim rods, operating in conjunction with the control rods, has recently been implemented.

Sodium coolant, which surrounds the reactor, is pumped into the high-pressure and low-pressure chambers in the grid plenum of the reactor vessel (two inlets for each chamber, on opposite sides of the reactor vessel) from the two centrifugal main primary-sodium pumps, which operate in parallel. The high-pressure-coolant plenum is the source of coolant for the core and inner-blanket-region subassemblies. The low-pressure-coolant plenum is the source of coolant for the outer-blanket-region subassemblies.

The reactor's nuclear instrumentation consists of three wide-range nuclear instrument channels (A, B, and C) and two linear channels (7 and 7A). Channels A, B, and C each provides continuous neutron-flux monitoring, alarm functions, and automatic protective functions in the form of a reactor scram for short periods and over power conditions. Channels 7 and 7A each provide linear power indication for startup of the reactor and indication for reactor control

in the power range. The detectors for all the channels are in the reactor's J thimbles (J1, J3, and J4).

3.3.3 Primary Sodium System

The main function of the primary cooling system is to remove the heat produced in the reactor core by nuclear fission and transport it to the intermediate heat exchanger (IHX), where it can be transferred to the secondary cooling system. To perform this function, liquid sodium is pumped from the primary tank by two centrifugal pumps, through high pressure and low-pressure (throttled) piping to the reactor, and then to the shell side of the shell-and-tube IHX, from which it is returned to the primary tank. The entire primary cooling system, and the reactor vessel itself, are contained within the primary tank. Operation with the reactor vessel and other primary-system components completely submerged in the liquid sodium coolant ensures a high degree of reliability and safety of the primary system.

The primary cooling system, shown in Fig. 3.12, consists of the following major components: two primary-coolant pumps, a primary auxiliary pump, the IHX (primary side), reactor supply and discharge piping, and instrumentation associated with the system.

When the reactor is operating at the full design power of 62.5 MWt, the primary-system flow is 1064 lb/s (484 Kg/s) through the reactor vessel, the reactor inlet sodium temperature is 700°F (371°C), and the mixed mean outlet temperature is 883°F (473°C). Certain leakage paths from the cooling system to the primary tank have been designed into the system. These leakages occur: at the ball-joint disconnects of the primary pumps; at the low-pressure throttle valves; at the reactor inlet plenums; from the reactor-vessel cover, rod-drive penetrations, etc.; at the outlet plenum; and at the IHX inlet.

The primary-cooling-system instrumentation originally included 10 flowmeters, both electromagnetic (EM) and head type, on the high-pressure and low-pressure legs of each primary pump and on the reactor outlet piping to the IHX. Of these flowmeters, only three remain operable. Similarly, the original system had 12 pressure transmitters, two in each of the

Fig. 3.12. Primary Cooling System

primary-pump discharge pipes, one in each of the high- and low-pressure piping to the reactor, one in the high-pressure plenum, one in the low-pressure plenum, one in the upper plenum, and one in the reactor discharge pipe. Of these transmitters, only three remain operable. Of the many thermocouples and resistance temperature detectors installed throughout the system, most are still operable. The loss of instruments, particularly flowmeters, is a concern for EBR-II. Currently the number available is sufficient for plant protection needs, however, loss of more flowmeters would cause difficulties. Strategies to alleviate the problem include using pressure transmitters as flowmeters and adding pump speed sensors.

The EBR-II primary-coolant pumps are a matched pair of vertically mounted centrifugal pumps designed to pump liquid sodium from the primary tank, through in-tank piping, and through the reactor core. Each pump is mounted on a penetration nozzle in the primary-tank cover, and the pump structures are inside the primary tank. Figure 3.13 is a cutaway view of a primary pump.

The pumps take their suction directly from the 700°F (371°C) bulk sodium in the primary tank and discharge to a 12-in. (0.3 m) pipe nozzle that is integral with the pump assembly. Each pump has its own motor-generator (MG) set located outside the reactor building, in the power plant. Pump speed is controlled by varying the generated frequency to the pump drive motor through use of a magnetic slip clutch between the motor and generator of the MG set.

Heat from the radioactive primary sodium is transferred to the nonradioactive secondary-sodium system by the intermediate heat exchanger (IHX). The IHX is a shell and counterflow, single-tube-wall unit which operates completely submerged in the reactor sodium coolant. It is suspended from above by a shield plug and positioned within a well casing which is an integral extension of its primary-tank nozzle. The IHX and shield plug assembly, shown in Fig. 3.14, is approximately 5-ft (1.52 m) dia and 24-ft (2.32 m) long and weighs about 69,000 lb (31,364 Kg) when flooded with sodium. The total heat transfer surface is about 4800 sq ft (446 m²).

Fig. 3.13. Cutaway View of a Primary-coolant Pump

Fig. 3.14. Intermediate Heat Exchanger

The primary tank is a double-walled cylindrical tank about 27 ft (8.23 m) in diameter by 27-ft (8.23 m) high. It contains the nuclear reactor, the entire primary cooling system, and portions of the fuel handling system, all submerged in 700°F (371°C) sodium. The primary-tank assembly (see Fig. 3.15) consists of the primary tank and cover, biological and blast shields, an air baffle tank, and a primary-tank support structure. All penetrations into the tank are through the primary-tank cover, above the bulk sodium.

The primary tank and the support structure are completely independent of each other except at the top. The primary tank, containing the bulk sodium, and those components connected to the primary-tank cover are supported by six hangers, which are welded to each of the radial beams in the tank cover and bear on six high-strength steel rollers supported by the top support structure. This roller support system (see Fig. 3.16) is necessary to avoid high stresses caused by the differential expansion between the primary tank and the top support structure. As shown in Fig. 3.15, the basic top support structure consists of six radial beams, spaced at 60° intervals, and a large central ring. The basic structure of the primary-tank cover, shown in Fig. 3.17, is essentially the same as the basic top support structure, and the six radial beams of the support structure and six similar beams within the tank cover are coincident.

Containment of the 90,000 gals (341 m³) of radioactive primary sodium by the primary tank system is essential for the continued, safe, operation of EBR-II. The primary tank is of double-wall construction to provide maximum reliability. The tank consists of two vessels completely independent of each other, except for their common attachment to the primary-tank cover, from which they are suspended. Both tanks and the cover are constructed entirely of Type 304 stainless steel and of welded construction. There are no penetrations in the tanks except for the cover.

Because of their unique construction, the uniform temperature maintained on both tanks, and the common cover, mechanical and thermal stresses are very low. An argon atmosphere is maintained in the annulus between the tanks and thermocouples and a sodium leak detector probe is also provided in the annulus. Corrosion of stainless steel when exposed to argon or 700°C (371°C) sodium is nil. If a small leak in the main tank developed, sodium would slowly

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Fig. 3.15. Primary-tank Assembly

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Fig. 3.16. Roller Support System

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Fig. 3.17. Primary-tank Cover

fill the annulus and be contained by the outer tank. The volume of the annulus is approximately 990 ft³ or 7400 gal (28.0 m³). If a leak in the inner tank filled the annulus, it would drop the primary tank bulk sodium level about 22 in. (0.56 m), which would still maintain the reactor core under sodium. If a slight leak developed, there is a reasonable chance that it could be repaired or its effects minimized to permit continued reactor operation.

For shielding, the cover is filled to a depth of about 26 in. (0.66 m) with 3/8-in. (9.5 mm) -dia carbon steel balls. The entire tank is insulated to minimize heat losses from the bulk sodium. Numerous thermocouples are located throughout the tank to monitor temperatures of the sodium, the inert-cover-gas blanket above the sodium, the shells of the inner and outer vessels, and the cover.

3.3.4 Secondary-Sodium System and Balance of Plant

The secondary-sodium system removes heat from the primary-sodium system, via the intermediate heat exchanger (IHX), and transports it to the steam generator where it is used to produce steam. The secondary-sodium system consists of the main secondary-sodium loop, the secondary-sodium recirculation system and the secondary-sodium auxiliary and subsystems.

The function of the main secondary-sodium loop is to transport reactor-generated heat from the primary cooling system to the steam generator, where high-pressure, high-temperature steam is produced. At the full design reactor power of 62.5 MWt, the nonradioactive secondary sodium is circulated at about 5580 gpm (0.352 m³/s) (93% flow) through the tube side of the intermediate heat exchanger (IHX), where heat is transferred from the primary to the secondary sodium. The secondary sodium leaves the IHX at about 865°F (463°C) and flows through main-secondary-sodium-loop piping to the sodium boiler building, where it enters the shell side of the superheaters. It then flows to the evaporators, where it enters the shell side at about 800°F (427°C), to the surge tank at about 587°F (308°C), the main EM pump, and back to the IHX in the reactor building.

For reliability, the main secondary-sodium loop contains no valves. Unlike the primary system, most secondary-system components, including instrumentation, are accessible for maintenance or replacement.

The main secondary loop is fabricated of mainly two materials -- type 304 stainless steel, and 2 1/4% Cr-1% Mo ferritic steel. Generally, the stainless steel is used in the reactor building and yard piping, and the chrome-moly steel is used in the sodium boiler building.

During plant standby, the thermal driving head caused by heating in the IHX and cooling in the steam generator can produce a natural-convection flow of about 6%. In this condition the secondary sodium pump is operated by reversing the pump current. This technique is used to reduce, but not actually reverse, secondary flow during a shutdown so that only the decay heat is removed.

The total sodium volume contained in the secondary-sodium system, which includes the storage tank, dump piping and recirculating system, as well as the main secondary loop, is about 14,000 gal (57.9 m³) at 250°F (121°C)). The sodium volume in the secondary system has diminished by about 1800 gal (6.8 m³) over the years because of maintenance, repair, and sampling activities.

Sodium is circulated through the main secondary loop by an AC linear induction electromagnetic (EM) pump. This pump forms an integral part of the secondary piping system and circulates sodium through the IHX to the superheaters and evaporators where heat is transferred to water and/or steam. A conventional steam system, balance-of-plant converts the steam to electricity using a single turbine-generator. Steam is also used for site heating and feedwater heating in four separate heaters.

3.4 Reactor Shutdown System

The reactor shutdown system (RSS) is designed to automatically shut down the reactor in the event of equipment or operator malfunction or an environmental problem that could damage the reactor, injure personnel, or release contamination to the environment.

The RSS will cause the control rods (and, in some cases, the safety rods) to automatically scram to their least reactive position, to shut down reactor operation, if any of the selected plant parameters exceed established limits. During fuel handling, the safety rods will scram to their least reactive position if the fuel-handling shutdown circuit opens.

For redundancy, two shutdown circuits are provided. One is called string A, and the other string B. The purpose of string B is to provide reactor shutdown if string A fails to do so. String B is only used during reactor operation; it has no function during fuel handling.

String A provides two modes of operation: (1) reactor operation, and (2) unrestricted fuel handling.

Basically, the shutdown strings will provide plant shutdown in the event of two major problem conditions: (1) overpower, and (2) undercooling. Overpower conditions are indicated by high neutron-flux levels and high temperatures. Undercooling conditions are indicated by high temperatures and low coolant flow rates. Both strings also provide protection from other problem conditions, such as earthquakes or malfunctions of other essential plant equipment.

The reactor shutdown circuits consist of relay coils and contacts and keyswitch contacts, connected in series and series-parallel, to form a circuit between the two electrical buses, RCL-1 and RCL-2. Figure 3.18 is a simplified schematic diagram of string A and string B. The actual path of the electrical circuit depends on the mode of operation and the reactor power level. When all shutdown-string parameters are satisfied and their associated contacts are closed, string A control-power relays CPA, CPC-1, CP-1, and CP-2 will be energized, as will string-B relays CPB-1, CPB-2, and CPB-3.

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3.5 Decay Heat Removal

3.5.1 Shutdown Cooling System

Fission-product decay heat generated in the reactor during shutdown is normally removed from the primary system by the secondary-system sodium. However, if the secondary-system sodium is not flowing through the intermediate heat exchanger in the primary tank, or if the reactor cover is raised, the decay heat will be transferred to the primary bulk sodium. Although the bulk sodium has a large thermal capacity, an undesirable temperature rise would occur if decay heat continued to be transferred to the bulk sodium over an extended time period. Under these conditions the shutdown cooling system would transfer the excess heat from the primary system to the air outside the reactor building and would keep the primary bulk sodium within the acceptable operating temperature limit of $700^{\circ}\text{F} + 10^{\circ}\text{F}$.

The shutdown cooling system is one of the EBR-II sensitive systems and removes a total of 360 kW of heat when the temperature of the bulk sodium in the primary tank is 700°F . The decay heat produced in the reactor after shutdown decreases continually with time in a nonlinear manner. For this reason, a nominal value for decay-heat removal could not be selected as a reference value in the design of the shutdown coolers. Instead, the total integrated amount of decay heat produced in the reactor from shutdown to various times after shutdown was considered in the shutdown-cooler design.

Two identical cooling loops comprise the shutdown cooling system (see Fig. 3.19). Each consists of a shutdown-cooler plug, shutdown-cooler box assembly, and associated piping equipment. The system provides heat transfer through the natural convection of the NaK eutectic alloy (22% Na and 78% K). In each closed loop, flow occurs between a heat-exchanger plug, located in the primary tank and mounted in the L-1 or L-2 nozzle and heat-exchanger core in a box assembly on the outside wall of the reactor

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Fig. 3.19. Shutdown Cooling System

building. NaK flow varies from 2 to 10 gpm (0.0076 to 0.038 m³/min), when the air dampers (described below) are closed and the bulk sodium is at 700°F (371°C), to 40 to 70 gpm (0.15 to 0.76 m³/min) when the bulk-sodium temperature increases to 710°F and the air dampers are open.

The natural-convection flow of NaK in the closed-loop system is completely independent of any power source and functions whenever the bulk sodium in the primary tank is hotter than the outside air. A small surge tank, which is part of the box assembly and is blanketed with argon gas, permits thermal expansion in the system. Decay heat is transferred from the shutdown-cooler NaK to the atmosphere by the natural convection of air tube is controlled by dampers above and below the heat exchanger and by a sheet-metal chimney above the upper damper. The NaK cooling system, external to the primary tank, is instrumented with thermocouples, electromagnetic flowmeters, and level indicators.

The shutdown cooling system has been in operation since 1963 and has provided very reliable operation. Only minor maintenance has been required, mainly on the air dampers and actuators.

3.5.2 Shield Cooling System

The shield cooling system provides forced-air cooling of the biological shield, the blast shield, and the rotating plugs. It circulates air between these components and cooling and filtering components through ductwork embedded in concrete or exposed. As shown in Fig.3.20, there are actually two separate but identical cooling and filtering systems available for use. One is operating while the other is on standby. These systems are operated alternately to equalize fan operating times and to allow maintenance activity during standby periods.

The shield cooling system is part of the reactor building ventilation system. It operates at a pressure slightly less than that of the building atmosphere. This permits a certain amount of in-leakage and simplifies the cooling of certain areas in the shield that cannot be connected to a closed system. The top structure of the primary-tank support system, the rotating shield plugs, and the top surface of the primary-tank cover are cooled primarily by air drawn from the building atmosphere. This air is drawn into the shield cooling system through ducting in the rotating plugs, the annuli between some of the primary-tank nozzles and the biological-shield sleeves, and around the top cover of the primary tank through air ducts in the top biological shield. It is then joined by air which has circulated through the air baffle tank, an annular tank surrounding the primary tank. The blast shield, the radial and lower biological shield, and the primary tank are cooled by the recirculated, cooled air passing through the air baffle tank. The air flow is then split into two paths, one to the exhaust stack and the other through coolers and then into the shield cooling system.

The system recirculates 15,000 cfm ($7.08 \text{ m}^3/\text{s}$) of air and exhausts 5000 cfm ($2.36 \text{ m}^3/\text{s}$) from the reactor building. An amount equal to the exhaust, 5000 cfm ($7.08 \text{ m}^3/\text{s}$), is drawn into the system from the reactor building. The total heat load in the system is about 506,000 Btu/hr (148 kW).

The recirculated air is filtered by cleanable filters before it enters the cooling coils. The exhausted air passes through high-efficiency particulate-air (HEPA) filters, referred to as "absolute" filters, before it is discharged to the suspect exhaust stack. The exhaust from the shield cooling system combines with the exhaust from the thimble cooling system and the combined exhaust is discharged from the reactor-building containment through a penetration. One valve outside the reactor building and two mechanically interlocked valves inside the building are controlled by the reactor-building isolation system. If the two valves in the exhaust duct to the suspect exhaust stack are closed under isolation conditions, the third valve opens to discharge the air back to the basement of the building.

Thermocouples in the rotating plugs monitor temperatures to avoid causing distortion of the plugs, which could cause binding of the rod drives or other equipment that penetrate the

plugs. Dampers adjust the amount of air flowing through the rotating plugs. Thermocouples are also in the blast and biological shields and in the cooling-air stream. Some of these thermocouples are connected to high-temperature alarms. Low air flow in the operating system also will cause an alarm.

Under normal operating conditions, power for the shield cooling system is supplied by the 480-V distribution system. Under emergency conditions, the 100-kW diesel generator supplies power for the exhaust-air portion of the system, and the 400-kW diesel generator supplies power for the recirculating portion of the systems, including the power for the refrigeration system that provides the cooling.

The shield cooling system is comprised of the following subsystems, each with its own components. The components of the regular (40-ton) refrigeration system are a compressor, a water-cooled condenser, a cleanable-filter bank, and an evaporator (cooling coil). The standby (20-ton) refrigeration system has two compressors, two water-cooled condensers, two evaporators, and cleanable-filter bank. The recirculating air system has two automatic dampers, three manual dampers, and a supply fan with a capacity of 14,420 cfm at 58°F and 12.3 psia. The fan is powered by a 20-hp electric motor. These components are duplicated in the standby recirculating-air system. The exhaust air system has a HEPA filter bank, two manual dampers, one automatic damper, and a fan with a capacity of 7730 cfm at 96°F and 14-in.-H₂O static pressure. A 30-hp motor drives the fan. These components are duplicated in the standby exhaust air system.

3.6 Unique Features of EBR-II

EBR-II has a variety of unique features that directly enhance reactor safety through relatively benign responses to accident initiating events. These features can all be grouped under the umbrella phrases "passive safety" or "inherent safety" and refer to the system's ability to survive potential accident sequences without core damage via natural features of the materials and design, rather than through the action of any active mitigating function. These attributes are not just theoretical concepts. EBR-II has carried out an extensive program of plant testing

which demonstrated the performance of these passive safety features under a variety of accident conditions. The Shutdown Heat Removal Test series (SHRT) culminated in the Inherent Safety Demonstration Tests performed in April 1986 during which an Unprotected Loss-of-Normal Power and an Unprotected Loss-of-Heat Sink were performed without incurring any damage and demonstrating passive neutronic shutdown capability.

The essential characteristics which make this behavior possible are use of sodium coolant with its good heat transfer capability, metallic uranium fuel with its high thermal diffusivity and the pool design which makes available a large heat sink and inherent negative reactivity feedbacks.

Specifically, the features of EBR-II which lead to this benign response to various accident conditions are as follows:

Large Excess Heat Capacity

The design of EBR-II is such that in the primary tank, sodium and structure there is a large heat capacity with the result that, in the event of loss of all decay heat removal capability, it takes several days at decay heat levels to increase temperatures to fuel or structural damage levels.

Large Heat Removal Capability

The heat removal capability of the balance of plant exceeds the heat generated following a scram and the plant would cool down if measures were not taken to prevent this happening. A unique operational consideration at EBR-II involves overcooling the system rather than overheating. This cooldown is driven by natural circulation of the secondary sodium, and the cooldown rate is easily controlled by running the secondary pump backwards in order to retard the natural circulation flow. Cooldown is undesirable because of the time required to return to temperature and possibly stressing of components. There is no safety concern associated with cooldown following scram.

Natural Circulation Decay Heat Removal

Removal of the decay heat from the fuel does not require that primary pumps be operating. The ability to remove the decay heat using natural circulation of the primary coolant has been demonstrated experimentally in the Shutdown Heat Removal Test (SHRT) series.

Reactivity Feedbacks

The reactivity feedbacks in EBR-II are such that any heating leads to power reduction. Many different feedbacks contribute but because of the small core size the dominant feedback is the sodium density effect. Various unprotected accidents have been run as experiments in EBR-II to explicitly demonstrate this effect.

Metal Fuel

EBR-II uses metallic uranium fuel enclosed in stainless steel clad. Metal fuel has a high thermal diffusivity which means that it responds faster under accident conditions than oxide fuel.

In addition, because EBR-II was designed in the 1950's as a research reactor, throughout the design of EBR-II there are conservative features which were included because of lack of knowledge concerning possible accidents. For example the steam generator tubes are double-walled with a double walled tube-sheet thus requiring two failures to bring sodium and water into contact. Furthermore there is a detection system to monitor for such leaks.

One interesting feature of EBR-II is the leakage paths between primary system pipes and plena and the bulk sodium in the primary tank. These allow for natural circulation flow to remove fission product decay heat from the core even if the main coolant circuits were to be disabled for some reason.

3.7 Whole Plant Experiments in EBR-II

The thermal hydraulic testing program at EBR-II was initially conducted in support of the continued safe and reliable operation of EBR-II and evolved into an experimental program contributing to the design assessment of advanced LMRs. The goals of the experiments have been to understand the detailed response of EBR-II to a wide variety of upset conditions both with and, ultimately without scram. The full scope of the testing program is described in detail in ref. 3.1. The following is a brief description of the experiments and their significance, both in general and more specifically to the Probabilistic Risk Assessment.

Early testing concerned natural circulation phenomena at either decay power or fission power levels, [3.2]. The next area addressed was the transition from forced flow to natural circulation from a wide variety of conditions with different sequences of scram, primary pump trip, and secondary pump trip times.

Loss-of-forced-flow transients were conducted from full primary flow at hot-standby conditions, from auxiliary pump flow at decay power levels, and from various reduced levels of fission power and primary pump flow. The analysis of these test results clearly demonstrated that transients initiated from conditions in which relatively normal temperature gradients existed in the system (i.e., power-to-flow ratios close to unity) lead to a smooth and benign transition to natural circulation [3.3]. However, as predicted in the original Hazards Summary Report [3.10], when the transient imposed resulted in a significant degradation of the thermal head in the primary circuit prior to the loss of forced flow, a substantial and temporarily sustained total reversal of flow occurred in the lower power regions of EBR-II (i.e., the radial blanket and reflector regions) [3.4]. No damage to the blankets was predicted or observed.

The SHRT Program was developed jointly by thermal-hydraulic specialists from EBR-II and Department of Energy contractor organizations, largely in support of the design of LMR plants in the U. S. The objectives of this program were to:

1. demonstrate passive removal of decay heat following reactor shutdown by natural circulation of primary sodium coolant;
2. demonstrate passive reactor shutdown following a loss of forced circulation without scram (LOFWS);
3. demonstrate passive reactor shutdown following a loss of balance-of-plant (BOP) heat sink without scram (LOHSWS); and
4. provide test data for validating computer codes used in the design, licensing, and operation of LMRs.

Each of the total of 58 tests in the SHRT Program was one of the five following types:

- LOF/scram to natural circulation
- Scram with delayed LOF to natural circulation
- Reactivity feedback characterization
- LOFWS
- Balance of plant transients
 - dynamic frequency response test
 - LOHSWS
 - steam drum pressure reduction

3.7.1 LOF/Scram to Natural Circulation

The essential parameters that were studied in this sequence are the initial power and primary flowrate and the trip/non-trip of the secondary pump. The first test, which duplicates one that was conducted earlier was initiated from 36% of full power and 40% of full flow by tripping the primary pumps. In addition, several different combinations of primary pump rundown times and delays in the secondary pump trip were studied. In all cases, the final state of the plant is natural circulation in both the primary and secondary systems. The subsequent test initial conditions were power and primary flow levels of 75%/100%, 75%/75%, and 100%/100%, respectively. For each of these test conditions (with the auxiliary pump turned off), the secondary flow was in the first case slowly reduced to about 20% or in the second case permitted to rapidly coastdown to natural circulation. Thus, the final test in this group directly simulated a total loss of pumping power (normal and emergency) to the plant while operating at full power and flow.

3.7.2 Scram with Delayed LOF to Natural Circulation

Most of these tests were initiated from normal full power and flow where the first step was to scram the reactor while maintaining either full primary flow or auxiliary pump (~ 5%) flow. After a prescribed delay during which time the primary system approached isothermal conditions, the primary pumps (or the auxiliary pump) were tripped and the flow coasted down to natural circulation. The effect of the level and rate of coastdown of secondary flow was also studied. During these transients, the coolant flow in the radial blanket-reflector region either temporarily stagnated or reversed due to the adverse thermal conditions delaying the transition to upward natural convection.

3.7.3 Reactivity Feedback Characterization

These tests were designed to provide measurements of the dynamic plant response to flow and inlet temperature perturbations in order to permit validation of reactivity feedback models. A series of thermal-hydraulic disturbances was imposed upon the core,

maintaining fixed (or precisely changed) control rod positions, in order to measure the resulting power changes and thereby to compute the effective reactivity feedbacks through an inverse kinetics calculation. The first type of disturbance was a change in primary flowrate, holding the reactor inlet temperature and control rod positions fixed. This causes the reactor ΔT to change initially, affecting the reactivity balance, which drives a power change. Initial power levels of 25%, 50%, and 70% were used.

The second type of disturbance was a change (either decrease or increase) in the reactor inlet temperature (caused by changes in the secondary flowrate), while the primary flowrate and control rod positions were fixed. Initial power levels of 25%, 50%, and 70% were used. A variation of the two types of disturbances employed was a flowrate or inlet temperature disturbance followed by a change in control rod position sufficient to return the power to its initial value. The results of some of these feedback tests are described in [3.5].

3.7.4 Loss of Flow without Reactor Scram

This sequence of tests involved coolant pump trips without permitting a reactor scram. The initial tests used different combinations of rate of primary pump coastdown, trip or non-trip of the secondary pump, and operation of the auxiliary pump on emergency power, battery power, or not operated at all. These tests were initiated from various power and flowrates, culminating in 100% power and flow. The mildest transients, in terms of maximum core temperatures, were conducted first, with the severity gradually increased. Included among these tests were an unprotected LOF to natural circulation from full power as well as an unprotected station blackout (loss of normal and emergency a-c electric power without scram), [3.6].

3.7.5 Balance of Plant Transients

3.7.5.1 Dynamic Frequency Response Tests

These tests were designed to generate data over a wide disturbance frequency range that can be used to validate whole plant dynamic simulation models. There were types of cyclic tests in which a relatively small periodic disturbance was added to either control-rod-inserted reactivity or secondary loop electromagnetic pump voltage. This periodic disturbance consisted of a fundamental frequency and a number of harmonics. The propagation and attenuation of these frequencies in the form of power, temperature, flow, and pressure disturbances was measured throughout the plant. Spectral analysis was applied to both the measured output and the simulation results, providing transfer functions for each. A comparison of the two sets of results (measured vs. calculated) is a direct measurement of the efficacy of the calculational models.

3.7.5.2 Loss of Heat Sink Without Scram

This test had as its objective the demonstration that the reactor will passively shut itself down following a total loss of heat sink, including the emergency decay heat removal system. Experimentally, this was accomplished by essentially stopping the secondary flow (including natural circulation) without permitting the reactor to scram, and maintaining full primary flow.

These particular conditions were chosen to umbrella all possible loss of normal heat sink transients. In this case, the loss of heat removal from the primary system caused the reactor inlet temperature to increase and thus the reactor power to decrease due to the negative reactivity feedbacks. Although the inlet temperature rose substantially, the resulting power reduction was sufficiently large and rapid to cause a reduction in the outlet temperature so that the core ΔT actually decreased to almost zero and the entire primary system approached an isothermal condition [3.6].

3.7.5.3 Steam Drum Pressure Reduction

Two tests were run which involved a rapid opening of the steam bypass valve to the condenser and causing a rapid steam depressurization. The primary and secondary flowrates were held constant and reactor power and temperatures were allowed to freely respond. The decrease in the steam drum pressure and in its water saturation temperature reduced the secondary sodium temperature exiting the evaporators. This reduced secondary sodium temperature overcooled the primary sodium in the IHX, thus reducing the reactor inlet temperature after a time delay. Due to the negative reactivity feedbacks, the reactor power slowly increased to meet the increased steam power demand. The reactor and secondary loop ΔT s were fixed by the reactor and secondary flowrates and the power demand [3.7]. These tests represent ways in which Balance-of-Plant faults could lead to reactivity insertion and hence an overpower transient.

3.7.6 Overcooling Transients

One potential source of reactor overpower and fuel overheating arises if the primary system is overcooled. If there is no scram, the negative inlet temperature coefficient combined with the falling inlet temperature leads to increasing power. This condition was investigated in the PICT and IOT experiments [3.8, 3.9]. It was demonstrated that variations in the primary flow rate were accommodated by power level changes so that the power-to-flow ratio remained at less than 1. Overcooling by faults on the secondary side, such as the increase of secondary flow to the maximum possible, will lead to a power-to-flow ratio of greater than 1. In this case the transient is limited by the capacity of the secondary pump to remove heat and the accident terminates without any damage to fuel.

3.7.7 Summary

Throughout its long and successful operating history the response of EBR-II to all conceivable forms of off-normal conditions has been analyzed and wherever practical this behavior confirmed by experiment. The implication for the EBR-II PRA is that the outcome of

system transients can be predicted with a high degree of confidence, allowing for a finer discrimination between degrees of damage than might normally be applied.

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4.0 SPECIAL FEATURES OF EBR-II

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4.0 SPECIAL FEATURES OF EBR-II

4.1 Passive Safety Features

The term passively safe or inherently safe has been used to characterize various reactor designs in recent years. The particular features which lend themselves to this description have no mechanical moving parts and rely upon natural phenomena to provide the mitigating functions rather than active systems which are asked to perform, on demand, in response to a given stimulus.

EBR-II has many design features which contribute to the safety of the reactor which can be termed 'passive' or 'inherent'. The role that these features play are reflected in the PRA. In an upset condition, some of these design features provide times for operator recovery on the order of eight hours. One example of this type of feature is the large heat capacity of the system. Other contributors to passive safety are the inherent physical limits to system performance which appear as constraints to the severity of the associated initiating event, e.g., a physical limit on pump over speed. The final type is the use of natural phenomena to mitigate accidents, e.g., natural circulation cooling of the core and reactivity feedbacks limiting the temperature excursion in unscrammed transients.

The classic functional approach to reactor safety and accident sequence mitigation utilizes a fundamental safety philosophy which can be described as follows: first, scram the reactor to ensure neutronic shutdown and second, provide means of removing decay heat. In LWRs, the front-line systems designed to prevent core damage are active, on-demand systems; if these systems fail, there is a high probability of the occurrence of core damage. In pool-type LMRs, if all active systems fail, passive systems provide sufficient shutdown and heat removal such that there is a low probability of even minor core damage.

EBR-II is designed with the same fundamental safety philosophy as LWRs but with a different architecture. Just as in LWRs, a failsafe scram system is provided in EBR-II to shut

off the nuclear chain reaction; if the mechanical scram system fails, neutronic shutdown is effected by reactivity feedbacks. Thereafter the decay heat removal can be carried out by passive means, utilizing natural circulation. In the primary tank, if the primary pumps are not operating, natural circulation transfers heat from the core to the bulk sodium. Heat can be extracted from the tank sodium in several ways. The preferred heat transport path is through the secondary sodium loop. In fact, if the secondary sodium is flowing under natural circulation conditions, the rate of extraction of heat is too large and the system will cool too rapidly. This is prevented by retarding the natural circulation flow by reversing the field of the secondary electromagnetic pump. Under certain conditions heat is not removed through the secondary system and the decay heat is then removed by two shutdown coolers which are passive systems.

Table 4.1 lists the various design features which can be considered as part of the passive safety position for EBR-II and how they are reflected in the PRA. A number of these design features serve to reduce the frequency of initiating events by removing or reducing the consequences of a single failure. An example of this is the passive accommodation of sodium after a primary vessel leak by the guard vessel which therefore requires a coincident leak or breach of the guard vessel before a core threatening sequence can occur.

In EBR-II the sodium coolant is at a low pressure and therefore the stresses in the coolant piping are very low and the probability of crack-initiated piping failure is much less than a high pressure system such as an LWR. Furthermore any pipe break results in a loss-of-flow rather than any loss-of-sodium because of the submerged primary system. The result is that the only events with the potential for loss of a significant amount of sodium are very low probability events such as rupture of the primary tank and guard vessel. This is discussed further in Section 5 and Appendix D.1.

Other design features ensure that for anticipated events, e.g., a fault on a single coolant pump, the core will not be threatened even in the event of total failure of the safety systems. This is ensured by an appropriate design of the pumps, with a suitably long coast down time. In the event of a fault affecting both coolant pumps (such as loss of normal power) the coastdown is sufficiently long to preclude core damage even if all protective systems fail.

Table 4.1. EBR-II Design Features Contributing to Passive Safety

<u>Design Features</u>	<u>Effect on EBR-II PRA</u>
Low pressure system	Low initiating event frequency
Guard vessel	Limits severity of initiating event
Submerged primary system	Limits severity of initiating event
Inherent limits on pumps	Limits severity of initiating event
Pump coastdown time	Characteristic of initiating event
Large heat capacity and heat removal capability	Success criteria for decay heat removal
Double-walled tubes	Low initiating event frequency
Natural circulation	Sequence analysis
Reactivity feedbacks	Sequence analysis

Similarly the capacity of the pumps has been chosen such that the maximum overspeed is not sufficient to lead to damaging transients.

The large heat capacity of the primary tank sodium and associated structure ensures that, in the event of some form of loss of heat removal capability, there are long times available to act and there are several passive systems capable of removing the heat.

Use of double-walled tubes in the steam generators ensure that the probability of any sodium-water reaction is reduced to a minimum. This contrasts with designs in place at commercial and prototype LMRs which have less conservative designs.

Two natural phenomena are relied upon in the design of EBR-II, and to a certain extent in all LMRs, to achieve passive safety. Natural circulation of the sodium ensures passive accomplishment of the decay heat removal function and negative reactivity feedbacks ensure passive shutdown in a wide variety of anticipated transients, including Loss-of-Flow, Loss-of-Heat-Sink and reactivity addition events. These phenomena are discussed in detail below.

4.2 Natural Circulation

The passive safety features that lead to the benign response to protected transients are natural circulation heat removal from the core and the large heat capacity of the primary tank, sodium and associated structures.

Normal forced cooling of the core is carried out by two centrifugal pumps pumping liquid sodium through the core. If the pumps lose power, or for any other reason cease to operate, the cooling of the core is carried out by natural circulation. Hot sodium generated in the core region rises through the flow paths available to the tank while cold sodium, at tank temperature, replaces it in the core. By this means the core heat is transferred to the bulk sodium in the reactor tank. The large thermal capacity in the tank ensures a very slow rate of temperature rise in the tank and very long times for heat removal by the secondary system, shutdown coolers, shield cooling and heat losses.

It is necessary to ask how natural circulation might be defeated. Firstly, normal natural circulation cannot function if critical flow paths are blocked or if there is excessive flow resistance. Secondly, natural circulation cannot be sustained if the necessary thermal conditions are not preserved. No arbitrary unspecified means of defeating natural circulation has been hypothesized and it has not proved possible to establish any mechanism that can defeat natural circulation, other than massive structural failure which alters the geometry totally, disrupting the flow paths.

As discussed in Section 5, blockage of the large inlet and outlet pipes are considered to be very low frequency events, as there is no source of debris large enough to completely block pipes apart from massive structural failure. The auxiliary pump is on the outlet Z-pipe and conceivably could retard natural circulation if the head were reversed. The pump is tested at reactor startup for each run and no mechanism for reversing polarity has been identified. There are no valves on the main inlet and outlet flow paths which could be misaligned. The throttle valve on the low pressure piping is locked fully open and never manipulated. Gas bubbles could conceivably form by disentrainment, for example, in the Z-pipe but this is not possible at the flow velocities found under forced flow conditions. Furthermore EBR-II is designed with a variety of flow leakages which bypass the main natural circulation path (see Fig. 4.1). These leakages are sufficient to allow cooling of the core under decay heat conditions.

Extensive experimentation has been performed in EBR-II demonstrating the ability of natural circulation to remove the decay (shutdown) heat. These are the Shutdown Heat Removal Tests (SHRT) summarized in reference [4.2].

In summary, no credible means of defeating natural circulation in EBR-II has been identified. Therefore the existence of natural circulation will be assumed in sequence calculations. The thermal effects of the transition to natural circulation will be included in the evaluation of accident response.

Fig. 4.1. Design Flow Leakage Paths in EBR-II

4.3 Reactivity Feedbacks

The passive safety features that control the response to Anticipated Transient Without Scram (ATWS) events are the reactivity feedbacks. Although they are sometimes referred to as a shutdown capability, reactivity feedbacks alone cannot drive the reactor subcritical. They do, however, tend to reestablish a stable state (which may be essentially zero fission power) where heat production and removal are in balance.

The passive feedbacks are temperature-driven reactivity changes - expansion, bowing and Doppler broadening. They can be divided into three categories of importance [4.3,4.4]. Sodium expansion reactivity is by far the strongest feedback. Because of the small core size, the sodium expansion reactivity has a large leakage component and is negative everywhere. The second category consists of fuel axial expansion, control driveline expansion, steel expansion and radial expansion/bowing. Doppler broadening is the least important of the feedbacks, accounting for only a few percent of the total; its smallness is due to three factors: the hard neutron spectrum, the high fuel enrichment and the small temperature gradient across the metal fuel. All these feedbacks are always negative except for bowing, which can be positive or negative, depending on the particular core loading. Since bowing reactivity never dominates, the net feedback reactivity is always negative. Thus if an accident initiator makes the core temperature rise, the resulting feedback reactivity tends to reduce the reactor power and thereby reduce the fuel temperatures.

The reactivity feedbacks of EBR-II have proven to be stable over the years, despite the differences in core loadings dictated by its role as an irradiation facility. Since 1977 the measured prompt power coefficient has varied 21% and the measured average power coefficient has varied 36%. For four widely different loadings, the calculated average power coefficient varied by only 10% while the fraction of fueled subassemblies that were non-metal experiments varied from 0 to 19% [4.4]. This stability occurs because the coolant volume fraction is nearly constant, independent of core subassembly type, and the coolant expansion dominates the passive feedbacks. As a consequence of the stability, it is reasonable to use a generic metal-fueled

loading in the sequence analyses. The influence of experimental subassemblies on EBR-II feedbacks can be treated as one source of feedback uncertainty.

The reactivity feedbacks are simply a manifestation of the laws of physics. For example, it is inevitable that the sodium will expand as its temperature rises. This will necessarily be a negative reactivity effect. The only way the feedback influence could be negated is if an accident initiator significantly modifies the core geometry. Only a severe earthquake appears capable of this. Thus, except in the presence of such severe external initiators, reactivity feedbacks are assumed to be present when performing sequence analyses.

Reactivity feedbacks will fail to prevent damage if they are not strong enough to keep temperature rises caused by the accident initiator within safe bounds. The failure analysis for passive safety consists of predicting the effect of feedbacks on the transient outcome and evaluating the reliability of the prediction. The predictions for EBR-II should be more reliable than predictions made for most LMR designs, since there is a wealth of experimental data related to the feedbacks, including the Inherent Safety Demonstration Tests. Data from these tests have been used to test calculational predictions. This is discussed further in Section 5.5.

The results of calculations presented in Sections 5.4 and 5.5 and supported by a large number of plant experiments, including direct demonstration of many transients, is summarized below.

- Loss of Heat Sink - In the event of a Loss of Heat Sink the reactor temperatures reduce uniformly as directly demonstrated by the SHRT B302R experiment [4.7].
- Pump Overspeed - The power-to-flow ratio does not exceed unity and the temperature does not increase. This result has also been directly demonstrated in EBR-II [4.8].
- Loss of Flow - The SHRT plant experiments directly demonstrated that no damage occurs if the flow coastdown is long enough [4.7]. However,

each initiating event must be considered on its own merits because different coastdown times can occur.

- Loss of Normal Power - This transient is a particular example of a loss of flow with long coastdown time combined with a Loss of Primary Heat Sink simulated by the SHRT 45 experiment [4.7]. In this case, the peak coolant temperatures are close enough to boiling to require detailed analysis of the effect of uncertainties in the feedback, but no damage occurs with a nominal prediction.
- Reactivity Insertion - For the case of slow reactivity insertions, feedbacks limit the rate of rise of power, and damage will result only if the total reactivity insertion is large enough. Therefore, the reactivity insertion events are classed according to the total reactivity inserted.

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5.0 ACCIDENT SEQUENCE ANALYSIS

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5.0 ACCIDENT SEQUENCE ANALYSIS

This section contains the description of the accident sequence analysis. In Section 5.1, all the relevant initiating events are identified and categorized. Section 5.2 gives the description of the quantification of the initiating event frequency. Therefore, at the end of Section 5.2 there is the complete list of initiating events for the EBR-II PRA, the associated frequency and the basis for the estimation.

Section 5.3 describes the event trees development and includes the discussion of safety functions, their success criteria and the associated criteria for plant transient definition and damage category assignation. At the end of this section a full description of the event trees for each initiating event is available together with the damage assignation.

Sections 5.4 and 5.5 give further details on the behavior of EBR-II for certain key categories of initiating event. These calculations form the basis for the damage assignation of Section 5.3. Because the damage analysis of Section 5.4 is a best estimate analysis, the uncertainty in the calculations is analyzed in Section 5.5.

5.1 Initiating Event Identification

5.1.1 Introduction

This section discusses the identification and classification of Initiating Events (IE) for EBR-II. Initiating events are defined to include any event that leads to an accident sequence. Most initiating events, combined with safety system failures, have the potential to lead to some degree of core damage, (since even a routine shutdown involves challenges to plant safety systems). However, there are some events that can be shown to lead to a transient which will not damage fuel but are sufficiently important that they will be retained. An example of this type of event is a loss of normal power with the highly unlikely failure to scram, is

predicted not to lead to fuel damage on a best estimate basis but the failure to scram would be occasion for serious concern. There are also some very rare initiating events which lead to damage, irrespective of safety system response. An example of this type of event is a very large, rapid reactivity insertion.

The process of core refueling is treated as an independent analysis and is documented in Appendix A. Events that occur while the system is shutdown are treated to the extent that for the shutdown decay heat removal process an extended mission time is considered, either 14 or 45 days (see 5.2). During this mission time failures of support systems and shutdown heat removal capability are considered. All other events are considered to be benign or not credible. For example, once the system is shutdown, a flow transient has no effect (benign) and rod induced reactivity insertions are not credible since the control rods are unlatched. Accidents related to the transfer of subassemblies from EBR-II to HFEF (Hot Fuel Examination Facility) are not considered as they affect only one subassembly at a time, outside the reactor, and cannot lead to widespread core damage. Accidents of this type would be incorporated into an extended PRA or containment study because of the risk to on-site workers.

5.1.2 Classes of Initiating Events

EBR-II is one of two operating Liquid Metal Reactors (LMRs) in the USA, the other being FFTF. Other operating LMRs exist in the UK (PFR), France (Phenix and Super-Phenix), USSR (BOR-60, BN-350, BN-600), Japan (JOYO) and other countries. There are detailed design differences between EBR-II and other LMRs and significant differences compared to LWR reactors (PWR and BWR) as discussed in Section 3. EBR-II has been extensively characterized and studied over its 26 year operating history and a good basis for initiating event selection exists in the safety documentation for EBR-II [5.1]. In addition, despite the design differences, there are enough design similarities among all LMRs so as to make the initiating event lists used in the CRBRP PRAs, [5.2] and [5.3], JOYO, [5.4], and an EPRI study of comparative risk between LWR and LMFBR (LMR), [5.5], directly applicable. A review of the EBR-II UORs since 1976 was undertaken to try to identify any other plant-specific initiating events and to provide data for quantification.

The generic classes of initiating events identified by the review of the EBR-II safety documents and the analysis of other LMRs are listed in Table 5.1. Each class was then developed into a comprehensive list of generic and EBR-II specific potential causes incorporating the events identified in [5.1], [5.2], [5.3], [5.4] and [5.5]. The resulting list is given in Table 5.2.

In the following sections each class of initiating event is discussed, the EBR-II-specific initiating events identified, and the basis for the quantification of the frequency is given.

Table 5.1. Generic Classes of LMR Initiating Events.

Loss of Flow
Reactivity Insertion
Loss of Primary Heat Sink
Overcooling
Loss of Decay Heat Removal Systems
Transients (including shutdown)
Loss of Sodium
Core Support Structural Failure
Support System Failures
Local Faults
External Events

Table 5.2. EBR-II Potential Initiating Events Listed by Event Category

Loss of Flow

Loss of clutch power
Loss of primary pump power
Single pump seizure or coast down
Single pump seizure and coastdown of other pump
Total blockage of core inlet
Inlet pipe break (HPP)
Inlet pipe break (LPP)
Inlet pipe break (all inlet pipes)
HPP to LPP leakage excessive
Missing S/A's
Leakage past S/A's

Reactivity Insertion

ACRDS insertion @ 12¢/sec (Fast Speed)
ACRDS insertion @ 1¢/sec (Slow Speed)
Control rod insertion (one or many)
Core support failure
Safety rod insertion
Voiding or gas bubble in core
Subassembly motion due to hydraulic force
Sudden core radial movement (step insertion)

Loss of Primary Heat Sink

Group 1

Condenser Cooling Water Low Flow
Loss of Condenser Vacuum

Group 2

Loss of Condenser System Flow
Loss of Feedwater Flow

Group 3

Major Steam Leak

Group 4

Secondary System Water-to-Sodium Leak
Secondary System Pump Leak
Sodium Leak in the Secondary Sodium System
Outlet Pipe Rupture
Inadvertent Drain of Secondary Sodium

Table 5.2. EBR-II Potential Initiating Events Listed by Event Category (cont'd)

Group 5

Loss of Secondary Sodium Pump

Overcooling

Primary pump faults
BOP transients (steam leak)
Secondary pump faults

Loss of Decay Heat Removal Systems

Loss of shutdown cooler
Loss of shield cooling

Transients

Scram, spurious
Anticipatory shutdown
Normal shutdown
Core characterization transients

Loss of Sodium

Siphon of Na (defeating siphon break system) through sodium purification system
Primary & Guard Tank Failure and Leaks
Pumpout Na via FERD
Pumpout Na via BFTF
IHX Failure
Shutdown Cooler Failure

Core Support Failure

Core Distortion to more-reactive/less-coolable configuration

Support System Failures

Loss of Normal Power (13.8 kV)
Loss of Instrument Air
Loss of water systems
Loss of constant power
Loss of 125V DC
Loss of 480V AC

Table 5.2. EBR-II Potential Initiating Events Listed by Event Category (cont'd)

Local Faults

Driver fuel failure
Experiment failure
Local blockage (driver or experiment)
Others

External Events

Earthquake
Fire (sodium and non-sodium)
Aircraft Impact
Tornado/High Winds
Lightning
Flood
Volcanism

5.1.3 Loss-of-Flow

Loss-of-flow transients must be divided into separate categories depending upon whether the faults are ones which affect one pump and its associated piping, termed single pump LOFs, faults which affect total flow or both pumps nearly simultaneously, referred to as double pump LOFs, loss of vital electrical power supplies which affect both pumps and possibly other systems and finally a loss of inlet piping integrity. This subdivision is required because of different degrees of damage possible and different responses of the mitigating systems in the various transients.

5.1.3.1 Single Pump LOF

The single pump LOF is discussed first since the double pump LOFs can be described as combinations of single pump LOFs. Factors leading to the loss of one or both primary pumps can be broadly classified as either electrical system faults or mechanical faults.

5.1.3.1.1 Electrical Faults

The major electrical support system to the primary pumps are the primary pump motor-generator sets, one set for each pump. Additional electrical faults can occur in the primary pump motors. The abbreviated schematic below shows the relationship of the motor-generator (M/G) set, the pump and the controller, Fig. 5.1. The electrical failures for the M/G set and pump combination can be subdivided into three sets which are differentiated by the shape of their associated pump coastdown curves, Fig. 5.2.

Measurements of primary pump speeds from recent primary flow coastdown tests [5.6, 5.7] are used as being representative of coastdown conditions during electrical failure faults of the primary pumps. Figure 5.2 shows the pump speed coastdown curves for pump #1 for the three types of electrical failure. Figure 5.2 also shows the assumed pump #1 coastdown behavior during a pump shaft seizure LOF. The coastdown behavior of pump #2 is essentially

Fig. 5.1 Schematic of Pump and Motor Generator Set

Fig. 5.2 Coastdowns for Four Types of Pump or Motor-Generator Set Failure

the same as that of pump #1, except that the stop time of the impeller of pump #2 varies slightly from that of pump #1, for a given coastdown type. For an M/G input coastdown, the pump #1 stop time is 82 s, while the stop time of pump #2 is 87 s.

M/G Input Faults

A failure to the M/G motor leads to the slowest coastdown considered. Failures of this type include the spurious trip of the input breaker to the M/G set or a failure internal to the M/G motor. These faults are developed in detail in the primary pump M/G set system notebook. The loss of normal power would also result in this slow coastdown but is considered separately. This coastdown has been referred to by a number of descriptive terms including input breaker trip, M/G input breaker trip, M/G input, M/G motor. Here the term M/G input will be used for the faults and the term slow will be used to describe the coastdown. The M/G input coastdown is the slowest of the electrical failure coastdowns, because the pump is driven by the combined rotational energy stored in the M/G set, the pump motor, and the pump impeller.

Clutch Faults

Coastdowns associated with faults in the clutch control of the M/G sets have a faster coastdown than that of the M/G input faults. The important faults in a clutch control initiated LOF are the loss of the continuous power supply associated with the respective pumps, faults in the clutch control circuits, spurious trip of the breaker between the continuous power supply and the clutch control circuits, failure of the clutch, and operator error. Operator error during reactor startup or power operation is excluded here because the normal means of tripping the primary pumps is through stop switches that are bypassed by relays powered from RSS control power in the clutch control circuits. Operator error would, therefore, enter the fault tree through an "AND gate" with the failure of either the stop switches, control power relays, or associated wiring. Since all of these hardware faults are modelled as direct clutch faults, operator error, even if explicitly modelled, would be subsumed into these hardware faults.

There are other modes of clutch control failure that do not lead to a LOF but create undesirable operating conditions. These faults cause loss of the ability to change the clutch slip ratio and therefore the pump speed. The pumps, however, continue to operate. In such cases, operator action is necessary to change pump speed by going to either the manual or bypass mode of pump operation depending on the exact location of the fault. The frequency of these events is much less than that of clutch faults that cause a coastdown directly and the most severe outcome is bounded by a clutch failure. Therefore, these events are subsumed into this category.

The clutch failure coastdown is faster than the M/G input coastdown, because the rotational energy of the M/G motor is not involved in the coastdown. The clutch failure coastdown is driven by the combined rotational energy of the M/G generator, the pump motor, and the pump impeller. As shown in Figure 5.2, the coastdown is initially the same as the M/G input coastdown. After about 10 s, the coastdown curves diverge with the clutch failure coastdown decaying more rapidly. The stop time of each of the primary pumps during a clutch failure coastdown is about 23 s. This coastdown decays rapidly because the generator field current, which is supplied by an exciter unit mounted on the output shaft of the M/G motor [5.8], is not affected by a clutch breaker trip. The constant electric field induced by the generator field current produces a braking force which slows the generator drive shaft, which in turn produces a rapidly decaying generator output current causing a rapid coastdown [5.9].

M/G Output Faults

The third subdivision of primary pump electrical faults includes M/G set generator faults, pump motor faults, and trips of the circuit breaker connecting the two. These faults can be considered together because the rotational inertia involved is that of the pump only and they result in the same coastdown.

Other electrical faults are also associated with the primary pump motors since electrical power is used to run small air blowers on each pump which aid in cooling the primary pump motor by transferring heat from a self-contained argon cooling loop within the housing of each

primary pump. These blowers are interlocked with the primary pump starting circuits but not with the run circuits and therefore do not cause an automatic primary pump shutdown if a blower fault occurs. Therefore the blowers are not essential to the primary pump motor operation. The long term effect of blower loss would be a higher operating temperature for the pump and shorten motor lifetime. For these reasons the argon blowers are not modeled.

As shown in Figure 5.2, the M/G output failure coastdown is initially faster than the clutch failure coastdown, because the rotational energy of the M/G set is not involved. The M/G output failure coastdown is driven by the combined rotational energy of the pump motor and the pump impeller. The stop time of this coastdown is greater than the stop time of the clutch failure coastdown because the generator output current, which decays rapidly during a clutch failure coastdown, is not supplied to the pump motor. The stop time of pump #1 during a M/G output failure coastdown is about 32 s, while stop time of pump #2 is about 37 s.

5.1.3.1.2 Mechanical Faults

Mechanical faults are subdivided into binding events and bearing failure. Binding events result in either a limited reduction of pump speed, a postulated very slow coastdown of a pump, or a seizing of a pump that has already stopped so that it cannot be restarted. The much more abrupt seizure is the abrupt stoppage of a pump either due to shaft failure (fracture), shaft seizure, or a complete loss of the shaft coupling. Both of these types of failures have been the focus of an extensive evaluation with respect to degradation and wearout. During the study, a failure modes and effects analysis was performed and documented in the primary pump system notebook (Appendix C.2).

Binding

Primary pump binding could possibly occur due to failure of the upper or lower motor bearings, failure of the hydrostatic bearing, lubrication problems, mild shaft bowing, or buildup of material on the pump shaft. The problems with buildup of material is typical of the events that have been observed in EBR-II. In one case, a lubrication failure allowed lubricating

material to leak onto the pump shaft where it formed a compound with sodium and caused a degradation of pump performance. The pump motor internals were modified to prevent recurrence of this problem. On two other occasions there was a buildup of sodium oxide on a primary pump shaft. The baffle section of the pump was modified to eliminate free interchange of aerosol-laden cover gas with the argon within the baffle by plugging existing upper drain holes and drilling new ones at the sodium level and the lower labyrinth was replaced with one incorporating a reverse spiral, which was designed to enhance clean argon flow down the shaft toward the new drain holes. The Primary-Tank Cover-Gas Sampling-Supply System (PTCGSSS) also supports the pumps by providing a small argon flow along the shafts of the pumps to prevent sodium vapor or sodium oxide from entering the pump motor. The use of the argon adds a weak dependence of the primary pumps on the PTCGSSS. However, the effects of the loss of support from the argon system would be long term and are modelled as binding of both pumps due to a common cause.

The term binding is used collectively to describe all of these slowly developing initiating events. It is also very difficult to assign a coastdown to any of these since the events do not result in a complete stoppage of even a single pump. Therefore, all of the binding events are assumed to be enveloped by a slow coastdown typified by the coastdown for M/G input initiating events.

Seizure

An abrupt seizure of a primary pump may occur due to catastrophic shaft failure in the form of a fracture. Shaft seizure due to extensive shaft bowing may occur but would probably be preceded by binding. Another failure mode would be a complete loss of the motor to shaft coupling caused by fracture of the drawbar or its keying mechanism. This failure would drop the pump impeller onto the pump casing and cause excessive vibration while the pump motor attempted to keep the pump rotating. The results of such an occurrence are difficult to predict and, for the PRA, it is assumed that an abrupt seizure would occur in order to envelop the possible outcomes.

Although not all the abrupt failure modes may be a true seizure, the term seizure is used to group these initiating events. There have been no seizure initiating events in the 25 year history of EBR-II. Therefore, there is no known coastdown associated with a primary pump seizure. For calculational purposes, a primary pump seizure is assumed to be a nearly instantaneous stoppage which is modeled as 0.1 second ramp to zero pump speed in order to aid the numerical convergence of the accident analysis code.

Hydrostatic Bearing Failure

One failure mode considered was the hydrostatic bearing failure at low speed. The pump vendor does not specify a minimum speed at which damage may occur. During initial acceptance testing, the pumps were tested at 175 rpm for 5200 hours. Testing during the 1982 #2 pump repairs included operation at 50 rpm. EBR-II operational procedures prohibit operation below 260 rpm and in the automatic mode of pump speed control, the controller has a lower setpoint. In addition, operation of a single primary pump below the minimum speed while at power would result in a flow below the scram setpoint. Therefore, the hydrostatic bearing failure at low speed is excluded.

5.1.3.2 Double Pump LOF - General

A double pump LOF is defined as the loss of one pump followed by the loss of a second pump within a time interval sufficiently short that the transient caused by the loss of the first pump is still in progress. This LOF includes both independent failures affecting the two pumps within a short time interval and common cause failures. In protected transients, the scram system would act to shut down the reactor following the loss of the first pump within a few seconds. In unprotected transients the reactivity feedbacks inherent in the EBR-II design have sufficient time to reduce power on a timescale commensurate with the full coastdown time thus reducing the effect of the loss of the second pump. Experimental data based on the SHRT tests using a controlled flow reduction has verified these feedback effects [5.10]. The unprotected LOFs studied in the SHRT tests included coastdowns from 95 seconds to as long as 600 seconds (10 minutes).

5.1.3.3 Double Pump LOF - Loss of Vital Electrical Power

The loss of either normal power or loss of constant power are analyzed as a separate class due to certain unique aspects of these transients.

5.1.3.3.1 Loss of Normal Power (LONP)

The loss of normal power is defined to be loss of 13.8 kV, 2400 V, and 480 V normal AC supplies. Included in this event is the loss of 2400 V bus alone. For this case 480 V normal power is still available via the INEL loop, however the effects are similar so it is assumed that the fault also causes loss of 480 V normal power. There is an anticipatory trip in String A of the Reactor Shutdown System directly sensing undervoltage on the 2400 V supply which gives enhanced protection over the LOF trips.

The loss of normal power will result in the same double pump LOF as a double M/G input fault. It is considered separately because it also requires further plant-wide action which alter the means of decay heat removal. When the reactor is operating at power and parallel with the INEL loop (offsite power) there are two sources of power to the 2400 V bus. In circumstances where the INEL loop is lost or EBR-II separates from the INEL loop, the EBR-II generator will continue to supply electrical power to the 2400 V bus, this operating state is termed "on hotel load". Therefore loss of offsite power is not in itself an initiating event while the reactor is at power but it can lead to a loss of normal power. For example the loss of normal power can be caused by fluctuations on the INEL loop severe enough to cause the EBR-II generator to trip as well as losing offsite power. These severe fluctuations have been caused principally by lightning. Another cause of LONP has been electrical transients when reconnecting to the INEL loop while operating on hotel load.

One additional route to a loss of normal power is a shutdown or scram when EBR-II is operating on hotel load (i.e. disconnected from the INEL loop). Since a turbine trip and a trip of the turbine generator output breaker accompany the scram, the generator, which is the sole source of 13.8 kV power when on hotel load, is isolated. This will result in the loss of the

2400 V normal power bus and coastdown of the primary pumps. If the fault was a single pump LOF, a double pump LOF will result. This condition is included in the event tree analysis because operating on hotel load can lead to combinations of initiating events and different consequences.

5.1.3.3.2 Double Pump LOF - Loss of Constant Power

A loss of constant (continuous) power to a primary pump control circuit will result in a LOF involving that pump with the same coastdown as any other clutch failure. The loss of constant power is treated as a distinct LOF because of difference in the plant response. Also the nature of the interconnections between the two constant power static switches is such that there exists a possibility for a common cause power failure to both constant power busses.

The scram clutches on the control rods are held latched by constant power. Therefore, if both constant power supplies fail, the probability of a scram signal is 1.0. The failure to scram then becomes only the mechanical failure of the rods to drop. Thus, the branch point probabilities for failure to scram are lower than for a clutch or clutch control failure.

There are three related terms used to describe the constant power system. These are constant power, continuous power, and uninterruptible power. In a strict sense, the system is designated the constant power system but the other terms, especially continuous power, are used by EBR-II personnel and in some of the plant documentation. The two distribution busses are designated continuous power bus #1 and #2. The normal power supply to the continuous power busses are a pair of uninterruptible power supplies (UPS). Therefore, constant power is sometimes referred to as continuous power based on the bus designation and sometimes uninterruptible power based on the use of UPS to supply power to the bus.

The constant power system (CPS) provides 120 V, 60 cycle grounded single phase uninterruptible power to critical instruments and components throughout EBR-II. The system is designed to reduce reactor scrams and forced shutdowns due to failure or fluctuations in the

continuous power system. The continuous power system utilizes two Uninterruptible Power Supplies (UPS) to power two major distribution panels, Continuous Power Panels 1 and 2.

The UPS units were designed to meet the ANL policy on reactor safety which requires an uninterruptible power supply for the nuclear instruments. The intent of this policy is to ensure that nuclear instrumentation and detector capability is available to monitor the reactor during a station power outage until the reactor shutdown can be verified and the rest of the facility placed in a normal shutdown configuration.

Since UPS units 1 and 2 provide alternate power supplies for each other, reliability of the overall function of the system is increased by reducing the frequency of transients resulting from the failure of a single UPS such as a spurious scram or single pump LOF due to loss of clutch control power. However, this cross-connection of the two UPS units through the two static bypass switches allows the introduction of a potential common failure of both constant power panels.

Another way in which loss of both constant power panels can occur is if only one UPS is in operation. The minimum configuration allowed during normal power reactor operation and during unrestricted fuel handling is that one of the two continuous power supply batteries and its associated UPS system must be operable and normal feeder power must be available to at least the operable UPS unit. Therefore reactor operation with only one UPS is not precluded.

5.1.3.4 Loss of Piping Integrity (LOPI)

5.1.3.4.1 Pipe Break

The CRBR PRA gives a low frequency for this type of initiating event [5.3]. The combination of the low pressure differential and essentially no thermal gradient across the piping produces low pipe stresses from pressure. A break, other than a double guillotine rupture, of an inlet pipe can be bounded by a single pump seizure. A break in the outlet pipe would result in dumping hot sodium to the primary tank rather than the

IHX. The effect on the plant would be that of a loss of primary heat sink rather than loss-of-flow. The frequency of pipe breaks in the primary sodium circuit is, therefore, subsumed into the appropriate pump faults, or the loss of primary heat sink categories.

A double ended guillotine break of a section of inlet piping leads to a more severe transient than the pump seizure because of the ability for the sodium to bypass the core and flow directly from high pressure plenum to the tank through the ruptured loop and is thus identified as a separate initiating event (LOPI). Double ended guillotine breaks of both sets of high pressure inlet piping can occur only due to total disruption of primary system geometry for which no internal initiating event has been identified. In addition it should be noted that thin walled pipes are not believed to exhibit guillotine breaks as a failure mode.

5.1.3.4.2 Ball-Joint Bellows Failure

Each primary pump outlet is connected to the fixed primary sodium piping by a ball-joint bellows which is attached to the outlet of each pump. Failure of these bellows due to either pressure or thermal cycling has been considered. Pressure cycles occur with the starting and stopping of the primary pumps. The normal differential pressure in the primary coolant is 56 psi (386 kPa) and the design leakage paths where the bellows is connected to the fixed piping provides some pressure relief for starting surges. The thermal cycles are usually limited to two cycles from 700°F (371°C) to 350°F (177°C) and back to 700°F (371°C) each year. The rate of temperature change is sufficiently slow to prevent stresses due to transient temperature changes. The bellows of #1 pump have been inspected in 1970 and 1988. The bellows of #2 pump was inspected in 1982. No failures or cracks were detected in any of these inspections. If a crack did occur, its most likely consequence would be an increase in the leakage of the ball-joint bellows. Due to the small consequences and low frequency associated with this type of failure, it has been subsumed into the appropriate pump faults or loss of primary heat sink categories.

5.1.3.5 Other Primary System LOF Considerations

5.1.3.5.1 Auxiliary Pump

For any appreciable amount of decay heat, EBR-II is designed to utilize natural circulation to transfer heat from the reactor core to the primary tank and/or the secondary sodium system. This natural circulation flow is initially 3.3% of full flow following reactor scram. The auxiliary pump is an electromagnetic pump placed in the outlet Z-pipe, whose function is simply to smooth the transition to natural circulation and can supply about 50% of the natural circulation primary system head increasing the available flow to 5%. No credit is taken for operation of the auxiliary pump because analysis indicates that the auxiliary pump does not affect the accident progression and therefore the damage categorization.

5.1.3.5.2 Primary Flow Path Blockage

Previous PRAs for CRBR and SNR300, as well as those for LWRs, exclude total blockage of large coolant pipes. The inlet piping from each pump to the high pressure plenum is 12 inches in diameter and the primary vessel outlet piping is 14 in. in diameter. No debris source has been identified nor is there an identified mechanism to cause debris collection in the primary piping. Furthermore the effects of an inlet blockage can be bounded by a single pump seizure LOF. It is possible to imagine outlet blockage being caused by pinching of the Z-pipe caused by relative movement of the components of the primary system but the only feasible cause for this would be a severe earthquake.

5.1.4 Reactivity Insertion

There are a variety of ways that reactivity might be inserted into the core of EBR-II. There are also ways which have been considered in other PRAs which are not possible in EBR-II. The CRBRP PRA [5.2] identified a class of events such as voiding or a gas bubble in the core. Because EBR-II has a negative void worth everywhere, these events can not lead to reactivity insertion. Overcooling of the primary system can also lead to reactivity insertion

and is considered separately. The remaining possibilities require relative motion of core and control (or safety) rods. The EBR-II control rods are made of fuel with a poison follower above the fuel and are raised to make the reactor critical. Since during shutdown the rods are fully down, the restart involves raising the control rods. Motion of one or more control rods is a leading candidate for a reactivity insertion event (one which has occurred once at EBR-II through a human error). One specific control rod has the capability to be driven by the Automatic Control-Rod Drive System (ACRDS). The ACRDS was installed in order to produce limited transients for core characterization and fuel testing experiments [5.11]. The modes of operation of the ACRDS are slow-speed manual (i.e., just as a normal control rod), slow-speed automatic (up to 5 in./min.), and fast-speed automatic (72 in./min.). In the automatic modes the rod is computer controlled. Fast or slow speed automatic operation is not permitted during periods of normal reactor operations (in fact, the computer is disconnected). Nevertheless the ACRDS offers an additional, low probability, route to reactivity insertion.

All possible means of reactivity insertion by control rod motion can be classified as "slow" because of the drive speeds available. Slow indicates that the reactor will respond quasistatically to the reactivity input. The slow reactivity insertions are further subdivided into four separate events depending upon the total amount of reactivity to be inserted (and different success criteria in terms of the number of rods required to scram). The basic events identified which cause these four slow reactivity insertion events are:

- Control Rod Insertion, (one or many),
- Safety Rod Insertion,
- ACRDS Insertion,
- Combinations of these three events.

Fast reactivity insertion can be envisaged to arise from a sudden core radial movement to a more reactive configuration and internal rearrangement in an experimental subassembly. Technical specifications allow the PRD (Power Reactivity Decrement) to vary 15% from run to run. Assuming a PRD of $\leq 0.4\%$, which has been true for all metal fuel dominated cores, this implies a 0.06% allowable change in reactivity from one run to the next. Therefore, even if all

of this reactivity were available for insertion upon radial motion, the effects are limited. EBR-II safety analysis on the rearrangement of the fissile material in a subassembly indicates that reactivity insertion of 0.33\$ in 1.3 is a bounding estimate, [5.12]. The only identified mechanism for insertions of 0.35\$ to \$1.00 is limited motion of all control rods by slow motion of the control rod lifting platform. One event considered in other PRAs, loss of hydraulic holddown, is prevented by the design of the EBR-II vessel with its holddown fingers, one per assembly. Analysis also has shown that the flow is insufficient to levitate an assembly.

A very large fast reactivity insertion can only be caused by some core support failure. The core is supported directly upon the vessel bottom with a latticework of I-beams, and there is no redan or similar structure as found in other LMR designs. Therefore, the only mechanism identified to lead to fast reactivity insertion are primary vessel catastrophic failure, complete and circumferential, allowing a two inch drop of the vessel and core and ~\$1 of reactivity input. While the most likely cause of vessel failure appears to be an earthquake the possibility of a random non-seismic failure is also considered.

5.1.5 Loss of Primary Heat Sink

Two forms of loss of primary heat sink are identified. These are the loss of secondary sodium heat transport and the loss of steam plant heat removal. The first type includes all faults in the secondary sodium flow paths, faults in the secondary pump and rupture of the primary sodium outlet Z-pipe in the primary tank leading to bypass of the IHX. All of these faults are modelled as an instantaneous loss of secondary sodium heat removal capability. These comprise groups 4 and 5 of Table 5.2. The second type includes all faults in the steam plant which ultimately lead to loss of heat removal capability. In practice the effects of this event are delayed because the heat capacity of the secondary sodium is available to remove heat but this is neglected in the damage analysis. In addition, in the event of loss of normal power, emergency electrical power is used to power the secondary pump and retard the secondary natural circulation flow and the heat load is placed upon the shutdown coolers. These events are those identified in groups 1 and 2 of Table 5.2. Group 3 in Table 5.2 is initially an overcooling event which transfers to a loss of primary heat sink upon operator action and is

treated under overcooling. Water to sodium leaks in the steam generator are treated separately under Steam Generator Tube Rupture.

5.1.6 Overcooling

Overcooling of the primary system will result in a reactivity addition due to the negative temperature coefficient of the reactor if no scram takes place. The potential paths to overcooling were identified as inadvertent operation of the shutdown coolers, primary coolant pump run-up, secondary coolant pump run-up, and steam leak.

5.1.6.1 Steam Leak

Steam leaks which can occur differ in size over several orders of magnitude. Most of the leaks are minor in nature and present operational problems with no reactor safety consequences. These minor steam leaks are initiating events only in the sense that they result in a plant shutdown. Therefore, minor steam leaks are summed with transients as initiators. Major steam leaks will initially cause an overcooling event by removing excessive amounts of energy. If no operator or automatic actions take place, the steam generators will dry-out, or, if the secondary sodium is dumped by the operators to prevent steam generator dry-out, the overcooling portion of the accident ends and a loss of primary heat sink sequence commences, due to the loss of the normal heat removal path. Nonetheless, it is theoretically possible to remove enough heat, given enough feedwater and no operator action or scram, to cause the reactor power to rise high enough to damage fuel.

5.1.6.2 Primary Pump Run-up

The effects of an increase of primary coolant pump speed were investigated during the Inherent Operability Tests (IOT) conducted in 1987 [5.13]. In this test, the primary pump speed and power were initially 30%. Pump speed was then increased to 100% and the power was allowed to adjust through the inherent reactivity feedbacks in EBR-II. The final reactor power was $< 100\%$ and the power-to-flow ratio < 1 for the whole transient.

Since this test showed that the effects of primary pump run-up are minor, it could be removed from consideration as a separate initiating event and just included in the transient category. It will be retained as a distinct initiating event to allow consideration of the accident sequences, and the display of the above result.

5.1.6.3 Secondary Pump Run-up

The effects of an increase of secondary coolant pump speed were investigated during the PICT tests conducted in 1987. In these PICT tests, the secondary pump speed was varied to determine the effect on the primary system [5.14]. This test showed that the effects of secondary pump run-up are minor; it also could be removed from consideration as a separate initiating event, but is retained for the reasons stated in Section 5.1.6.2.

5.1.6.4 Shutdown Coolers Inadvertent Operation

Overcooling due to the inadvertent activation of the shutdown coolers has been removed from consideration due to the small percentage of heat removal relative to rated power, $\sim 1/2\%$.

5.1.7 Steam-Generator Failures

The EBR-II steam generator transfers heat from secondary sodium on the shell side to water/steam on the tube side. The steam-generating system of EBR-II consists of seven parallel-connected natural-circulation evaporators, a steam drum, two parallel-connected superheaters, and interconnecting piping. The evaporators and superheaters are straight-tube heat exchangers. The only differences between the evaporators and superheaters are that core tubes are installed in the steam tubes of the superheaters to increase the steam velocity, and that the superheaters are inverted in relation to the evaporators. The evaporators and superheaters are designed to minimize the possibility of interaction between sodium and water/steam by using "duplex" tubes and double tubesheets. Therefore, there is no location where only a single weld, tube, or tubesheet separates the sodium from the water and/or steam [5.15].

A leak in the duplex tube would result in the injection of high pressure steam/water into liquid sodium. A failure in the welds which connect the duplex tubes to the duplex tube sheets results in leakage into the airspace which has leak detectors. An undetected failure in the outer-tube weld to its tube sheet and an undetected failure in the inner-tube weld to its tube sheet are required to obtain a sodium/water reaction in the airspace between the two tube sheets. The vigorous sodium/water reaction would produce heat, hydrogen, and sodium compounds. Left unchecked, the size of the original leak would increase due to self-wastage of the failed steam tube. Other steam-generator tubes in the vicinity of the leak could also corrode, producing extensive damage to the steam generator and possible damage to the IHX through a pressure pulse or corrosion due to reaction products.

The only detected leak in over 25 years of operation has been a small water leak due to a manufacturing defect in a water-tubesheet weld of an evaporator, which resulted in a small release of steam to the air gap between the tubesheets. The leaking steam was prevented from interacting with sodium by the intact sodium tubesheet and tubesheet-to-tube welds. At no time since have additional leaks been detected in any of the steam-generating equipment.

The only operational difficulty encountered with the steam generating system has been a degradation of heat transfer caused by separation of the duplex-tubing interface resulting from material creep in a superheater containing mechanically bonded tubes. One of the original eight evaporators, which contained metallurgically bonded tubes, was removed, converted into a superheater, and used to replace the original superheater. This operational problem had no safety consequences, [5.15].

5.1.8 Loss of Primary Sodium Accidents (LOSA)

The causes and effects of Loss of Primary Sodium Accidents (LOSA) in EBR-II are influenced greatly by the pool design of the EBR-II plant. The reactor, reactor vessel, and entire primary coolant system are submerged in approximately 90,000 gallons (340 m³) of liquid sodium with no penetrations in the side walls or bottom of the primary and guard vessels. All penetrations are through the top cover, above the bulk sodium surface level. Because the entire

primary coolant system is submerged in the bulk sodium, primary system pipe breaks cannot cause a loss of the primary sodium inventory. A severe leak from a piping rupture would result in a loss of some forced coolant flow, but not a loss of coolant. Primary coolant system pipe breaks are, therefore, considered a Loss-of-Flow (LOF) accident, not a LOSA. Even with a major primary system piping rupture, some flow will be maintained through the core from natural circulation in the primary system and the built-in-leaks in the connections in inlet plena and system piping. Therefore, the definition of a LOSA for EBR-II is cast in terms of the sodium level in the primary vessel relative to that of the various components within the tank.

If the bulk sodium level were to fall 24 inches (0.6 m), the outlet Z-pipe would become uncovered. With the primary pumps operating there will be no effect, but once the pumps are tripped, natural circulation through the IHX can be compromised by the designed leakages at the junction of the Z-pipe and the IHX shell casing. This drop would also stop the operation of the Auxiliary Pump since the bulk sodium is used as part of its conduction circuit. If the pumps are tripped, once the pump coasts down, the sodium can drain out of the Z-pipe through the IHX casing holes, thus causing the sodium to find one level and effectively stopping natural circulation flow through the Z-pipe. The possibility of core damage arises only when the primary pumps are de-energized after a successful shutdown. This scenario is categorized as a loss of circulation, delayed (LCDL), and is produced by a limited loss of primary sodium.

For a LOSA to disrupt the forced flow in EBR-II, the level of the primary sodium must fall below the level of the primary pump inlets. The inlets are located approximately 5 feet (1.5 m.) below the nominal sodium level in the primary vessel. A LOSA of this magnitude would disrupt the primary flow circuit, preventing forced flow through both the inlet and outlet piping, and limit natural circulation to inside the reactor vessel and through the designed leakage paths. This is categorized as a loss of circulation, rapid (LCRA) caused by a large loss of primary sodium.

For the reactor core to actually become uncovered, the sodium level would have to drop approximately 14 feet (4.3 m.). In the analysis of the EBR-II plant, no accidents that could lead to an uncovering of the reactor core without the potential for a large reactivity addition were identified.

Reference 5.16 categorizes the volume of sodium and the corresponding in primary sodium level in the primary associated with potential initiators. Based on the results of that study, eight potential initiating events were identified as sources of sodium of NaK for liquid metal fire considerations and potential LOSA initiating events. One of the events, a primary vessel catastrophic failure, has reactivity insertion consequences in addition to initiating a LOSA. Therefore this initiating event is considered as a large reactivity insertion event (RIFL, see 5.1.4.).

Two of the potential initiating events, the shutdown cooler system rupture and the Fuel Element Rupture Detection system piping leak, do not have the force necessary to remove the primary sodium from the primary vessel, and are therefore not considered as LOSA initiators. They have been included in considerations for liquid metal fire potential in Section 10. The shutdown cooler system failure is also included in the section on transients, since it represents a reduction of decay heat removal capability which could require a reactor shutdown.

Of the five remaining initiating events, all could cause a delayed loss of circulation. Only one, a rupture in both the primary and guard vessels, has the potential to cause a rapid loss of circulation, but only if the leak rate is large enough to lead to early uncovering of the primary pump skirts. The five initiating events which have been identified as potential paths to a LOSA are 1) primary vessel rupture, 2) primary and guard vessel rupture, 3) BFTF failure, 4) primary Na syphon break system failure, and 5) IHX/secondary Na system breach. Each event is considered below.

5.1.8.1 Primary Vessel Rupture

A rupture in the primary vessel allows primary sodium to enter the gas filled annulus between the primary and guard vessels. The drop in the primary bulk sodium level will be from 24 to 34.7 inches (0.6 to 0.9 m.) depending on whether the lower deck on the guard vessel remains sealed, as designed and fabricated, or not. The annular gas filled space between the vessels includes detection systems designed to recognize changes in pressure and leak detectors. There is also a system to measure the level of the primary bulk sodium in the

primary vessel. On receipt of signals corresponding to level changes in the sodium, the emergency procedures call for shutdown of the reactor. However, the leak will continue until the levels equilibrate in the tank and the primary-guard vessel gap.

5.1.8.2 Primary and Guard Vessel Rupture

In the event of a leak in the primary vessel, the failure of the guard vessel must also be considered. The same concerns as the primary vessel rupture apply here except that the predicted drop is 82.6 to 98.7 inches (2.1 to 2.5 m.), depending on the guard vessel lower deck seal and the temperature of the sodium in the air baffle area. In this very improbable initiator the sodium will come into contact with the shield cooling air with the corresponding potential for a sodium fire.

5.1.8.3 Breached Fuel Test Facility (BFTF) Failure

The Breached Fuel Test Facility (BFTF) is a thimble assembly extending through the small rotating plug into the upper plenum of the reactor vessel and is closely coupled to the top of a subassembly in grid position 5A3. The flow from the subassembly is monitored by several diagnostic instruments, including flow meters and a delayed neutron detector. Two independent barriers must fail before sodium can flow out of the primary vessel. The maximum flow through the subassembly is about 20 gpm ($1.3 \times 10^{-3} \text{ m}^3\text{s}^{-1}$), so leakage rate of any sodium loss would be lower. Any failure below the top cover would flow back into the bulk sodium, so only a limited portion of the system would allow sodium to leave the primary vessel. At this low flow rate, substantial time exists for corrective action to be taken, and the reactor shutdown. This failure does not contribute to the LOSA potential because of the low flow rate and multiple annunciator systems included to notify the control room of BFTF problems.

5.1.8.4 Primary Na Purification System Leak

Because major sections of the primary sodium purification system and the radioactive sodium chemistry loop are located below the level of sodium in the primary tank, and the system draws primary sodium directly from the primary tank, the potential exists for siphoning primary sodium through a system piping break at a low elevation. To combat this possibility, there is an Engineered Safety Feature (ESF), the syphon break system, designed to prevent the syphoning of sodium through the purification system. A vacuum is necessary to raise the sodium to a surge tank in the system. If a leak should occur in the purification system, the syphon break system will admit argon gas into the surge tank, breaking the vacuum and removing the ability to draw sodium out of the primary vessel. If this system were to fail, primary sodium could be removed at the maximum rate of 100 gpm ($6.3 \times 10^{-3} \text{ m}^3\text{s}^{-1}$), the rating of the EM pump in the system. All piping outside the purification and sodium chemistry cells is above bulk sodium levels, and the extensive detection systems in the cell mean this failure would not contribute to a LOSA, but is a liquid metal fire source.

5.1.8.5 IHX Tube Failure

During normal operation the secondary sodium system is operated at a pressure of about 10 psi (69 kPa), which is higher than the primary sodium pressure of 3.4 psi (23 kPa). A leak in the IHX would result in secondary sodium entering the primary vessel and a corresponding rise in the bulk sodium level. The action required by the emergency procedures to stop the in-leakage of sodium include shutting down the reactor, stopping the secondary pump, and reducing secondary system cover gas pressure to less than 0.1 psi (1 kPa). If this action is not sufficient to stop the level increase, the secondary sodium is dumped. The secondary sodium dump to the storage tank in the presence of an IHX leak that provides a possible path for primary sodium to be syphoned out of the primary vessel. The secondary sodium system includes an argon vent system that maintains the secondary system pressure and allows for argon to fill the piping as the sodium is dumped. This system also acts to break any syphon, since it is connected to the inlet and outlet of the IHX at elevations above the bulk

sodium level. Because of the number of independent system failures that must take place, this event does not contribute to the LOSA potential.

5.1.9 Support System Failures

Initiating events in this category arise from the malfunction of support systems which are not otherwise specifically assigned to another category. Normal plant shutdown and scrams are defined in section 5.1.11. Specifically excluded are plant experiments and tests such as the SHRT series [5.10]. The nature of these tests are such that a general envelope cannot be provided and the frequency of the tests is unpredictable from a statistical point of view since they are willful human actions. Therefore, experiments and testing are excluded from the PRA in the same sense that other willful acts such as sabotage are excluded. It should be noted that all plant tests are required to go through a review process and therefore are accompanied by a separate safety analysis. In addition the testing period is characterized as having heightened operator awareness so that all human actions, especially manual scram, will have a significantly different probability.

The principal support systems capable of causing plant transients are the electrical systems and the instrument air systems. Other systems such as the water systems may also contribute to transients but act on a timescale that is sufficiently long as to allow corrective action. Also included in this category is the unrecovered loss of a shutdown cooler.

5.1.9.1 Instrument and Plant Air

The plant and instrument air systems are normally cross-connected with the plant air compressor shutdown and isolated, with the instrument air compressors furnishing air to both systems. The plant air system serves as a partial backup because the plant air compressor cannot furnish enough air to maintain pressure in the entire instrument air system unless portions of the air systems are isolated. Therefore, the systems can be considered as one system from the viewpoint of initiating events.

Numerous valves in the feedwater and steam systems are controlled with instrument air, including the steam drum level control. Loss of air supply to these valves can initiate a loss of primary heat sink accident.

Instrument air is also used to maintain the reactor building isolation valves open and the shutdown cooler louvers in the shut position. The set of isolation valves and the two shutdown coolers each have air accumulators and one directional isolation through check valves which prevent the spurious operation of these devices in the event of a temporary loss of the instrument air system. The reactor isolation valve accumulator is sized to allow for two complete cycles of the isolation valves after a loss of instrument air. Therefore this system should function normally. The accumulators for the shutdown coolers have no operational requirements because even if the shutdown coolers operated at full capacity due to the louvers opening at power, the heat removal would increase from 60 kW to 360 kW causing a 0.48% change in power. The only considerations in shutdown cooler inadvertent operation are operational ones at shutdown conditions which would require additional heat input from either the primary pumps or the primary tank heaters to compensate for heat loss.

The pneumatic-operated valves in the primary tank pressure control system will fail closed on a loss of instrument air; and argon cannot be made up to the floating head tank on a low level or pumped down on a high level. The primary tank pressure-vacuum relief system will maintain the cover gas pressure in a safe region between -4 and 13 in. of water. In addition, manual control of cover gas pressure is possible to maintain pressure near normal.

Loss of instrument and plant air is considered to be a separate initiating event, because a total air supply loss will require an anticipatory reactor shutdown and lead to a loss of feedwater, as well as affecting the state of the containment isolation. Thus, it is preserved as a separate event for consideration in a follow-on containment study.

5.1.9.2 Reactor Auxiliaries and Plant Cooling Water Systems

The cooling water system supplies water for direct heat transfer to various components located in the power plant and the sodium boiler building. Cooling of components in the reactor building uses an intermediate heat transfer medium such as freon and silicone oil to transfer the heat to the cooling water system. The cooling water support system consists of the raw water system, the reactor auxiliaries cooling water system, the plant cooling water system and two cooling towers.

No accident initiating events have been identified that can result immediately from loss of the plant or reactor auxiliaries cooling water system. However, cooling water is required for normal operation of critical components such as the MG set clutches, secondary sodium main EM pump, air compressors and generator hydrogen coolers. Therefore, an extended total loss of cooling water would ultimately require an anticipatory reactor shutdown and is included in that category.

5.1.9.3 Electrical Systems

The electrical distribution system is divided into the following systems.

- 138 kV Distribution/Substation Equipment
- 13.8 kV Distribution System
- 2400 V Distribution System
- 480 V Main Distribution System
- 480 V Emergency System
- Continuous Power Supply System
- 125 V DC Distribution System

Of these systems only the 2400 V distribution system and the continuous power supply system are directly involved in initiating events as described above.

2400 V System

The 2400 V system is directly connected with the control power relay string of the reactor. An undervoltage condition will cause the 2400 V undervoltage relay to operate contacts in string A of the reactor shutdown system, causing string A to de-energize, scrambling the reactor in anticipation of a loss of primary cooling flow. This is discussed under Loss of Normal Power.

Continuous Power Supply System

The loss of either continuous power panel will directly cause a reactor scram and also loss of clutch control power to the primary pump connected to the continuous power panel that is lost. The scram occurs in anticipation of a reduction of primary cooling flow and is discussed in the LOF section. In the case where Continuous Power Panel No. 1 fails, the scram also occurs due to the loss of the control rod latch power bus. The simultaneous loss of both continuous power panels results in a total loss of primary flow and requires a reactor scram to limit fuel damage as described in the two pump LOF section. This can also happen if only one uninterruptible power supply (UPS) is in operation and both continuous power panels are supplied from this one source.

There are no other direct initiating events resulting from malfunctions of the continuous power system. If normal feeder power to both units is lost or if neither UPS unit is operable, an anticipatory shutdown of the reactor must be performed before the batteries are depleted. Normally, loss of a UPS does not cause a reactor scram, since the two Continuous Power Panels are then fed from the operable unit or normal power.

Balance of Electrical System

Failures resulting from malfunctions of the 138 kV Distribution/Substation Equipment, 13.8 kV Distribution System (without loss of 2400 V), 480 V Normal Distribution System, 480 V Emergency System, or 125 V DC Distribution System are not direct initiating events.

The loss of all 480 V AC systems, including the emergency power from the diesel generators, does not cause an automatic reactor scram. Loss of 480V power will cause a loss of secondary sodium surge tank level indication and a subsequent trip of the secondary pump. This event is included in the TSEC initiating event. If 480 V AC feeder power is lost to the charging systems of the auxiliary pump battery, station battery or continuous power supply battery, an anticipatory reactor shutdown is required before required functions are jeopardized by excessive battery discharge.

A review of EBR-II Unusual Occurrence Reports (UORs) indicated some LOFs originating from loss of 480 V power early in the plant history. This was because the original design of EBR-II powered the primary pump clutch control from 480 V. A plant modification in 1986 changed the source for clutch control power from normal power to the constant power panels. Transients resulting from the loss of the constant power panels have been discussed earlier.

In the case of the loss of normal power, if the diesels fail to pick up the loads, emergency 480 V power is not available. As mentioned above, this station blackout is no different in its effect on core damage than loss of normal power because EBR-II is designed not to require 480 V AC for reactor operation. However, extended station blackout can have effects on certain key structures which would compromise the ability to restart the reactor. Specifically, the concerns are the temperatures of the concrete biological shield and reactor tank and plugs. Since the difficulties associated with these increased temperatures are operational (not safety-related) there is no unique initiating event associated with loss of 480 V power or complete station blackout.

The only identified effect of the loss of 125 V DC system are the inability to normally control the 13.8 kV, 2400 V, 480 V normal, and 480 V emergency switch gear, selected motor control centers, as well as the emergency DC lighting system. While some protection functions for the switch gear is lost, the AC busses remain in operation. Manual operation of the switch gear would serve as a backup means of electrical distribution.

5.1.9.4 Shutdown Coolers

Under normal operation the shutdown cooler louvers are closed and each cooler removes 30 kW of heat from the primary tank due to natural circulation in the NaK loops and ambient losses through the piping and shutdown cooler box walls. The natural circulation flow of NaK with the louvers closed is monitored to assure that a minimum flow of 25 gpm is maintained. The purpose of maintaining a minimum flow is three-fold: (1) to assure that the flow of NaK is not reduced by either leakage or blockage; (2) to maintain the proper temperature and flow fields so that the transition to natural circulation full flow with the cooler louvers open may be accomplished; and (3) to reduce the thermal shock to the system if the cooler louvers open. If the minimum NaK flow of 25 gpm can not be obtained the reactor must be shutdown via a normal shutdown. Since the cause of below minimum flow is most likely due to an "unrecoverable" loss of a shutdown cooler, either a blockage or leak, the necessary procedure would be to plan to maintain decay heat removal with the balance of plant and keep the balance of plant available until repair of the affected cooler could be completed or until decay heat levels were sufficiently low to allow long term decay heat removal to continue with either a single cooler or the shield and thimble cooling systems. Thus since loss of a shutdown cooler assumes a plant response with reduced decay heat removal capacity, it is maintained as a separate initiating event.

The most likely malfunction of the shutdown coolers is the inadvertent opening of the louvers on one or both of the shutdown cooler boxes. The opening of both sets of louvers is possible due to faults in the primary tank bulk temperature sensing and alarm electronics or a shut valve in the portion of the instrument air system common to both coolers. Even if both cooler box louvers open for both of the shutdown coolers, the resulting increase in heat removal

is a net 300 kW (180 kW each with louvers open, 30 kW each with louvers shut). This represents less than 1/2% of full power (300 kW/62.5 MW) and would not require a shutdown but only result in degraded plant efficiency.

If the opening was the result of the loss of instrument air, the initiating event would be based on the loss of BOP connected with the loss of instrument air and not the involvement of the shutdown coolers. Therefore, the inadvertent opening of the shutdown cooler louvers is excluded as a separate initiating event due to a lack of safety related consequences.

5.1.9.5 Shield and Thimble Cooling

The shield cooling system provides cooling to maintain the top and radial biological shields below 225°F and 150°F respectively. The EBR-II technical specifications require "shield temperatures satisfactory" with a minimum configuration of one temperature alarm channel and one exhaust fan operable. The emergency procedures place additional restrictions of one recirculating fan and one refrigeration system operable. The required action for loss of shield cooling temperature monitoring is a normal shutdown. Loss of the minimum required fans and refrigeration necessitates an anticipatory shutdown or termination of unrestricted fuel handling.

Only 3% of the heat transfer to the biological shields is due to gamma heating with the remainder being a combination of thermal radiation and convection. Therefore, even if the reactor is shutdown, a cooldown of the primary tank is required if the upper and radial biological shields are expected to exceed 225°F and 150°F, respectively. With a total loss of shield cooling these limits are not expected to be exceeded for at least 24 hours, for the upper biological shield, and 100 hours for the radial biological shield as shown in Appendix C.5.

The thimble cooling system provides cooling of the nuclear detectors located in thimbles J-1, J-3, and J-4. Since loss of the neutron detectors would degrade the RSS, it is required that an anticipatory reactor shutdown be performed within 15 minutes if thimble cooling flow is lost

or if temperature $> 135^{\circ}\text{F}$ occurs in two of the three thimbles. Reduced flow or loss of temperature indication requires a normal shutdown.

The total heat removal from the shield cooling and thimble cooling systems is 150 kW which is approximately equivalent to the differential heat removal obtained by opening the louvers on one shutdown cooler at a primary tank temperature of 700°F (371°C).

In evaluating the plant response to loss of shield or thimble cooling, the decay heat removal capacity is reduced and, hence, is evaluated separately from other shutdowns. The initiating frequency for this event is calculated based on the loss of either part of the shield cooling system or the thimble cooling system. However, the plant response assumes that the entire heat removal capacity of both systems is lost. This conservatism is the result of the simplifying assumptions of categorizing the shield and thimble cooling as a single unit for decay heat removal comparable to a single shutdown cooler in capacity. This assumption allowed the use of a two of three logic for short term decay heat removal.

5.1.10 Shutdown and Scram

The number of shutdowns and scrams are based on plant operational data. Included in these categories are spurious trips and anticipatory shutdowns which occur because of faults in noncritical systems. A distinction is made between the normal shutdowns, unplanned or planned, and the annual long shutdown. In the long shutdown the secondary sodium is drained after the reactor is shutdown and the system temperature has been reduced to 350°F (177°C). This process requires at least 18 hours for cooldown and several additional days for reheat to 700°F . Therefore the balance of plant is not immediately available for decay removal after this time. However, the decay heat generation is lower and the starting temperature for the sequence may be lower.

5.1.11 Local Faults

The issue usually referred to as Local Faults has been a subject of research for LMRs for a considerable time. The specific concern is that given some form of local fault, random fuel pin failure or blockage in one subassembly, the fault might propagate until the whole core is involved in a damaging excursion. This issue has been considered in detail in [5.17]. EBR-II has a mission as a research reactor and routinely irradiates experiments to fuel pin failure and beyond without any impact upon driver fuel, i.e. local faults are deliberately initiated in EBR-II. Experimental irradiations which are judged to have the potential to cause problems are irradiated in special hardware (hexcans), designed to prevent damage propagation to the rest of the core. As a result a number of different local faults initiating events were considered, including inlet and outlet blockages, driver and experimental pin failure, and multiple pin failures.

5.1.12 Initiating Event Categories

Table 5.3 gives the EBR-II initiating event categories and the initiating events assigned to each category. This table lists all the considered initiating events.

Table 5.3. Initiating Event Categories

<u>Initiating Event Category</u>	<u>Identifier</u>	<u>Initiating Events</u>
Slow Flow Transient (one loop) Coastdown A	LF1A	M/G set input breaker trip Faults in the motor (of the motor generator set) Shaft binding due to excessive material build-up Slowly developing blockage in loop
Flow Transient due to clutch problems at M-G set (one loop) Coastdown B	LF1B	Loss of power to clutch Clutch internal fault Clutch control malfunction
Fast Flow Transient (one loop) Coastdown C	LF1C	Fault in generator (of motor generator set) Primary pump motor failure M/G set output breaker trip
Immediate Flow Loss in One Loop Coastdown D	LF1D	Shaft seizure Instantaneous blockage of inlet piping Inlet pipe rupture (partial)
Loss of Primary Inlet Piping	LOPI	Inlet piping rupture - double ended guillotine
Loss of Normal Power	LONP	Loss of 2400 V bus Loss of generator output while on hotel load Loss of INEL loop while generator off-line
Loss of Constant Power	LOCP	Loss of both constant power sub-systems Loss of the only operating constant power subsystem while cross-connected
Slow Flow Transient in Both Loops	LF2A	As LF1A in both loops - allows common mode evaluation
Clutch Disconnect on MG Set in Both Loops	LF2B	As LF1B in both loops - allows common mode evaluation
Fast Flow Transient in Both Loops	LF2C	As LF1C in both loops - allows common mode evaluation
Immediate Flow Loss in Both Loops	LF2D	As LF1D in both loops - allows common mode evaluation
Immediate Flow Loss in One Loop and Other Flow Transient in Second Loop	LF2E	Double pump faults of mixed type including one seizure
Other Combination Flow Transients	LF2F	Double pump faults of mixed type - no seizures

Table 5.3. Initiating Event Categories (cont'd)

<u>Initiating Event Category</u>	<u>Identifier</u>	<u>Initiating Events</u>
Slow Reactivity Insertion of < 0.20\$	RISA	Control Rod Insertion Safety Rods Insertion from shim position Multiple Control Rod Insertion
0.20\$ to 0.65\$	RISB	ACRDS insertion fast speed mode or slow speed mode ≥ 2 control rod insertions
0.65\$ to \$1.00	RISC	Multiple rod insertions ACRDS + control rod insertion
> 1.00\$	RISD	Control rod bank insertion
Fast Reactivity Insertion of < 0.35\$	RIFS	Sudden core radial movement Internal failure of experimental S/A
Fast Reactivity Insertion of > 1.00\$	RIFL	Primary vessel catastrophic failure Primary and guard vessel catastrophic failure
Loss of Secondary Sodium Heat Removal	TSEC	Inadvertent dump of secondary sodium Loss of surge tank level indication (trip of secondary pump) Secondary system leak Secondary pump controller error - flow to zero IHX heat transfer degradation Secondary sodium leak Primary outlet piping rupture Loss of secondary sodium pump
Loss of Steam Plant Heat Removal	TBOP	Loss of condenser system water flow Loss of condenser vacuum Loss of feedwater flow Steam generator failure (no sodium water interaction) Turbine generator trip
Local Faults (Rapid)	FLTA	Complete subassembly inlet blockage Complete subassembly outlet blockage
	FLTB	Simultaneous failure of multiple pins
Overcooling Steam Leak	OCSL	Major steam leak Relief valves stay open
Overcooling Primary Pumps	OCPD	Primary pump flow run up to maximum speed of pump

Table 5.3. Initiating Event Categories (cont'd)

<u>Initiating Event Category</u>	<u>Identifier</u>	<u>Initiating Events</u>
Overcooling Secondary Pumps	OCSF	Secondary pump flow run up to maximum speed of pump
Steam Generator Tube Rupture (sodium-water interaction)	EVTR	Steam generator tube rupture Small leaks in duplex tube Steam generator large leak, tube-to-tubesheet weld failure
Rapid Loss of Circulation	LCRA	Primary and guard vessel rupture
Delayed Loss of Circulation	LCDL	Primary tank leak or rupture Primary and guard vessel leak or rupture FERD system leak BFTF leak Primary sodium purification system leak IHX tube failure (combined with other failures of secondary dump tank, etc.)
Transient - Unrecoverable Loss of Shutdown Cooler	TSDC	Break of cooler NaK pipes Leakage of NaK Plugging of NaK loop
Transient - Loss of Instrument Air	TAIR	Loss of Instrument Air
Transient - Unrecoverable Loss of Shield Cooling	TSHC	Loss of shield cooling if unrecoverable gives a shutdown with impaired decay heat removal
Long Shutdown	SHDL	Long shutdown - secondary sodium drained
Short Shutdown	SHDS	Shutdown- secondary sodium at normal conditions
Scram	SCRM	Scram, spurious
Anticipatory	ANTC	Anticipatory shutdowns
Local Faults (Slow)	MDLF	Driver subassembly pin failure
	MELF	Experiment pin failure (metal)
	OELF	Experiment pin failure (oxide)
	SFPB	Partial subassembly blockage
	SFLE	Loading error

Table 5.4 compares the risk-significant accident initiators in the reference LMFBR of [5.5] with the reference list for EBR-II. In the comparison some specific features are worth noting. EBR-II has a uniformly negative sodium void worth so that gas bubbles, etc., in the coolant cannot lead to reactivity insertion. This feature is important because recently PHENIX experienced (negative) reactivity transient attributed to a gas bubble. In addition the spurious, or otherwise, trip of the primary pumps does not apply to EBR-II because the primary pumps are not connected to the PPS.

Table 5.4. Comparison of EBR-II Initiating Events with Risk Significant Accident Initiators in the Reference LMFBR of [5.5]

		<u>EBR-II I.E.</u>
1.	Fuel failure caused by the formation of coolant blockages of fuel subassemblies due to crud, fuel particles, or external objects.	SFPB
2.	Fuel failure caused by fuel pin overenrichment errors.	MDLF
3.	Voiding or gas bubbles in core.	N/A
4.	Loss and re-establishment of hydraulic holddown (19¢ step for 30 assemblies moving) ^a .	N/A
5.	Failure of the core to be supported because of reactor vessel or core support failure.	RIFL
6.	Single control assembly withdrawal at maximum design speed (9 in/min) (< 10¢/sec ramp) ^a .	RISA
7.	Single control assembly withdrawal at maximum speed capability (i.e., 72 in./min for primary rods) (10 to 50¢/sec ramp) ^a .	RISA
8.	Control assembly group withdrawal.	RISD
9.	Earthquakes.	
a.	Sudden core radial movement initiated by an earthquake ($\leq 60\text{¢/step}$) ^a -- stick-slip motion.	
b.	Core, radial blanket, and control rod movement due to a safe shutdown earthquake (60¢ step) ^a .	EXTERNAL EVENT
c.	Core, radial blanket, and control rod movement due to a safe shutdown earthquake (30¢ step) ^a .	
d.	Other system failures.	

Table 5.4 Comparison of EBR-II Initiating Events with Risk Significant Accident Initiators in the Reference LMFBR of [5.5] (Cont'd)

		<u>EBR-II I.E.</u>
10.	Reactor vessel rupture.	RIFL/LCRA/L- CDL
11.	Redan failure	N/A
12.	Loss of one intermediate heat transport loop.	TSEC
13.	Loss of flow in four primary pumps	LF2X
14.	Pump trip in four primary loops due to plant protection signal.	N/A-
15.	Rupture of an intermediate loop in the steam generator bay.	TSEC
16.	Internal failure of a steam generator module.	SGTR
17.	Loss of the main condenser system.	TBOP
18.	Loss of the main feedwater supply.	TBOP
19.	Loss of a steam generator loop.	TBOP
20.	Loss of offsite power.	LONP
21.	Turbine-generator trip.	TBOP
22.	Normal reactor shutdown.	SHDS
23.	Fires.	EXTERNAL EVENTS

^aCRBR - specific estimates of reactivity insertion potential.

5.2 Initiating Event Quantification

Frequency estimates for each initiating event category involves use of a number of different techniques. The first source for initiating event frequencies is plant data. In common with other DOE facilities, EBR-II has been endeavoring to reduce the number of scrams experienced by the plant. A thorough review of the Reactor Shutdown System (RSS) led to many features being removed from the original scram string, the present alignment being in place since 1976, with some minor exceptions. Many other plant modifications were instituted in the period from first criticality, 1964, to 1976 and therefore the historical data used for initiating event quantification was selected to be 1975 to 1989, the results then being collected into three five-year periods. In this way trends, either advantageous or disadvantageous, could be identified and the most appropriate time interval chosen. If the last five years showed a marked change in the number of occurrences, then this was the period adopted for quantification. The quantification uses a Bayesian update of a non-informative prior [5.18].

The review of the EBR-II Unusual Occurrence Reports (UORs) for 1975-1989 gave estimates for the most frequent initiating events. The method used defines the frequency of failures, when N have occurred in T years, as $(2N + 1)/2T$. Use of this formula implies that data based frequencies will be greater than or equal to $3.3 \times 10^{-2} \text{ yr}^{-1}$, because if there have been 0 failures in 15 years, ($T = 15 \text{ yrs}$, $N = 0 \text{ failures}$), the resulting frequency is given as $[2(0) + 1]/[2(15)] = 3.3 \times 10^{-2} \text{ yr}^{-1}$. Less frequent events must therefore be quantified using the fault tree models for EBR-II specific systems. This technique is used for slow reactivity insertion events, flow transients due to pump and motor generator set faults and support system failures such as loss of shutdown cooler when the reactor is at power. Frequencies for these events typically fall in the range of 10^{-2} yr^{-1} or below. Finally, some very rare events are not amenable to quantified fault tree analysis because they rely on more subjective arguments. Examples of initiating events quantified in this way are fast large reactivity insertion, loss of coolant accidents due to catastrophic failure, and steam generator tube rupture.

It is important to bear in mind the distinction between initiating events quantified in different ways. Put simply, for those initiating events whose frequency is based upon historical

data at EBR-II the numerical results can be viewed as "realistic". Those sequences that arise from initiating events whose frequency is derived from fault tree analysis can also be considered "realistic" but the quantification is more uncertain. Note that it is this category that includes most of the classic LMR accidents, particularly slow reactivity insertion and loss of flow events, that have been extensively studied. This feature does not arise by accident, the risk endemic in these events was recognized from the earliest days of the design of LMRs such as EBR-II. As a consequence the design, and later plant modifications, have been so carried out as to reduce the likelihood of events in this class. In fact when EBR-II experienced one control rod insertion event, caused by a human error, the response was to modify the control panel to preclude that particular error, and similarly, when one double pump loss-of-flow arose from a dependency between the clutches on the motor generator sets, the response was to eliminate the dependency. The final class of initiating events, whose frequency is evaluated by other techniques, usually represent a source of unmanageable and irreducible risk to the reactor. Typically the frequencies are derived by semi-quantitative arguments appealing to engineering judgment. Clearly there must be less confidence placed in the numerical results for these transients.

Figure 5.3 shows the operating history for the period 1984 to the long shutdown of 1989. This clearly demonstrates the improved performance achieved by EBR-II in 1987 and 1988. The period chosen ensures that the effects of earlier operation are included to some degree because the good performance of 1987 and 1988 cannot be guaranteed to continue.

5.2.1 Flow Transients (Single Pump)

The initiating event frequencies for each of the single pump LOFs are calculated in two ways. First the primary pump fault trees are combined with the M/G set trees. Results are shown below for the contribution of each pump and for a combination of pumps, Tables 5.5 and 5.6. Secondly a direct data based estimate is evaluated where meaningful data can be found.

EBR-II OPERATING POWER HISTORY

LATEST SIX CALENDAR YEARS

Fig. 5.3. EBR-II Operating Power History

Table 5.5. Annual Initiating Frequencies for Single Pump LOFs by Cause

Initiating Event	Primary Pump 1	Primary Pump 2	Either Pump	Data Based Results
	Annual Frequency Error Factor	Annual Frequency Error Factor	Annual Frequency Error Factor	Annual Frequency Error Factor
Binding	$2.0 \cdot 10^{-1}$ 2.6	$2.0 \cdot 10^{-1}$ 2.6	$4.0 \cdot 10^{-1}$ [1] 2.0	$5.0 \cdot 10^{-1}$ [1] 2.6
M/G Input Faults	$2.3 \cdot 10^{-1}$ 2.8	$2.3 \cdot 10^{-1}$ 2.8	$4.7 \cdot 10^{-1}$ 2.2	$3.3 \cdot 10^{-2}$ 8.4
Clutch Faults	$6.3 \cdot 10^{-1}$ 4.6	$6.3 \cdot 10^{-1}$ 4.5	1.28 4.1	$1.5 \cdot 10^{-1}$ 3.3
M/G Output Faults	$4.2 \cdot 10^{-1}$ 2.3	$4.2 \cdot 10^{-1}$ 2.8	$8.3 \cdot 10^{-1}$ 1.7	$2.5 \cdot 10^{-1}$ 2.6
Seizure	$1.4 \cdot 10^{-3}$ 8.4	$1.4 \cdot 10^{-2}$ 8.4	$2.7 \cdot 10^{-2}$ [1] 5.1	$3.3 \cdot 10^{-2}$ [1] 8.4

Note: [1] The data based results assume that the primary pump runs the entire year since the only time the pumps are usually off is during fuel handling. The fault tree data is based only on critical operation. Therefore, the value is smaller by a factor of $300/365 \approx 0.8$

Table 5.6. Annual Initiating Frequencies for Single Pump LOFs by Group

Initiating Event	Coastdown	Category	Either Pump Annual Frequency Error Factor	Data Based Results
Binding M/G Input Faults	-----> Slow	LF1A	$8.7 \cdot 10^{-1}$ 1.7	$5.0 \cdot 10^{-1}$ 2.6
Clutch Faults	----- Clutch	LF1B	1.28 4.1	$1.5 \cdot 10^{-1}$ 3.3
M/G Output Faults	----- Fast	LF1C	$8.3 \cdot 10^{-1}$ 1.7	$2.5 \cdot 10^{-1}$ 2.6
Seizure	----- Zero	LF1D	$2.7 \cdot 10^{-2}$ [1] 5.2	$3.3 \cdot 10^{-2}$ [1] 8.4

Note: [1] The data based results assume that the primary pump runs the entire year since the only time the pumps are usually off is during fuel handling. The fault tree data is based only on critical operation. Therefore, the value is smaller by a factor of $300/365 \approx 0.8$

Of the three primary pump binding incidents, only one required automatic protective action. The other binding at power was sufficiently slow that an anticipatory shutdown to 50 kW was achieved, followed by manual scram. The remaining binding occurred during startup. The data was grouped into three five year intervals to allow trending. There was one occurrence in the 1975-1979 period, none between 1980 and 1984, and two in the 1985-1989 period. The data based results for the annual frequency using the last five, ten or fifteen years are 0.5, 0.25, and 0.23 yr⁻¹ respectively. Therefore the two incidents that occurred in the last five year period were used for estimating the frequency.

The minor differences between primary pump 1 and 2 as modeled in the fault trees are due to differences in the calculated reliability of the electrical busses supplying power to the respective M/G sets. When the fault tree derived frequencies are compared to plant data the fault tree results are greater by factors of 14, 8.5, and 3.5 for faults of the M/G input, clutch, and M/G output respectively. This discrepancy is due to using generic data for the components involved. There is no specific plant data available at the component level. The five failure types of single pump faults discussed above are combined into four initiating events as shown in Table 5.6.

The annual frequency of initiating event LF1A, slow coastdown, is calculated to be 8.7×10^{-1} which compares favorably with the data based result of 5.0×10^{-1} . Initiating event LF1B, clutch coastdown, is calculated to be 1.28 which is an order of magnitude higher than the data based number of 1.5×10^{-1} . For initiating event LF1C in which the fault tree analysis predicts an annual initiating event frequency of 8.3×10^{-1} compared to 2.5×10^{-1} for the data. Therefore, the initiating event frequencies for all single pump LOF's are directly based on the EBR-II plant data. Therefore, the initiating event frequencies, for all single pump LOF's are directly based on the EBR-II plant data. It should be recognized that the conservatism introduced into the fault tree analysis by the use of generic data are carried into the calculations for double pump LOF's frequencies which are performed by combining fault trees. This is not a concern because the initiating event frequencies for various double pump LOFs are dominated by hypothesized common cause failures.

5.2.2 Flow Transients (Double Pump)

The double pump LOF can be considered as a combination of single pump LOFs. The first pump failure is calculated based on a 100 day reactor run time to correspond with maintenance and testing of the system. The failure of the second pump is based on a mission time of 0.1 hour (6 minutes). This 6 minute interval is sufficient to separate the failure of the two pumps such that the second failure occurs as a separate event with respect to shutdown and flow transient. Although the second pump can fail any time within the 6 minute interval, it is assumed to fail simultaneously (i.e., before the scram system or feedbacks can act) in the damage calculations. This method reduces the total number of accident sequence calculations required. The annual frequency for loss of both primary pumps is calculated based on the results for a 100 day reactor run multiplied by 3.0.

5.2.2.1 Double Pump LOF Fault Tree

Using the five categories of failure for a single pump LOF, a matrix of potential double pump LOFs can be constructed according to Table 5.7. Primary pump 1 is assumed to be the first pump to fail and primary pump 2 is assumed to fail within six minutes.

The lack of symmetry of the data in Table 5.7 for events involving the clutch is primarily due to the effect of the primary pump STOP switches for the second pump which are located at the control panel and the primary panel.

In order to reduce the number of accident sequence calculations required and eliminate manual cut set editing, the same approach as in the single pump LOF was taken to group the double pump LOFs according to their respective coastdown characteristics. The events on the diagonal of the matrix were retained for comparison with the corresponding single pump initiating events and to quantify the common cause aspects of the double pump LOFs. The remaining, mixed mode, double LOFs were grouped based on the commonality of the potential end states of the accident sequences (Table 5.8). All of the mixed mode LOFs which involve

Table 5.7. Annual Initiating Frequencies for Double Pump LOFs by Cause

Pump 2 -->	M/G Input	Clutch	M/G Output	Binding	Seizure
	Frequency Error Factor	Frequency Error Factor	Frequency Error Factor	Frequency Error Factor	Frequency Error Factor
Pump 1					
M/G Input	2.2 10 ⁻⁴ 9.9	1.1 10 ⁻⁴ 7.4	9.2 10 ⁻⁷ 3.8	6.8 10 ⁻⁷ 4.3	4.5 10 ⁻⁸ 10.
Clutch	1.7 10 ⁻⁶ 7.3	4.7 10 ⁻³ 5.5	2.2 10 ⁻⁶ 7.2	1.5 10 ⁻⁶ 8.1	9.6 10 ⁻⁸ 14.
M/G Output	9.4 10 ⁻⁷ 3.7	1.4 10 ⁻⁴ 7.1	2.2 10 ⁻⁴ 9.8	8.2 10 ⁻⁷ 3.5	5.3 10 ⁻⁸ 9.8
Binding	6.8 10 ⁻⁷ 4.0	9.8 10 ⁻⁵ 7.2	8.0 10 ⁻⁶ 3.5	2.1 10 ⁻² 10.	3.8 10 ⁻⁸ 11.
Seizure	4.6 10 ⁻⁸ 11.	6.6 10 ⁻⁶ 16.	5.4 10 ⁻⁸ 9.8	4.0 10 ⁻⁸ 11.	2.6 10 ⁻⁹ 21.

Table 5.8 Annual Initiating Event Frequencies for Double Pump LOFs by Group

Initiating Event	Frequency	Error Factor	Fraction Modelled As Common Cause	Corresponding Single Pump LOF
LF2A	$2.1 \cdot 10^{-2}$	11.	1.00 [1]	LF1A
LF2B	$4.7 \cdot 10^{-3}$	5.3	0.95	LF1B
LF2C	$2.2 \cdot 10^{-4}$	9.8	0.99	LF1C
LF2D	$2.6 \cdot 10^{-9}$	21.	[2]	LF1D
LF2E	$7.0 \cdot 10^{-6}$	13.	N/A	----
LF2F	$3.6 \cdot 10^{-4}$	6.2	N/A	----

Note: [1] Independent failures account for $3.7 \cdot 10^{-5}$ of total

[2] No common cause identified except seismic

the seizure of one pump (zero time coastdown) and any non-seizure fault of the other pump lead to the same end states both in the protected and unprotected case. Therefore, these are all grouped together as LF2E meaning that a shaft seizure plus another pump fault is involved. The remaining mixed mode faults lead to the same states in the protected case and closely related cases in the unprotected case. Therefore, these are all grouped as LF2F, meaning any double pump faults not included in another group.

The common cause failures of both pumps identified, apart from dependence on electrical supply, were due to the STOP switches, binding, M/G input breakers, and M/G output breakers. Electrical supply common cause is considered separately as Loss of Normal Power (LONP) and Loss of Constant Power (LOCP).

The fault trees used for this analysis can be found in Appendix B.

5.2.2.2 Pipe Breaks

Inlet Piping

The inlet plena in EBR-II are fed by pipes from the two pumps. The accident of concern here is a leak or break in one or more of the inlet pipes. One obvious candidate common-cause mechanism for causing a pipe break is an earthquake. This is being treated separately by assigning a fragility to the piping and integrating into the overall seismic assessment. No other candidate common-cause mechanism can be identified.

The piping in the EBR-II primary tank does not have a large differential pressure across the walls. In a recent analysis of leaks in the primary piping for FFTF a leak frequency of 10^{-6} yr^{-1} was proposed [5.19]. This estimate is based upon LWR estimates for long lengths of high pressure piping in the range of 10^{-4} to 10^{-5} yr^{-1} . The reasons for adopting a smaller frequency were 10 years of operation with no leaks, no source of stress corrosion cracking and the applicability of modern probabilistic fracture mechanics (10^{-7} to 10^{-8} yr^{-1} calculated for LWRs). It was also noted that FFTF has 1500 ft of primary piping compared to 3000 ft in a

typical PWR system. EBR-II has all the same advantages that FFTF possesses as well as a piping length an order of magnitude lower. Therefore for EBR-II a leak frequency of 10^{-7} yr^{-1} has been adopted.

Studies on the aging effect of irradiation and thermal cycling have been carried out for the inlet piping [5.20]. The effects are predicted to be negligible.

A small leak in the inlet piping of EBR-II has no effect on the operation of the reactor. EBR-II has leaks designed into the primary circuit and a small extra flow path is irrelevant. The only concern is for a large rupture of the inlet piping. Recently, probabilistic fracture mechanics analyses have been performed for a typical LMR piping system [5.21]. The results presented indicate a cumulative leak probability (over the 40 year life of the system) of approximately $7.8 \cdot 10^{-7}$ for small leaks (less than 3 gpm) and $1.6 \cdot 10^{-8}$ for large leaks (≥ 3 gpm). The corresponding mean failure rates are $2.0 \cdot 10^{-8}$ and $4.0 \cdot 10^{-10}$ per year. These same analyses predicted a cumulative double ended guillotine break probability of approximately 10^{-12} to 10^{-14} yr^{-1} . The ratio of 10^{-2} between small and large leaks, and large leaks and guillotine ruptures is typical of those predicted in PWRs. This suggests using 10^{-9} yr^{-1} for large leaks and 10^{-11} yr^{-1} for double-ended guillotine break.

In summary; for EBR-II the frequencies of leaks has been estimated to be:

Small leak	10^{-7} yr^{-1}	
Large leak or rupture	10^{-9} yr^{-1}	
Guillotine Rupture	10^{-11} yr^{-1}	(LOPI)

As a result small and large leaks are subsumed into the pump faults and the guillotine rupture is the frequency adopted for the LOPI initiating event. The LOPI initiating event frequency is below that used for screening events (10^{-10}) and is thus not a credible initiator.

5.2.3 Loss of Vital Electrical Power

5.2.3.1 Loss of Normal Power (LONP)

Loss of offsite power at EBR-II is a relatively frequent event, occurring typically once per year with a total unavailability of between 1/2 and 1 day per year. As discussed earlier, if offsite power is lost the EBR-II reactor picks up the site load (hotel load) and no initiating event occurs. The Loss of Normal Power arises if the transfer is not made correctly, due to electrical transients when reconnecting to the INEL loop or transients on the loop which take down the generator as well. There has been a significant improvement in EBR-II performance with respect to these events such that analysis of the UORs indicates seven events in 1975-1979, four in 1980-1984 but none in 1985-1989. Therefore the initiating event frequency chosen was based upon no events in five years, 0.1 yr^{-1} .

5.2.3.2 Loss of Constant Power (LOCP)

The motor-generator set clutches have been powered by the constant power system since 1986 and no events attributable to problems in this system have occurred. On the other hand incidents connected with the constant power system have occurred, typically due to maintenance error. Analysis of the UORs identified a total of seven incidents in the last fifteen years, however it should be realized that this total included all UORs which are constant power related. Many occurred after a lightning strike in 1977. Fault tree analysis of the constant power panels gave a frequency of loss of each panel at $5 \cdot 10^{-2} \text{ yr}^{-1}$ and operation with the two systems cross-tied is not limited in any way. In view of all of the above an estimate of the frequency of loss of all constant power was based upon no events in the last five years leading to 0.1 yr^{-1} .

5.2.4 Reactivity Insertion

Since the slow reactivity insertion initiating events have been classed by reactivity insertion potential rather than mechanical cause, two factors must be combined to give the

frequency of each initiating event. The factors are the frequency of each mechanical fault and the probability that a given fault will lead to reactivity insertion of the four classes of insertion (RISA, 0-20¢, RISB, 20-65¢, RISC, 65¢ - 1.00\$ and RISD, greater than 1\$).

Fault tree analysis, as documented in Appendix B, led to the following frequencies:

$$\text{RISA} = 3.0 \cdot 10^{-2} \text{ yr}^{-1}$$

$$\text{RISB} = 7.7 \cdot 10^{-3} \text{ yr}^{-1}$$

$$\text{RISC} = 1.7 \cdot 10^{-6} \text{ yr}^{-1}$$

$$\text{RISD} = 4.4 \cdot 10^{-7} \text{ yr}^{-1}$$

The contribution to RISA is dominated by the frequency of the single rod insertion, RISC and RISD are dominated by two control rod and many control rod insertion events, respectively, while RISB has significant contributions from all events.

The only mechanism identified for a large, fast reactivity insertion (RIFL) arises if the core can drop away from the control rods. A rupture of the primary vessel would not cause such an accident because a failure would begin in one location on one side of the vessel, and as it progressed, if it did, the vessel would cock, jamming the control rods in their positions relative to the core. While some reactivity might be added, the amount would be limited.

However, if the vessel was to catastrophically fail, with an almost instantaneous circumferential failure, it would be possible for the primary vessel and core to fall away from the control rods, adding reactivity. The rupture frequency for the primary vessel was estimated to be $1.0 \cdot 10^{-8} \text{ yr}^{-1}$. A instantaneous circumferential failure is not what is observed in higher pressure tanks with more corrosive materials, therefore it is not expected for the primary vessel. It is believed that a stainless steel vessel at low pressure cannot fail in this unique manner but it is difficult to prove that this is absolutely so and therefore an additional factor of 10^{-2} is applied because of the specialized nature of the failure. This yields a frequency of $1.0 \cdot 10^{-10} \text{ yr}^{-1}$ for a rapid reactivity insertion from the catastrophic, circumferential failure of the primary vessel.

One mechanism for rupture of the primary and secondary vessel is overpressurization of either the primary tank or annulus by a malfunction of the argon system. Although the addition of reactivity is uncertain, the accident sequence would lead to direct core damage due to uncovering the core. Therefore, the argon system overpressure frequency of $2.9 \cdot 10^{-8}$ is added to RIFL for a total frequency of $2.9 \cdot 10^{-8}$.

To estimate the frequency of small, fast reactivity insertion events of $\sim 0.35\%$ (RIFS) is more difficult because the causative mechanism is a gross material motion in an experimental subassembly. This has never happened and there is no basis for an estimate. EBR-II controls the design, fabrication and safety analyses for experiment in such a way that this event has to be considered "unlikely", [5.22], and unlikely is usually interpreted to be in the range 10^{-4} to 10^{-2} . Consequently a screening number of 10^{-3} yr^{-1} will be used. This is consistent with no events of this nature having occurred in approximately 500 experimental subassemblies.

5.2.5 Loss of Primary Heat Sink

The initiating event frequencies are derived from plant historical data. All events, however, minor, have been included in this evaluation. A more critical review of the events could reduce these numbers. The review of the UORs leads to initiating event frequencies of $6.3 \cdot 10^{-1} \text{ yr}^{-1}$ for TSEC and $3.7 \cdot 10^{-1} \text{ yr}^{-1}$ for TBOP.

5.2.6 Overcooling

The initiating event frequencies are derived for these events were derived directly from plant historical data. Review of the UORs for the last 15 years identified a number of events which correspond to these initiators or precursors to these events, 1, 3 (in the last 10 years) and 0 respectively for OCSL, OCPP and OCSP. The initiating event frequencies are derived from these using the standard formulation.

$$\text{OCSL} = 1.0 \cdot 10^{-1} \text{ yr}^{-1}$$

$$\text{OCPP} = 3.0 \cdot 10^{-1} \text{ yr}^{-1}$$

$$\text{OCSP} = 3.3 \cdot 10^{-2} \text{ yr}^{-1}$$

5.2.7 Steam Generator Tube Rupture (SGTR)

Estimation of an initiating frequency for the failure of a steam-tube in an EBR-II superheater is a difficult matter because these superheaters are a unique and conservative design with double-walled tubes, tube sheets, etc., so that it takes breach of two welds to bring sodium and water into contact. There have been no instances of sodium water reaction in the operating history of EBR-II although events have occurred at other reactors that have less conservative steam generator designs, particularly BN-600 in the USSR. In 1987 there was a steam generator failure at the Prototype Fast Reactor (PFR) in Dounreay, Scotland. The design and nature of the failure are not relevant to EBR II steam generators, however it does serve to demonstrate the possibility of rapid propagation given the right circumstances. Estimates available in the literature are given below for single wall tubed heat exchangers:

Eide et al [5.23] and CRBR [5.3]

Small Leaks	5 10^{-6} /hr-unit	7 10^{-6}	~ 510 ⁻² /yr-unit
Medium Leaks	1 10^{-6}	1.4 10^{-6}	~ 110 ⁻² /yr-unit
Large Leaks	3 10^{-7}	2.8 10^{-7}	~ 210 ⁻³ /yr-unit

+Values for water/steam heat exchangers

These values are much too conservative for EBR-II's duplex tube design

A detailed analysis of the thermal degradation of the superheaters suggests a frequency of $\leq 10^{-4}$ yr⁻¹ for all failures, Section 7. Since this latter argument is based upon physical arguments it is believed this estimate is more appropriate than those based upon water/steam heat exchangers which do not have the same double-tube design. Even then the dominant failure mode would not lead to immediate sodium-water reaction, nevertheless 10^{-4} yr⁻¹ with an error factor of 10 is adopted for the screening value for initiating events occurring in the evaporator, EVTR, or superheater, SHTR.

5.2.8 Rapid Loss of Circulation

For a rapid loss of primary sodium (LCRA) to occur, the sodium level must drop

below the primary pump inlet, the pump skirt. Analysis [5.16] shows that this can only occur if both the primary and guard vessels rupture and sustain a rapid loss of primary sodium. In EBR-II, both these vessels were constructed as primary containment systems for the bulk sodium. The guard vessel was designed to hold the entire bulk sodium weight in case the primary vessel leaked.

While the FFTF plant and CRBR design contain significant difference from EBR-II in many aspects, the designs of the primary vessels contain significant similarities in size, design, and materials. All primary vessels are cylindrical containments. FFTF is about 20 feet in diameter and 43 feet high, CRBR was designed to be 20 feet in diameter and 59 feet high, and EBR-II is 26 feet in diameter and 26 feet high. FFTF and EBR-II are of all-welded construction, the same as was planned for CRBR. The main difference between EBR-II and other plants is that the guard vessel is designed as an additional primary vessel rather than just as a containment basin for leaked sodium coolant.

Because of these close similarities in primary vessel design and construction, it is appropriate to use the primary vessel rupture frequencies determined for FFTF and CRBR as the starting point for EBR-II. In the CRBR and FFTF analyses, primary tank rupture initiating event frequencies were determined to be $2.0 \times 10^{-8} \text{ yr}^{-1}$ and $1.0 \times 10^{-8} \text{ yr}^{-1}$, respectively. These values include consideration of the experimental and operational observations of many systems and theoretical fracture mechanics calculations that all point to a leak-before-rupture behavior in current systems and materials, primarily based on water reactor analyses. Liquid metal cooled reactor systems contain other features which make this leak-before-rupture behavior even more likely, including high quality standards, low pressures, an inert atmosphere, isothermal conditions, sodium coolant, etc. The leak-before-rupture argument for CRBR was accepted by the USNRC.

Because EBR-II has been operating a significant period of time, about 25 years, the time frame for initial material and weld defects, which contribute to the majority of large tank failures, has passed. Therefore the tank rupture initiating event frequency from FFTF of $1.0 \times 10^{-8} \text{ yr}^{-1}$ will be used as the basis for EBR-II, since it also has passed the stage of initial material

defect initiated failure. While FFTF has the capability to do in-service inspection and EBR-II does not, the annulus between the primary and guard vessels does include spark type sodium leak detectors and thermocouples that perform this same function, assuring no failure of the primary vessel has occurred. Therefore the value used for the EBR-II primary vessel rupture frequency is $1.0 \cdot 10^{-8} \text{ yr}^{-1}$.

EBR-II has been operating for a significant period of time, so questions of aging effects and irradiation damage are of natural concern. Analyses have been performed and show that the thermal cycling fatigue limits of the primary vessel are orders of magnitude higher than has and would be experienced even under an aggressive transient operational program. Under the normal operation of EBR-II, there is very little thermal cycling because to the isothermal condition of the bulk primary sodium. Calculations of the fast fluence at the peak flux locations and examination of surveillance components also show that the flux is low enough to preclude concerns of metal embrittlement in the EBR-II primary vessel.

The FFTF analysis determined that the leak of a guard vessel in the presence of sodium is $1.0 \cdot 10^{-4} \text{ yr}^{-1}$. However, for a rapid loss of primary sodium, the guard breach must be a rupture rather than a leak. The typical ratio of rupture to leak frequencies for higher pressure and more corrosive environments is 10^{-2} to 10^{-4} . Taking the conservative ratio of 10^{-2} yields a rupture frequency for the guard vessel of $1.0 \cdot 10^{-6} \text{ yr}^{-1}$. Since the guard vessel can not be visually inspected, it must be considered whether the guard vessel has already sustained cracking that would allow sodium to leak into the air baffle structure should the primary vessel rupture. While the guard vessel shows no indication of having sustained any leaking, there is no way to assure its integrity since the annulus gas system is not tested for influx of air or excessive argon makeup, so the failure frequency is increased another order of magnitude to $1.0 \cdot 10^{-5} \text{ yr}^{-1}$.

Based on the necessity of the rupture of both the primary and guard vessels, the rapid Loss of Circulation frequency has been determined to be $1.0 \cdot 10^{-13} \text{ yr}^{-1}$. For this initiator to contribute to core damage, its value would have to be increased by about six orders of magnitude, giving a significant margin of conservatism.

5.2.9 Delayed Loss of Circulation (LCDL)

A delayed LOSA (LCDL) is caused when the sodium level in the primary tank drops below the Z-pipe entrance to the IHX, 24 inch (0.6m) drop in bulk sodium level. This introduces the possibility of losing the natural circulation capability through the IHX if the pumps were to be turned off.

There are two potential initiators, the Fuel Element Rupture Detector (FERD) and the Shutdown Cooler System, that do not contribute to the initiator frequencies because it is not physically possible to move the quantity of bulk sodium needed out of the primary vessel. The primary sodium is drawn into the FERD loop by evacuating the loop and once the loop is filled, the vacuum is manually valved off. If a breach occurs in the FERD loop after the EM pump, the only way to continue to pump out primary sodium, the pump force is the only mechanism for removing the sodium. The 5 psi head produced by the pump is not enough to raise the amount of sodium necessary to keep the loop filled. The shutdown coolers work in natural circulation without any additional force. Even with the maximum temperature difference in the system, there is not enough force to raise sodium in the shutdown coolers to the level of the piping.

A rupture of the primary vessel alone could reduce the surface of the bulk sodium to a level where a LCDL could occur. Based on the discussion given above for the rapid loss of primary sodium, Section 5.2.8, the initiating event frequency is $1.0 \cdot 10^{-8} \text{ yr}^{-1}$.

The Breached Fuel Test Facility (BFTF) could sustain a breach of the dry-well and then a breach of terminal box assembly, allowing sodium removal. There are numerous sensors in the system including flowmeters, Na leak detectors, and thermocouples which would give indications of primary sodium leaks. Since the flow through the subassembly under the BFTF is only ten to twenty gallons per minute, the source for leakage out of the primary system given these multiple breaches would be even less, requiring extensive time before enough sodium could be removed for a LCDL to occur. The frequency was determined to be $5.2 \cdot 10^{-10} \text{ yr}^{-1}$ [5.16].

A leak in the piping of the primary sodium purification and sampling system could

provide a path for the syphoning of primary sodium. There is a syphon break system on the surge tank that draws a vacuum on the primary sodium, drawing it into the surge tank. This break system must fail for any sodium syphoning to occur. If the leak in the piping occurs outside the purification or sodium sampling cell, it would be at an elevation above the nominal sodium level, and would therefore break the syphon even if the syphon break system failed. There is also a pump in the system that increases sodium flow in the piping. There are numerous detectors in the cells, where the elevation is below the critical bulk sodium elevation in the primary vessel. Detection and shutdown of the system, which does not operate continuously, could be performed with the pump and various system valves. The initiator frequency from a leak in the primary sodium purification system was calculated to be $4.4 \cdot 10^{-10} \text{ yr}^{-1}$.

The final initiator identified, a failure of IHX tubes and other component failures, was calculated to occur with a frequency of $1.2 \cdot 10^{-12} \text{ yr}^{-1}$. For this to contribute to the LCDL, the IHX tubes must fail, the secondary sodium system must be dumped and the dump valves must remain open, the argon vent system, with mechanically blocked open valves, must plug during the dump process, and the secondary sodium storage tank must rupture (or a vent valve must open and remain open) to spill sodium out on the basement floor.

The total of all LCDL initiators is therefore $1.1 \cdot 10^{-8} \text{ yr}^{-1}$.

5.2.10 Support System Failures

The loss of shutdown cooler is based upon a fault tree analysis given in the system notebook. The failure mode here is break, leak or plugging of the NaK tube. PFR has had failures in its Air Dump Heat Exchangers which are of similar design but which are operated much differently. The frequency estimate is based upon estimates for liquid metal carrying pipes given in [5.23].

$$\text{TSDC} = 5.8 \cdot 10^{-3} \text{ yr}^{-1}$$

The extended loss of shield cooling is based upon plant data.

$$\text{TSHC} = 3.3 \cdot 10^{-2} \text{ yr}^{-1}$$

The loss of instrument air is based upon plant data.

$$\text{TAIR} = 1.8 \cdot 10^{-1} \text{ yr}^{-1}$$

5.2.11 Shutdowns and Scrams

There is one long shutdown each year where the sodium is drained after reactor shutdown, therefore the frequency of long shutdowns is one per year.

$$\text{SHDL} = 1.0 \text{ yr}^{-1}$$

The number of other shutdowns is dominated by the planned end-of-run shutdowns and shutdowns after plant transient testing and operator training periods. Therefore an overall estimate is provided:

$$\text{SHDS} = 6.0 \text{ yr}^{-1}$$

The frequency of scrams and anticipatory shutdowns has much improved in recent years and therefore the last five years data is used.

$$\text{SCRM} = 7.0 \cdot 10^{-1} \text{ yr}^{-1}$$

$$\text{ANTC} = 5.0 \cdot 10^{-1} \text{ yr}^{-1}$$

5.2.12 Local Faults

Local faults can be divided into two categories, rapid and slow. Rapidly developing local faults require a mechanism which involves total subassembly participation, such as a total blockage (FLTA) or simultaneous break of multiple pins (FLTB). Because of

the subassembly design for EBR-II, with its multiple coolant entrance locations, a complete subassembly inlet blockage is not a credible event. Every subassembly loaded into EBR-II is flow tested. There is also no source of debris within the reactor system to plug either the inlet or outlet. These are discussed in detail in [5.17]. The simultaneous break of multiple pins in EBR-II is also not considered a credible event since no initiating mechanism has been identified aside from loading or orificing errors, which are considered separately. In [5.17] a discussion is given of the thermal effects of gas release from multiple pins breaking, and they are minimal.

Slow local fault generation may occur from several sources: (1) metal driver element breaches, (2) metal experiment element breaches, (3) non-metal experiment element breaches, (4) partially blocked subassemblies, and (5) subassembly loading errors. The metal driver pins are designed to operate up to a designated burnup goal, defined from experimentally determined data. The operating maximum burnup is then defined based on a statistical basis to give a 95% probability of less than one driver element breach in a core loading, which last about two years. This produces an initiating event frequency for MDLF of $1.33 \cdot 10^{-1} \text{ yr}^{-1}$, with an error factor of 10.

In considering experimental pins, there have been a number of breaches on which to base an annual frequency. With about 90 metal experiment element breaches since about 1970, a yearly average of 4.5 (MELF) was determined with an error factor of 3. The number of non-metal experimental element breaches has varied significantly over the EBR-II operating history, depending on programmatic goals. A value of 4.5 per year (OELF) appears reasonable with an error factor of 3, the same as the metal experiments. In the case of experimental pins and subassemblies, a separate safety package must be produced and reviewed by appropriate groups within EBR-II to assure the safety of the reactor and personnel. They may also be operated under special procedures.

Every subassembly which is loaded into EBR-II is argon flow tested which will detect a complete blockage; however, the discrimination may not detect a partial blockage such as might be caused by an orificing error. This has happened on three different occasions, providing an initiating frequency of $1.4 \cdot 10^{-1} \text{ yr}^{-1}$ for the SFPB initiating event. During the 25 year

operation of EBR-II, one loading error took place during routine handling, was detected before reactor operation, and not removed because the conditions were close to what was requested. This is the basis for the initiating event frequency of $6.0 \cdot 10^{-2}$ for SFLE.

The initiating event frequencies are given below for the local fault considerations.

FLTA	-
FLTB	-
MDLF	$1.33 \cdot 10^{-1} \text{ yr}^{-1}$
MELF	4.5 yr^{-1}
OELF	4.5 yr^{-1}
SFPB	$1.4 \cdot 10^{-1} \text{ yr}^{-1}$
SFLE	$6.0 \cdot 10^{-2} \text{ yr}^{-1}$

5.2.13 Final Initiating Event Frequencies

The results of the analyses presented in the preceding sections are summarized in Table 5.9. Thirty-three initiating events have been identified with a frequency of 10^{-10} or greater. Those below 10^{-10} will not be considered further because their risk contribution is below the cutoff frequency applied in this study for any accident sequences. The total frequency of scrams and shutdowns is 11.11 yr^{-1} , Table 5.9, 7.5 yr^{-1} of which are anticipatory or normal shutdowns. The remaining 3.61 yr^{-1} are dominated by balance of plant faults, in excess of one per year, primary pump faults, about one per year, spurious scrams, 0.7 yr^{-1} , and electrical supply faults, 0.2 yr^{-1} .

Although the initiating event frequencies for local faults have been derived in the above description they are not included in Table 5.9 which represents a shutdown history. Local faults represent spatially-dependent initiating events and are therefore treated separately. However, they will be grouped with other internal events as a special class of plant transients. Table 5.10 lists each initiating event, the frequency and error factors together with the basis for the determination, including local faults.

Table 5.9. EBR-II Seismic and Shutdown History

Initiating Event Category	Period			Total	Period Chosen	Frequency [$(2N+1)/2T$] yr ⁻¹	Fault Tree Based Frequency	Other Methods	Final
	1975-1979	1980-1984	1985-1989						
LFA	1	0	2	3	1985-89	0.5	4.0 10 ⁻¹	--	5.0 10 ⁻¹
LFIB	4	0	1	5	1980-89	0.15	1.28	--	1.5 10 ⁻¹
LFIC	0	1	1	2	1980-89	0.25	8.3 10 ⁻¹	--	2.5 10 ⁻¹
LFID	0	0	0	0	--	--	2.7 10 ⁻²	--	2.7 10 ⁻²
LF2A	0	0	0	0	--	--	2.1 10 ⁻²	--	2.1 10 ⁻²
LF2B	1	1	0	2	--	--	4.7 10 ⁻³	--	4.7 10 ⁻³
LF2C	0	0	0	0	--	--	2.2 10 ⁻⁴	--	2.2 10 ⁻⁴
LF2D	0	0	0	0	--	--	2.6 10 ⁻⁹	--	2.6 10 ⁻⁹
LF2E	0	0	0	0	--	--	7.0 10 ⁻⁶	--	7.0 10 ⁻⁶
LF2F	0	0	0	0	--	--	3.6 10 ⁻⁴	--	3.6 10 ⁻⁴
LOPI	0	0	0	0	--	--	10 ⁻¹¹	--	1.0 10 ⁻¹¹
LONP	7	4	0	11	1985-89 ^a	0.1	--	--	1.0 10 ⁻¹
LOCP	5	2	0	7	1985-89 ^a	0.1	--	--	1.0 10 ⁻¹
RISA	0	0	0	0	--	--	3.0 10 ⁻²	--	3.0 10 ⁻²
RISB	0	0	0	0	--	--	7.7 10 ⁻³	--	7.7 10 ⁻³
RISC	0	0	0	0	--	--	1.7 10 ⁻⁶	--	1.7 10 ⁻⁶
RISD	0	0	0	0	--	--	4.4 10 ⁻⁷	--	4.4 10 ⁻⁷
RIFS	0	0	0	0	--	--	1.0 10 ⁻³	--	1.0 10 ⁻³
RIFL	0	0	0	0	--	--	2.9 10 ⁻⁸	--	2.9 10 ⁻⁸
TSEC	1	3	5	9	1975-89 ^a	0.63	--	--	6.3 10 ⁻¹
TBOP	1	3	1	5	1975-89	0.37	--	--	3.7 10 ⁻¹
OCSL	1	0	0	1	1975-89	0.1	--	--	1.0 10 ⁻¹

Table 5.9. EBR-II Scram and Shutdown History (Cont'd)

Initiating Event Category	Period			Total	Period Chosen	[(2N+1)/2T] Frequency yr ⁻¹	Fault Tree Based Frequency	Other Methods	Final
	1975-1979	1980-1984	1985-1989						
OCPP	0	2	1	3	1985-89 ^a	0.30	--	--	3.0 10 ⁻¹
OCSP	0	0	0	0	1975-89	0.033	--	--	3.3 10 ⁻²
EVTR	0	0	0	0	--	--	--	1.0 10 ⁻⁴	1.0 10 ⁻⁴
SGTR	0	0	0	0	--	--	--	1.0 10 ⁻⁴	1.0 10 ⁻⁴
LCRA	0	0	0	0	--	--	--	1.1 10 ⁻¹³	1.0 10 ^{-13 f}
LCDL	0	0	0	0	--	--	--	1.0 10 ⁻⁸	1.1 10 ⁻⁸
TSDC	0	0	0	0	--	--	5.8 10 ⁻³	--	5.8 10 ⁻³
TSHC	0	0	0	0	1975-89	0.033	--	--	3.3 10 ⁻²
TAIR	0	1	1	2	1980-89	0.25	--	--	2.5 10 ⁻¹
SHDL	5	5	5	15	--	--	--	1.0 ^d	1.0
SHDS (unplanned)	4	3	0	7	--	--	--	6.0 ^e	6.0
SCRM	29	9	3	41	1985-89	.7	--	--	7.0 10 ⁻¹
ANTC	3	13	2	18	1985-89	.5	--	--	5.0 10 ⁻¹
TOTAL									11.11

^a Cause removed by plant modification.

^b Problems identified and cured by plant modification.

^c Apparent increase in 1985-1989 caused by software error in secondary pump controller.

^d One per year, each April.

^e Includes planned and unplanned shutdowns - based upon 2 planned shutdowns at end of each 100 day run and 4 other shutdowns after plant transient tests at start of a run.

^f Below screening frequency and not considered further.

^g Control System modified in 1985.

Table 5.10. Initiating Event Frequencies and Error Factors for Quantification

Initiating Event Category	Frequency yr ⁻¹	Error Factor	Basis for Quantification		
LF1A	5.0 10 ⁻¹	2.6	EBR-II Data		
LF1B	1.5 10 ⁻¹	3.3	EBR-II Data		
LF1C	2.5 10 ⁻¹	2.6	EBR-II Data		
LF1D	2.7 10 ⁻²	5.3		Fault Tree	
LF2A	2.1 10 ⁻²	11.		Fault Tree	
LF2B	4.7 10 ⁻³	5.3		Fault Tree	
LF2C	2.2 10 ⁻⁴	9.8		Fault Tree	
LF2D	2.6 10 ⁻⁹	21.		Fault Tree	
LF2E	7.0 10 ⁻⁶	14.		Fault Tree	
LF2F	3.6 10 ⁻⁴	6.2		Fault Tree	
LONP	1.0 10 ⁻¹	8.4	EBR-II Data		
LOCP	1.0 10 ⁻¹	8.4	EBR-II Data		
RISA	3.0 10 ⁻²	5.0		Fault Tree	
RISB	7.7 10 ⁻³	5.0		Fault Tree	
RISC	1.7 10 ⁻⁶	10.		Fault Tree	
RISD	4.4 10 ⁻⁷	10.		Fault Tree	
RIFS	1.0 10 ⁻³	10.			Other
RIFL	2.9 10 ⁻⁸	10.			Other
TSEC	6.3 10 ⁻¹	2.	EBR-II Data		
TBOP	3.7 10 ⁻¹	2.	EBR-II Data		
OCSL	1.0 10 ⁻¹	3.3	EBR-II Data		
OCPP	3.0 10 ⁻¹	8.4	EBR-II Data		
OCSP	3.3 10 ⁻²	8.4	EBR-II Data		
EVTR	1.0 10 ⁻⁴	10.			Other
SGTR	1.0 10 ⁻⁴	10.			Other
LCDL	1.1 10 ⁻⁸	10.			Other

Table 5.10. Initiating Event Frequencies and Error Factors for Quantification (cont'd)

Initiating Event Category	Frequency yr ⁻¹	Error Factor	Basis for Quantification		
TAIR	2.5 10 ⁻¹	2.6	EBR-II Data		
TSDC	5.8 10 ⁻³	4.6		Fault Tree	
TSHC	3.3 10 ⁻²	8.4	EBR-II Data		
SHDL	1.0	2.0	EBR-II Data		
SHDS	6.0	2.0	EBR-II Data		
SCRM	7.0 10 ⁻¹	2.2	EBR-II Data		
ANTC	5.0 10 ⁻¹	2.6	EBR-II Data		
Subtotals	11.11 yr ⁻¹		11.016	9.68 10 ⁻²	1.20 10 ⁻³
MDLF	1.33 10 ⁻¹	10.	EBR-II Data		
MELF	4.5	3.0	EBR-II Data		
OELF	4.5	3.0	EBR-II Data		
SFPB	1.4 10 ⁻¹	2.2	EBR-II Data		
SFLE	6.0 10 ⁻²	3.3	EBR-II Data		
Subtotal	9.33 yr ⁻¹		9.33		

NOTES

Minimum Error Factor set equal to 2.0.
Initiating Events < 10⁻¹⁰ omitted.

Fault trees appear in Appendix B.

5.3 Event Tree Development

5.3.1 Introduction

This section presents the event tree development for the EBR-II PRA. Event trees provide a logical representation of the response of the plant to any abnormal event. The abnormal events are the initiating events discussed in the previous section. Event trees display each of the functions required to ensure safe shutdown, and the consequences of failure to perform the required functions. In order to establish the binary logic necessary to fit the progression of an accident sequence into this structure, success criteria for each system must be generated. The success criteria required of a single system may vary from sequence to sequence.

As described in Section 2 the methodology adopted is that usually characterized as small event tree - large fault tree. This is particularly suitable for EBR-II because of the relative lack of important active functions, when compared to LWRs, which in turn leads to a concise description of the accident sequences. The frequency of each of the accident sequences is quantified by transforming the sequence into a fault tree by linking the fault trees with an AND-gate and editing illogical cutsets.

5.3.2 Safety Functions

The safety functions that are required for safe and stable shutdown are as follows:

Reactivity control (shutdown)

Primary coolant system integrity and inventory maintained

Primary pumps operation (not required for safe shutdown but required by EBR-II

Technical Specifications [5.24])

Decay heat removal

Success Criteria for Safety Functions

Reactivity Control

Reactivity control by the reactor shutdown system (RSS) requires sensing of an abnormal situation, generating a scram signal and releasing the control rods. The safety rods can also be released by operator action. The following table from the Technical Specifications, Table 5.11 gives the normal and limiting configuration of the RSS. The RSS provides overpower, underflow, and over temperature procedures. The subassembly outlet temperature high (SOT) trip is, in effect, automatic protection against core damage in all events, except those events local to a particular part of the core, i.e. local faults.

The issue of scram worth was resolved by establishing probability density functions of scram worth for control rods based upon analysis of 4 years of operating data. The analysis is discussed in more detail in Section 7.3. Fig. 5.4 is a plot of the discrete probability density function for single-rod scram worth. Table 5.12 shows, in tabular form, this probability density function and its integral, the cumulative distribution function (cdf). The cdf value for a reactivity interval is the probability that the scram reactivity is less than the upper limit of the interval. Thus, given a reactivity value that must be overcome, this cdf can be used to find the probability that scrambling one rod will fail to shut down the fission chain reaction. In the absence of a reactivity insertion accident initiator, the reactivity to be overcome is the PRD (Power Reactivity Decrement - defined as the amount of reactivity needed to go from hot critical to full power). The PRD is typically about 30¢ and has never been more than 46¢. Using the cdf in Table 5.12, the probability of one rod failing to overcome the PRD is typically about 6.10^{-6} but can be as high as 10^{-2} . However, scram worth cdfs in Table 5.13 (and probability density functions in Fig. 5.5) show that two rods will more than match even the largest PRD. To make the scram success criterion as simple as possible but not optimistic, scram of two rods will be taken as the success criterion when there is no reactivity insertion accident. In the case of an event which involved reactivity insertion, successful scram requires enough reactivity to be removed to overcome the insertion and the PRD. In most cases two rods are sufficient. In fact, the reactivity insertion initiating events can be classed according to the number of rods

Table 5.11. Limiting Configuration of the Reactor Safety Shutdown System
in the Operate Mode

<u>Channel</u>	<u>Trip Logic</u>		<u>Bypass Limits</u>
	<u>Limiting</u>	<u>Normal</u>	
Earthquake detectors ¹			
Horizontal	1 of 2	2 of 3	
Vertical	1 of 2	2 of 3	No bypass permitted. A single channel may be placed in tripped condition.
High flux level ¹	1 of 2	2 of 3	
Reactor period ²	1 of 2	2 of 3	
Subassembly outlet temperature high ¹	1 of 2	2 of 4	One channel may be bypassed. A subsequent channel may be placed in tripped condition.
Reactor flow low	1 of 3	2 of 5 or 2 of 4	One channel may be bypassed or tripped. Operation with two channels failed is subject to review and approval by the EBR-II Division Director.
Loss of 2400-V bus to primary pumps	1 of 1	1 of 1	
Control-rod manual trip	1 of 1	1 of 1	No bypass permitted
Safety-rod manual trip ³	1 of 1	1 of 1	
Reactor-building exhaust radiation level high	1 of 2	1 of 2	Bypass of both channels permitted when reactor building is isolated.

¹ These trips result in building isolation.

² Required only to 25 MWt.

³ Also activates control-rod-scrum.

Fig. 5.4. Probability Density Function for Single-CR Scram Reactivity

Table 5.12. Discrete Probability Density Function (pdf) and Cumulative Distribution Function (cdf) for Scram Reactivity from a Single High Worth Control Rod.

Rho (cents)		pdf	cdf
MIN	MAX		
0	5	0.000000000	0.000000000
5	10	0.000000000	0.000000000
10	15	0.000000000	0.000000000
15	20	0.000000041	0.000000041
20	25	0.000000691	0.000000732
25	30	0.000004778	0.000005510
30	35	0.000020446	0.000025957
35	40	0.001042961	0.001068917
40	45	0.007465519	0.008534436
45	50	0.059038983	0.067573420
50	55	0.164624479	0.232197899
55	60	0.215690198	0.447888097
60	65	0.212298742	0.660186840
65	70	0.167526359	0.827713199
70	75	0.095187643	0.922900842
75	80	0.049038117	0.971938959
80	85	0.020095220	0.992034179
85	90	0.007965821	1.000000000
90	95	0.000000000	1.000000000
95	100	0.000000000	1.000000000

Table 5.13. Cumulative Distribution Functions for Reactivity from Scram of High-Worth Rods (Exact).

Reactivityv (¢)		1 Rod	2 Rods	3 Rods	4 Rods
min	max				
0	5	0.000000000	0.000000000	0.000000000	0.000000000
5	10	0.000000000	0.000000000	0.000000000	0.000000000
10	15	0.000000000	0.000000000	0.000000000	0.000000000
15	20	0.000000041	0.000000000	0.000000000	0.000000000
20	25	0.000000732	0.000000000	0.000000000	0.000000000
25	30	0.000005510	0.000000000	0.000000000	0.000000000
30	35	0.000025957	0.000000000	0.000000000	0.000000000
35	40	0.001068917	0.000000000	0.000000000	0.000000000
40	45	0.008534436	0.000000000	0.000000000	0.000000000
45	50	0.067573420	0.000000000	0.000000000	0.000000000
50	55	0.232197899	0.000000000	0.000000000	0.000000000
55	60	0.447888097	0.000000029	0.000000000	0.000000000
60	65	0.660186840	0.000000299	0.000000000	0.000000000
65	70	0.827713199	0.000000950	0.000000000	0.000000000
70	75	0.922900842	0.000002103	0.000000000	0.000000000
75	80	0.971938959	0.000582413	0.000000000	0.000000000
80	85	0.992034179	0.002903764	0.000000000	0.000000000
85	90	1.000000000	0.006612765	0.000000000	0.000000000
90	95	1.000000000	0.013770516	0.000000000	0.000000000
95	100	1.000000000	0.036788131	0.000000008	0.000000000
100	105	1.000000000	0.083375613	0.000000130	0.000000000
105	110	1.000000000	0.167729922	0.000000522	0.000000000
110	115	1.000000000	0.289569650	0.000001078	0.000000000
115	120	1.000000000	0.433270252	0.000147700	0.000000000
120	125	1.000000000	0.573137178	0.001619264	0.000000000
125	130	1.000000000	0.696419091	0.003983677	0.000000000
130	135	1.000000000	0.791164600	0.005343461	0.000000000
135	140	1.000000000	0.865813906	0.006726602	0.000000000
140	145	1.000000000	0.919849040	0.011165828	0.000000040
145	150	1.000000000	0.956679988	0.024530602	0.000000251
150	155	1.000000000	0.977135270	0.048488050	0.000000642
155	160	1.000000000	0.988947579	0.086154355	0.000001101
160	165	1.000000000	0.995509071	0.146082098	0.000594161
165	170	1.000000000	0.998757319	0.224036213	0.002501629
170	175	1.000000000	0.999652079	0.325329254	0.003955078
175	180	1.000000000	1.000000000	0.433088989	0.004410833
180	185	1.000000000	1.000000000	0.536989653	0.005268977
185	190	1.000000000	1.000000000	0.626306504	0.007392493
190	195	1.000000000	1.000000000	0.706362082	0.011162688
195	200	1.000000000	1.000000000	0.774202001	0.018658693
200	205	1.000000000	1.000000000	0.832638647	0.032955912
205	210	1.000000000	1.000000000	0.881118054	0.055944611
210	215	1.000000000	1.000000000	0.920147392	0.087996406
215	220	1.000000000	1.000000000	0.948674534	0.132500100
220	225	1.000000000	1.000000000	0.969223022	0.187082255
225	230	1.000000000	1.000000000	0.981361677	0.259964613
230	235	1.000000000	1.000000000	0.989695871	0.343038703
235	240	1.000000000	1.000000000	0.994242018	0.432391521
240	245	1.000000000	1.000000000	0.997375417	0.513186458
245	250	1.000000000	1.000000000	0.998656610	0.587577849
250	255	1.000000000	1.000000000	0.999720911	0.654726075
255	260	1.000000000	1.000000000	0.999909530	0.715690890

Table 5.13. Cumulative Distribution Functions for Reactivity from Scram of High-Worth Rods (Exact). (Cont'd)

Reactivity (β)		1 Rod	2 Rods	3 Rods	4 Rods
min	max				
260	265	1.000000000	1.000000000	1.000000000	0.768571418
265	270	1.000000000	1.000000000	1.000000000	0.814376142
270	275	1.000000000	1.000000000	1.000000000	0.854730144
275	280	1.000000000	1.000000000	1.000000000	0.890621930
280	285	1.000000000	1.000000000	1.000000000	0.920603027
285	290	1.000000000	1.000000000	1.000000000	0.944607494
290	295	1.000000000	1.000000000	1.000000000	0.963091575
295	300	1.000000000	1.000000000	1.000000000	0.976376901
300	305	1.000000000	1.000000000	1.000000000	0.984520983
305	310	1.000000000	1.000000000	1.000000000	0.990306934
310	315	1.000000000	1.000000000	1.000000000	0.994180993
315	320	1.000000000	1.000000000	1.000000000	0.996416691
320	325	1.000000000	1.000000000	1.000000000	0.997893169
325	330	1.000000000	1.000000000	1.000000000	0.999037496
330	335	1.000000000	1.000000000	1.000000000	0.999625937
335	340	1.000000000	1.000000000	1.000000000	0.999854788
340	345	1.000000000	1.000000000	1.000000000	1.000000000
345	350	1.000000000	1.000000000	1.000000000	1.000000000

Fig. 5.5. Exact Probability Density Functions for Scram Worth

required to scram to overcome the net reactivity as given in Table 5.14 below. Note that the RSS may sense that reactivity insertion is taking place through rising power and scram the reactor before the full insertion potential of the initiating event has been reached but the requirement for complete shutdown is that the full reactivity be overcome or, in the case of control rod insertion, that the drive mechanisms be detached from the power source and enough rods inserted to overcome the reactivity insertion that far into the transient.

Table 5.14. Number of Rods Required for Scram
As a Function of Reactivity Insertion

<u>PRD</u>	<u>Reactivity Insertion</u>	<u>Total Required Neglecting Feedback</u>	<u>No. of Rods</u>
$\leq .35 \$$	0 - 0.20\$	$\leq 0.55\$$	2
$\leq .35 \$$	0.20 - 0.65\$	$\leq 1.00\$$	3
$\leq .35 \$$	0.65 - 1.10\$	$\leq 1.45\$$	4
$\leq .35 \$$	>1.10 \$	>1.45\$	5 or more

A second form of reactivity control occurs in the case of failure to scram. In these unprotected events the reactivity feedbacks reduce the fission power and in many cases protect the fuel from damage. This issue is treated separately in the core damage evaluation.

Primary Coolant Inventory

Maintenance of the primary coolant inventory is required to sustain core cooling, however no internal initiating event has been identified, other than vessel failure, which can compromise this system. This event is of extremely low frequency and therefore this system is not included in the event trees except vessel failure is retained as an initiating event.

Primary Pump Operation

In contrast to other LMRs the EBR-II primary pumps are not tripped upon receipt of a scram signal. Primary pumping power is not necessary to achieve safe stable shutdown as long

as the scram has taken place. Nonetheless simultaneous loss of pumping power and some other initiating event must be identified as a separate accident sequence because it has the potential for a greater challenge to fuel integrity if scram fails to occur. The occurrence of coincident but random initiating events is of very low probability. However, there is a mode of operation at EBR-II where the ANL-W site is separated from the INEL loop and EBR-II is supplying the electricity required to carry what is termed the "hotel load". In this eventuality any initiating event which leads to turbine trip leads to a combination of that initiating event and a double pump LOF due to loss of feeder power to the motor generator sets.

The Auxiliary Pump is an Engineered Safety Feature designed to smooth the transition to natural circulation. It is able to provide 3.3% of nominal flow when operating upon battery power and up to 5% on rectifier power. No credit is taken for the Auxiliary Pump and therefore it does not appear as a top event. This course of action was followed because analysis has shown that it does not affect the damage categories.

Decay Heat Removal

Decay Heat Removal is accomplished in a variety of ways. The main method of removing decay heat is via the Intermediate Heat Exchanger to the secondary sodium and balance of plant (BOP). The heat removal capability of the secondary side is sufficiently large that the operating procedures instruct the operator to reduce the secondary natural circulation flow to avoid overcooling the primary tank and to balance the heat removed with the decay heat being generated. In the case of loss of feedwater or loss of normal power the operators do not have a reliable method of determining and maintaining the water level in the steam drum. The decay heat is then rejected through the shutdown coolers and using alternate (emergency 480V) power to reduce the secondary flow to minimize water evaporation. Shutdown coolers also provide a diverse method of decay heat removal in all other cases. The shutdown coolers are passive NaK loops which remove either 60 kW or 360 kW depending on whether the louvers on the natural draft NaK/air heat exchangers are closed or open. The louvers open automatically at a bulk sodium temperature of 710°F (377°C) and close at 700°F (371°C). Finally, the shield cooling system and thimble cooling system combined can remove up to 130 kW but rely on

electrical power for their operation. The decay heat removal capabilities are shown in Table 5.15 and the sources of heat in Table 5.16. For long term decay heat removal the decay heat of fuel in the storage basket may provide a significant heat source. The estimate in Table 5.16 is based upon half of a core, (basket inventory is 50 subassemblies) having been cooled for 7 days. Shutdown (or decay) heat removal has been demonstrated with a series of plant experiments [5.25, 5.26]. These experiments, the Shutdown Heat Removal Test, SHRT, series have demonstrated that the transition to natural circulation and long term decay heat removal will occur without fuel damage.

Comparison of Tables 5.15 and 5.16 allow simple bases for successful decay heat removal to be established. If successful shutdown is being accomplished with all decay heat removal systems available and electrical power available there is a maximum net source of heat of 80 kW apart from decay heat, due to the difference between primary pump heat and the counter-balancing sodium purification system. Taking no credit for the active systems i.e., louvers opening and the shield cooling operating, still gives a residual heat removal capability of 60 kW from the shutdown coolers, 25 kW from convection/radiation to the concrete blast shield and a minimum of 45 kW by the BOP, even if the secondary flow is retarded to minimum allowable. If electrical power is not available there are no extraneous heat sources. In both cases, therefore, the decay heat has to be counterbalanced by operation of the BOP, up to ~3800 kW, opening of the shutdown cooler louvers (a net increase of 150 kW each) or operation of the shield cooling (a net increase of 105 kW). Note that the figures above also show that for shutdown with any one of the shutdown coolers or the BOP completely unavailable through, for example, plugging of shutdown cooler NaK flow or dump of sodium, it is still true that sufficient residual heat losses exist to balance any pump heat if the pumps are not deenergized. This is summarized in Table 5.17. It should be noted that the heat removal capacities quoted are based upon measurements at the normal tank temperature of 700°F (371°C). As the accident progresses and the tank temperature rises the heat removal capability will rise accordingly. No credit is taken for this in the calculations of heat loss from the shield cooling or thimble cooling. The shutdown cooler heat removal has been modelled [5.27].

Table 5.15. Decay Heat Removal Capabilities

<u>System</u>	<u>Mode</u>	<u>Support System Requirements</u>	<u>Heat Removal</u>
BOP	Natural Circulation Retarded Minimum Retarded 0.1 - 0.3%	Water makeup to steam drum Electrical Power + Water Makeup Electrical Power (480V)	2350 kW - 3800 kW As Required 45 kW - 135 kW
Shutdown Coolers	Louvers Closed Louvers Open	Instrument Air, Electrical Power ----	60 kW 360 kW
Shield and Thimble Cooling	Forced Air Cooling Convection and Radiation	Electrical Power (480V) (No shield cooling operational)	130 kW 25 kW
Sodium Purifica- tion	Forced Flow	Electrical Power (480V)	50 kW

Table 5.16. Sources of Heat After Shutdown

<u>System</u>	<u>Mode</u>	<u>Support System Requirements</u>	<u>Heat Addition</u>
Primary Pumps	Forced Flow	Electrical Power (2400 V)	200 kW*
Core Decay Heat	---	---	See Fig. 5.6
Basket Decay Heat	---	---	< 100 kW

*If pumps still energized

Fig. 5.6. EBR-II Decay Heat

Table 5.17. Net Heat Sinks for Reduced Heat Removal Capability

With Electrical Power

<u>Heat Sources [1]</u>	<u>Heat Removal Path</u>		
Pumps 200 kW	Shutdown Cooler	30 kW	(louvers closed)
		180 kW	(louvers open)
	BOP Minimum [2]	45 kW	(0.1%)
	Shield Cooling	-130 kW	
	Radiation/Convection	25 kW	
<u>Net Removal</u>	5 kW (shield cooling and louvers closed)		
	50 kW (no shield cooling and louvers open)		

Without Electrical Power

<u>Heat Sources [1]</u>	<u>Heat Removal Path</u>		
None	Shutdown Cooler	30 kW	(louvers closed)
		180 kW	(louvers open)
	BOP	3800 kW	(nat. circulation)
	Radiation/Convection	25 kW	
<u>Net Removal</u>	3855 kW (BOP natural circulation and louvers closed) [3]		
	205 kW (No BOP heat sink and louvers open)		

[1] Not including decay heat

[2] BOP heat removal increased as necessary

[3] With BOP heat removal, tank temperature remains below 710°F (377°C); therefore shutdown cooler louvers remain closed.

Over the years a great deal of work has been performed establishing the performance of the decay heat removal systems, all of this work is summarized in [5.28] and as part of this work an effective heat capacity for the primary tank and associated structures has been established to be 80 kW hr/°F (5.18×10^8 J/°C), for a full power of 62.5 MW. This equates to 4.6 full power seconds (fps) per °F of tank temperature, (8.3 fps per °C), which is a very large capacity.

The Technical Specifications include a structural temperature limit of 800°F (427°C) which is based upon a conservative evaluation of yield in the reactor tank [5.23] when the freeze seals in the reactor plug are not molten. If the bulk temperature increases by 100°F (56°C), melting of the freeze seals is required and is primarily an operational consideration to prevent deformation which would interfere with the operation of the large and small rotating plugs for refueling. From a safety viewpoint, the concern with elevated primary tank temperature is the structural integrity of the primary tank including its cover.

Additional analysis of the primary tank cover [5.29, 5.30] shows that bowing of the cover due to the temperature gradient across the cover causes a bending and a shear stress at the top of the inner vessel. The failure mechanism of the inner vessel is primarily due to creep stress. Fatigue damage is negligible. These results are summarized in Table 5.18. As can be seen from the table, the primary tank can remain at a temperature at or below 1000°F (538°C) for tens of years without creep failure. Therefore 1000°F was selected as the limit for primary tank temperature in this analysis.

A second reason for the 1000°F limit is that, assuming a 50°F (28°C) temperature difference between the primary tank temperature and the fuel cladding, the fuel remains within its operating temperature limit. Therefore, it is not necessary to perform additional analysis of the driver fuel at decay heat levels and below 1000°F primary tank temperature.

Table 5.18. Primary Tank Limits at Elevated Temperature

Tank Temperature (°F) (°C)		Tank Cover ΔT (°F) (°C)		Stress Intensity (ksi) (MPa)		Time to Creep Failure (hr)	
850	(454)	182	(100)	12.5	(86)	3 10^5	(34 yrs)
950	(510)	206	(114)	14.9	(103)	3 10^5	(34 yrs)
1000	(538)	218	(121)	15.5	(107)	2 10^5	(23 yrs)
1050	(566)	230	(128)	16.0	(110)	2 10^4	(2 yrs)
1100	(593)	241	(134)	16.5	(114)	5 10^3	(208 days)
1150	(621)	254	(141)	17.0	(117)	8 10^2	(33 days)

The decay heat removal scenario developed shows that normal mode decay heat removal is not required for the first 8 hours after shutdown. The shutdown cooler louvers are closed which limits the heat removal rate to 30 kW each. It is also stated that there is no air flow in either the shield or thimble cooling system. Thermal radiation to the concrete of the biological shields accounts for 25 kW of heat removal. Therefore, the maximum decay heat removal for the first 8 hours is 85 kW in this configuration. If the scenario includes an unrecovered loss of a shutdown cooler, the decay heat removal is limited to 55 kW.

The overall decay heat removal scenario is divided into two time periods. Short term decay heat removal is defined as the first 14 days after shutdown. One of the two shutdown coolers is required to operate in the normal mode with its louvers open if the primary tank temperature is greater than 710°F (377°C) for the time from 8 hours after shutdown until 14 days. Decay heat is sufficiently reduced after 14 days that the shield and thimble cooling systems are able to remove the excess decay heat even if both shutdown cooler louvers are closed.

For the BOP to remove decay heat requires that sodium be present and system for providing the ultimate heat sink still exists, i.e., water can be provided to the steam drum. Only two general cases apply, (a) shutdown with the BOP available, in which case only some

subsequent failure or operator action can render it unavailable, or (b) shutdown with the BOP unavailable because of some problem in the secondary or steam systems. The very short term behavior is not of any particular significance from a core damage perspective because of the large heat capacity of the sodium in the tank.

The distinction is made between periods of Short Term Decay Heat Removal, defined as up to 14 days and Long Term Decay Heat Removal defined to extend from 14 days to 45 days. The success and failure criteria associated with decay heat removal are given in Table 5.19. Failure of long term decay heat removal requires the simultaneous failure of four systems and even then the temperature of the tank is readily controlled using parasitic losses (recovery action) and is only included for completeness.

Table 5.19. Success and Failure Criteria for Decay Heat Removal (DHR)

<u>Top Event</u>	<u>Success Criteria</u>	<u>Failure Criteria</u>
BOP decay heat removal maintained	Heat sink available within 8 hours and maintained for 14 days	Unrecoverable loss of heat sink within 14 days of shutdown
Short term decay heat removal initiated and maintained	One of two shutdown coolers available within 8 hours and maintain cooling for 14 days	Less than one cooler available within 8 hours or DHR cannot be maintained for 14 days
Long term decay heat removal maintained	One of three methods, BOP, shutdown coolers, or shield cooling available from 14 to 45 days	All DHR systems fail in 14 to 45 days

Success criteria for the various systems are summarized in Table 5.20.

5.3.3 Damage Definitions

The EBR-II core may contain a variety of fuel types at any one time. There are various forms of driver fuel; Mk II, consisting of U-5 fissium fuel and SS316 clad, Mk III, U-10Zr fuel and D-9 clad which have approved burnup limits and operating temperature limits specified in the Technical Specifications, [5.24]. In addition EBR-II will host a variety of irradiation experiments which might contain, for example, oxide fuel pins, deliberately faulted metal or oxide fuel pins, and structural materials. Clearly, the PRA cannot anticipate every type of experimental subassembly, therefore, the damage definitions applied must be general enough to encompass all these possibilities.

The approach adopted is to identify a plant transient category for each accident sequence and to treat the damage state of the core either by associating a specific fuel damage state or by consideration of the peak temperatures. The resulting set of accident sequences and plant transient states is then invariant under change of core, a very important attribute for EBR-II and its experimental mission. It also has the advantage of being most flexible in allowing different

Table 5.20. Success Criteria for Transients

<u>Decay Heat Removal</u>				
<u>Reactivity Control</u>	<u>Primary Pumps</u>	<u>Via BOP</u>	<u>Shutdown Coolers</u>	<u>Shield Cooling</u>
Two control rods or both safety rods removed from core	Both primary pumps continue to operate	Secondary sodium and balance of plant heat removal available within 8 hours and remains available for 14 days	Shutdown coolers available (louvers open) within 8 hours and remain operable for 14 days for short term decay heat removal	Shield cooling not required for short term decay heat removal
or	or			
Sufficient control rods and safety rods removed from core in the case of reactivity insertion. (Number of rods depends on magnitude of initiating event)	Remaining primary pump continues to operate in the case of a single pump fault			
		One of decay heat removal mechanisms is available for up to 45 days for long term decay heat removal		

definitions of core damage, which might result if the core were to change significantly. Passive safety is not included as a top event in the main tree. This approach prevents having to combine active systems such as removal of decay heat or scram, together with reactivity feedbacks, in to one tree. This view of the event trees for EBR-II is a systemic approach and corresponds to treating the top events as boundary conditions which completely specify the system thermal-hydraulic state. The results of the accident sequence for any particular core can then be evaluated using an LMR system analysis code such as SASSYS [5.31].

All the accident sequences have been classified under six basic plant transient categories. These are described in Table 5.21.

Table 5.21. Definition of Plant Transient Categories

<u>Transient Definition</u>	<u>Category</u>
Direct damage events resulting from vessel and primary piping leaks or rupture, structural failures.	P1
Protected loss of heat sink events; reactor shutdown neutronically but decay heat removal capability fails leading to uniform overheating of system and a long term core vulnerability.	P2
ATWS events leading to a short term core vulnerability (not necessarily leading to core damage, subject to passive safety evaluation).	P3
Protected events of the class that still leads to some degree of damage, distinct from P1 in that protection saves core from core damage category. Leads to minor core damage or potential experimental damage.	P4
Other faults which lead to a degraded containment function and where core damage cannot be ruled out, e.g., steam generator tube rupture.	P5
Local Faults, i.e. events that occur within one subassembly and lead to damage of that subassembly and possibly its neighbors.	P6

In this methodology the frequency of given sequences will be defined in the sequence event trees which will then only need to be updated with plant modifications. The role of passive safety and core damage definitions can then be depicted in a core response tree for each sequence. The basic core response tree forms a partition between degrees of damage. The partition is a direct result of the detailed accident evaluations. Another advantage of this method arises because a separate damage assessment is necessary for each analyzed sequence thereby allowing clear exposition of passive safety consistent with that described in ref. [5.32], where the emphasis is on "passively safe with respect to a particular accident" rather than just "passively safe". It should be noted that for many sequences the conditional probability of damage is 0 or 1 and a detailed analysis is not warranted.

The SASSYS systems analysis code has been used to analyze EBR-II transients using best estimate feedbacks, [5.33]. The fuel damage criteria against which results are judged are as follows:

- (1) If an average driver subassembly reaches boiling, fuel melting or fuel pin failure then widespread core damage is presumed, (CD). (Average is defined by a subassembly whose outlet temperature is that of the core average).
- (2) If the interior pins in the hottest allowed driver subassembly reach fuel pin failure, fuel melting or the sodium associated with these pins boils, but no damage occurs in average driver assemblies, then this is termed Minor Core Damage, (MCD).

In these criteria fuel pin failure is assessed using fuel pins at the peak allowed burnup for the driver fuel. Below this threshold (item 2) above no damage to driver fuel would occur but radionuclide release is not precluded because experiments may be irradiated more aggressively than driver fuel. Therefore the no damage region is subdivided into three regions based upon Technical Specification distinctions.

- (3) If the clad temperature in the hottest driver pins exceeds 1500°F (816°C) or lies between the eutectic temperature and 1500°F for greater than 60s then there is the potential for damage to experiments, (PED).
- (4) If the clad temperature in the hottest driver pins lies between the eutectic temperature and 1500°F for less than 60s then no damage will occur, (ND). This is the limited capability operating regime.
- 5) If the temperature never exceeds the eutectic temperature no damage occurs, (ND), and this is the unlimited capability operating regime.

In the CDFR PRA reported in [5.34] the damage criterion adopted was a sodium temperature of 900°C, including uncertainties. This roughly corresponds to classing PED, MCD and CD together as core damage. These criteria are summarized in the basic core response tree shown in Fig. 5.7. Note that the time ordering implied in Fig. 5.7 is only strictly valid for slow transients. Fast transients may go straight to fuel damage, CD or MCD.

The core response is of most importance in the unprotected sequences where the reactivity feedbacks are known to limit the temperature rise and hence the resulting damage. In this case, the final state is not a completely subcritical state and there is the possibility that a subsequent separate event could occur which would alter this state. At the end of the transient where a stable state has been attained, the fission power will be defined by the heat sink available. The probability of attaining this stable but still critical state is low because the scram reliability is high. If the cause of failure to scram lies in the signal, mechanical (relays) or electrical components then the control rod drives will still be connected to the motors and positive and negative reactivity insertion by rod motion is still feasible. Currently no procedure exists which directs operator action under these circumstances, however, since the recovery action is trivially obvious and easy to carry out, it is not unreasonable to assume that the necessary signal, or release of stuck relay etc., can be carried out in a timescale of a few hours. Generally, if the control and safety rods failed to drop because they are all stuck then rod-induced reactivity insertion due to rod motion is equally not possible and the reactor will

EVENT TREE FOR: Fuel Damage Sub-Tree

17-OCT-90

Fuel Damage Sub-Tree	T < Eutectic T in Hot Pin	Temperature Between Eutectic T and 1500 F for 60 s in Hot Pin	No Pin Failure or Boiling in T. S. Limit Pins	No Boiling or Fuel Melting in Average Core Assembly	Sequence	Class	Fuel Damage	Frequency
IDAM	TLTT	TGTT	NPFB	NBFM				
					IDAM-1		ND	
					IDAM-2		ND	
					IDAM-3		PED	
					IDAM-4		MCD	
					IDAM-5		CD	

Fig. 5.7. Core Response Tree for Fuel Damage Definition.

maintain its equilibrium power and temperature level until corrective measures are taken, meanwhile the power level will decrease as fuel is burned. All the potential events not involving positive reactivity insertion, e.g., LOF, LOHS, are benign. A basic assumption is that the recovery action to achieve reactor shutdown is assumed to be guaranteed and the accident is effectively terminated if the initial temperature transient does not lead to damage.

In the event trees, and the accident sequence descriptions that follow, the end states are assigned to a core damage bin on the basis of best estimate analysis. The uncertainties, and hence reliability associated with that assignment, are treated separately. In the case of long term heating of the core, tank, and structure through loss of decay heat removal a separate damage category is defined, which is associated with plant state P2. This category is specified by a single tank temperature limit above which some core and structural damage is assumed to take place. The temperature limit is chosen below the maximum allowable clad temperature because of uncertainties in the long term high temperature behavior of the various fuel-clad combinations in the EBR-II core. This category is denoted CSD (Core/Structural Damage). Table 5.22 combines the plant transient categories with the associated fuel damage possibilities.

Table 5.22 Plant Transient and Associated Fuel Damage Categories

Plant Transient	Fuel Damage
P1	CD
P2	CSD
P3	All possible including no damage
P4	PED or MCD
P5	ND or CSD
P6	PED, MCD or CD

5.3.4 EBR-II Event Trees

A great deal is known about the details of accident sequence progression in EBR-II and this understanding is embodied in the various analysis tools that have been applied to EBR-II such as NATDEMO, DSNP and SASSYS, [5.31, 5.35, 5.36]. Event trees were developed using this understanding and the direct involvement of EBR-II Safety, Engineering and Operations personnel.

5.3.4.1 Review of Event Trees in Other Relevant PRAs

EBR-II, or any reactor, is designed with a very simple philosophy; in the event of a transient (initiating event) the first line of defense is to scram the reactor and, if scram is successful, to provide various means of removing the decay heat. The scram is achieved by removal of control rods (with safety rods as a backup). As described above, the decay (or shutdown) heat removal is carried out through the secondary sodium system (except for loss of normal power or loss of feedwater) and the shutdown coolers. The shield and thimble systems will also remove some heat if they are available. This design can be expressed as a simple event tree as follows, Fig. 5.8. The protected transients without decay heat removal can lead to some potential core vulnerability but this occurs in the long term only, because the lack of a heat sink can only affect the core on timescales which are long compared to scram actions (Plant Transient Type P2). Finally failure to scram will lead to a core vulnerability in the short term, irrespective of the heat sink available (Plant Transient Type P3).

It is instructive to review the event trees from other relevant PRAs as a guide to construction of those for EBR-II. Figure 5.9 shows the basic event tree for (non-LOCA) initiators in the N-Reactor PRA [5.37]. A very close analogy can be drawn with EBR-II, as is also indicated in Fig. 5.9. In the case of N-Reactor, the normal heat removal system is backed up by an Emergency Core Cooling System (ECCS). The Graphite and Shield Cooling System (GSCS) can also limit damage and is analogous to the EBR-II shield cooling. In Fig. 5.9 one top event was removed. This removed event allows for local effects in the ECCS which has no direct analogue at EBR-II. The only remaining endstates that admit damage are Z1(DEL) and

EVENT TREE FOR: INITIATING EVENT

INITIATING EVENT	SUCCESSFUL SCRAM	DECAY HEAT REMOVAL	Sequence	Class	Fuel Damage	Frequency
I	SCRM	DHR
I-1		NS	
			I-2		LCV	
			I-3		CV	

Fig. 5.8. Functional Event Tree for EBR-II

Fig. 5.9. Transient Event Tree for N-Reactor with EBR-II Analogues (Local Effects Removed)

Z, which represent whole core damage delayed or immediate, and Y which is damage limited by the GSCS heat removal when no primary decay heat removal or ECCS was available. No attempt is made to distinguish between failure to scram sequences and failure to remove decay heat sequences in the extent of core damage or to evaluate, by calculation, the details of damage, although N-Reactor shares many of the advantageous features that EBR-II possesses, (i.e., metallic fuel and large excess thermal capacity).

In the CRBR PRA, the bins were defined as accident sequences and not damage states. Figure 5.10 is the core damage event tree for internal initiators in the CRBR PRA [5.3]. This event tree considers short term and long term cooling via the steam generators and core cooling via the DHRS (analogous to Fig. 5.9). The additional elements in the CRBR event tree, P, Q and I, are included as discriminators to allow specific sequences to be identified and evaluated within one overall tree. These discriminators are Pump Tripped (P), Initial Power Normal (Q) and Sodium Inventory (I). Again, it should be noted that all sequences which involve failure to scram have been assumed to lead to core damage, as does the Protected LOCA as well as the Protected Loss of Heat Sink and its delayed counterpart, (sequences TI, TSD, and TCD, respectively). For internal initiating events, only core damage sequences TKP, Unprotected Loss of Heat Sink (frequency 2.5×10^{-7}), and TSD, Loss of Heat Sink, (frequency 3.4×10^{-6}), exceeded a cut off frequency of 10^{-8} , (indicated by * in Fig. 5.10). For the EBR-II PRA an important issue is whether unprotected events do in fact lead to core damage.

5.3.4.2 Event Tree Construction

In the previous section definitions for fuel damage criteria, plant transient state and the success and failure criteria for the basic event tree top events were derived. In this section this information is used to present the event trees for each initiating event discussed in section 5.1.

In unprotected events the details of the heat sink are generally unimportant because the damage, or lack of it, occurs on a very short timescale. In fact calculations can be made which demonstrate that for unprotected transients where the reactivity feedbacks lead to a benign

EVENT NAMES

- T - Initiator
- K - Reactor Shutdown
- P - Main PHTS Pump Motors Tripped
- Q - Initial Power Normal
- I - Sodium Inventory Maintained
- S - Short-term Cooling via SGS Maintained
- C - Long-term Cooling via SGS Maintained
- D - Core Cooling via DHRS Initiated and Maintained

31183812

Fig. 5.10. Core-Damage Event Tree for Internal Initiators in the CRBR PRA.

response to the accident the loss of the heat sink is beneficial. This surprising fact reflects the negative inlet temperature coefficient which leads to a negative reactivity input for any inlet temperature rise. All unprotected transients will therefore be assumed to have full heat sink available for the first few minutes except if the initiator itself affects the heat sink. Once the initial transient period is over, and if no damage has occurred the system approaches a state where the fission power matches the heat rejection capability and the transient is over. Any subsequent failure of the decay heat removal capability is covered by the protected case equivalent sequence and the use of the rare event approximation for successful scram, 1.0 instead of $(1.0 - \epsilon)$.

In the simple functional event tree of Fig. 5.8 a top event of successful scram is used. Because the scram signal can have other effects other than causing the rods to drop this top event has been separated into scram signal generation and control or safety rods drop. The particular effect that the signal has is to trip the turbine which, in the case when the system is on hotel load, causes the pumps to coastdown.

A detailed discussion of the appropriate event tree for each initiating event now follows using the guidelines established here.

5.3.5 Event Trees for Flow Transients

5.3.5.1 Initiating Event Description

Flow transients include all single or double pump faults, flow blockages and failures in the coolant circuit. Pump faults can occur in several ways which lead to characteristic coastdown times. The modes of failure and resulting coastdowns are analyzed in detail in the system notebooks for the pumps and motor generator sets, Appendices C.2 and C.3, and discussed earlier in Section 5.

5.3.5.2 Normal Plant Response and Success Criteria

Whatever the rate of coastdown and number of pumps involved, the normal plant response is the same. The RSS has a LOF protection subsystem which generates a scram signal, and in certain cases, the SOT subsystem may precede it in generating the scram signal. The scram signal causes the control rods to drop and after a 6 s delay, the secondary pump to trip. The turbine will drop off-line which normally has no effect on electric power except for the case when the EBR-II site is separated from the INEL loop and the turbine is providing hotel load. The success criteria for the two relevant systems are given in Table 5.20. Insertion of two control rods or both safety rods will shut the reactor down. The normal heat sink is available for removing decay heat. The operators must retard the secondary flow according to procedure to reduce excessive heat removal due to natural convection in the secondary system.

5.3.5.3 Event Trees and Accident Sequences

Figures 5.11 to 5.20 show the event trees for single and double pump faults. Core damage potential due to LOF accidents is dependent upon the duration of the pump coastdowns and the response of the reactor shutdown system and is discussed in detail in 5.4.2.2.

EVENT TREE FOR: Single Pump Loss of Flow - Group A

31-MAY-91

Fig. 5.11. Event Tree for Single Pump Fault - Group A

EVENT TREE FOR: Single Pump Loss of Flow - Group B

31-MAY-91

Single Pump Loss of Flow - Group B	Scram signal during LOF event	Sufficient control or safety rods drop during LOF event	Primary pumps keep running	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
LF1B	FSIG	FROD	PRUN	BPHR	DHRS	DHRL				
<pre> graph LR LF1B[LF1B] --> LF1B_1[LF1B-1] LF1B --> LF1B_2[LF1B-2] LF1B --> LF1B_3[LF1B-3] LF1B --> LF1B_4[LF1B-4] LF1B --> LF1B_5[LF1B-5] LF1B --> LF1B_6[LF1B-6] LF1B --> LF1B_7[LF1B-7] LF1B --> LF1B_8[LF1B-8] </pre>							LF1B-1	P2	CSD	
							LF1B-2			
							LF1B-3			
							LF1B-4	P2	CSD	
							LF1B-5	P2	CSD	
							LF1B-6	P3	ND	
							LF1B-7	P3	MCD	
							LF1B-8	P3	ND	

Fig. 5.12. Event Tree for Single Pump Fault - Group B

EVENT TREE FOR: Single Pump Loss of Flow - Group C

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Single Pump Loss of Flow - Group C	Scram signal during LOF event	Sufficient control or safety rods drop during LOF event	Primary pumps keep running	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
LF1C	FSIG	FROD	PRUN	BPFR	DHRS	DHRL	LF1C-1			
							LF1C-2	P2	CSD	
							LF1C-3			
							LF1C-4	P2	CSD	
							LF1C-5	P2	CSD	
							LF1C-6	P3	ND	
							LF1C-7	P3	MCD	
							LF1C-8	P3	ND	

Fig. 5.13. Event Tree for Single Pump Fault - Group C

EVENT TREE FOR: Single Pump Loss of Flow - Group D

31-MAY-91

Single Pump Loss of Flow - Group D	Scram signal during LOF event	Sufficient control or safety rods drop during LOF event	Primary pumps keep running	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
LF1D	FSIG	FROD	PRUN	BPHR	DHRS	DHRL	LF1D-1			
							LF1D-2	P2	CSD	
							LF1D-3			
							LF1D-4	P2	CSD	
							LF1D-5	P2	CSD	
							LF1D-6	P3	ND	
							LF1D-7	P3	MCD	
							LF1D-8	P3	ND	

Fig. 5.14. Event Tree for Single Pump Fault - Group D

EVENT TREE FOR: Double Pump Loss of Flow - Group A										31-MAY-91
Double Pump Loss of Flow - Group A	Scram signal during LOF event	Sufficient control or safety rods drop during LOF event	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency	
LF2A	FSIG	FROD	BPHR	DHRS	DHRL					
						LF2A-1				
						LF2A-2	P2	CSD		
						LF2A-3				
						LF2A-4	P2	CSD		
						LF2A-5	P2	CSD		
						LF2A-6	P3	PED		
						LF2A-7	P3	PED		

Fig. 5.15. Event Tree for Double Pump Fault - Group A

EVENT TREE FOR: Double Pump Loss of Flow - Group B

31-MAY-91

Double Pump Loss of Flow - Group B	Scram signal during LOF event	Sufficient control or safety rods drop during LOF event	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
LF2B	FSIG	FROD	BPHR	DHRS	DHRL				
						LF2B-1			
						LF2B-2	P2	CSD	
						LF2B-3			
						LF2B-4	P2	CSD	
						LF2B-5	P2	CSD	
						LF2B-6	P3	CD	
						LF2B-7	P3	CD	

Fig. 5.16. Event Tree for Double Pump Fault - Group B

EVENT TREE FOR: Double Pump Loss of Flow - Group C							31-MAY-91	
Double Pump Loss of Flow - Group C	Scram signal during LOF event	Sufficient control or safety rods drop during LOF event	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage
LF2C	FSIG	FROD	BPHR	DHRS	DHRL			Frequency
						LF2C-1		
						LF2C-2	P2	CSD
						LF2C-3		
						LF2C-4	P2	CSD
						LF2C-5	P2	CSD
						LF2C-6	P3	MCD
						LF2C-7	P3	MCD

Fig. 5.17. Event Tree for Double Pump Fault - Group C

EVENT TREE FOR: Double Pump Loss of Flow - Group D

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Double Pump Loss of Flow - Group D	Scram signal during LOF event	Sufficient control or safety rods drop during LOF event	Sequence	Class	Fuel Damage	Frequency
LF2D	FSIG	FROD				
			LF2D-1	P4	MCD	
			LF2D-2	P3	CD	
			LF2D-3	P3	CD	

Fig. 5.18. Event Tree for Double Pump Fault - Group D

EVENT TREE FOR: Double Pump Loss of Flow - Group E

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Double Pump Loss of Flow - Group E	Scram signal during LOF event	Sufficient control or safety rods drop during LOF event	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
LF2E	FSIG	FROD	BPHR	DHRS	DHRL	LF2E-1			
						LF2E-2	P2	CSD	
						LF2E-3			
						LF2E-4	P2	CSD	
						LF2E-5	P2	CSD	
						LF2E-6	P3	CD	
						LF2E-7	P3	CD	

Fig. 5.19. Event Tree for Double Pump Fault - Group E

Fig. 5.20. Event Tree for Double Pump Fault - Group F

5.3.6 Event Trees for Loss of Vital Electrical Power

5.3.6.1 Initiating Event Description

There are two types of electrical power which when lost lead to scram of the reactor, normal power (2400 V) and constant power (125 V AC). The effects of the loss of normal power are that both pumps will coast down with the full inertia of the motor generator set. The loss of constant power panels 1 and 2 leads to deenergizing of the clutches to motor generator sets 1 and 2 respectively. This is discussed in detail in the system notebooks for the pumps and motor generator sets.

5.3.6.2 Normal Plant Response and Success Criteria

In the case of loss of normal power there is a scram breaker in string A which directly initiates scram. In the case of loss of constant power panels the resulting loss of control power will cause any number of fail safe scram systems to give a scram signal. In both cases the LOF and SOT subsystems will also generate a scram signal because the direct effect is to cause a flow transient. In the case of Loss of Normal Power the emergency power is used to reduce the secondary flow to minimum conditions thus disabling the BOP as a decay heat removal mechanism.

5.3.6.3 Event Trees and Accident Sequences

Figures 5.21 and Fig. 5.22 show the event trees that apply. They are identical to those for flow transients. As shown in 5.4.2.3, the LONP transient will not cause core damage, for either protected or unprotected transients. In the protected case, the peak cladding temperature is well below the onset of eutectic attack threshold temperature, allowing the reactor to retain unlimited operating capabilities. In the unprotected case, the peak cladding temperature is great enough to lead to the reactor being in the unacceptable operating region with a potential for damage to experiments, PED; however, sodium boiling does not occur.

EVENT TREE FOR: Loss of Normal Power

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Loss of Normal Power	Scram signal during LOF event	Sufficient control or safety rods drop during LOF event	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
LONP	FSIG	FROD	DHRS	DHRL
LONP-1	P2	CSD	
					LONP-2	P2	CSD	
					LONP-3	P2	CSD	
					LONP-4	P3	PED	
					LONP-5	P3	PED	

Fig. 5.21. Event Tree for Loss of Normal Electric Power

EVENT TREE FOR: Loss of Constant Power

31-MAY-91

Fig. 5.22. Event Tree for Loss of Constant Power

5.3.7 Event Trees for Slow Reactivity Insertion

5.3.7.1 Initiating Event Description

As discussed earlier the initiating events that lead to reactivity insertion are those due to motion of control or safety rods either singly or in combination. Other system faults which might result in reactivity insertion through reactivity feedback are treated separately (steam leak or pump overspeed leading to an overcooling and reactivity insertion via the temperature coefficient of reactivity). Other means of reactivity insertion which can be envisaged are some rearrangement of fuel on either the subassembly scale or a whole core motion (stick/slip).

5.3.7.2 Normal Plant Response and Success Criteria

Just as with other transients the normal plant response is to detect the transient, in this case with a 2 of 3 logic high flux monitoring system. The success criterion is to scram sufficient rods to shut the reactor down, sufficient being defined by the magnitude of the total insertion. If the scram signal is generated and the rods unlatch from the drive mechanism, a rod insertion initiator is terminated but it then requires rods to drop to achieve safe and stable shutdown. Decay heat removal is carried out in the usual way.

5.3.7.3 Event Trees and Accident Sequences

The event trees for reactivity insertion are shown in Figs. 5.23 to 5.26.

The definitions of the different slow reactivity insertion cases are given below with the associated damage category if unprotected.

RISA	<20¢	ND
RISB	20¢ - 65¢	MCD
RISC	65¢ - 100\$	CD
RISD	> 1.00\$	CD

The basis for these assignments is given in 5.4.2.5.

EVENT TREE FOR: Slow Reactivity Insertion - Group A (< 0.20 \$)

31-MAY-91

Fig. 5.23. Event Tree for Reactivity Insertion - RISA

Fig. 5.24. Event Tree for Reactivity Insertion - RISB

Fig. 5.25. Event Tree for Reactivity Insertion - RISC

EVENT TREE FOR: Slow Reactivity Insertion - Group D (> 1.10 \$)

31-MAY-91

Slow Reactivity Insertion - Group D (> 1.10 \$)	Scram signal during TOP event	Sufficient control or safety rods drop during TOP event	Primary pumps keep running	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
RISD	RSIG	RROD	PRUN	BPHR	DHRS	DHRL	RISD-1			
							RISD-2	P2	CSD	
							RISD-3			
							RISD-4	P2	CSD	
							RISD-5	P2	CSD	
							RISD-6	P3	CD	
							RISD-7	P3	CD	
							RISD-8	P3	CD	

Fig. 5.26. Event Tree for Reactivity Insertion - RISD

5.3.8 Event Trees for Fast Reactivity Insertions

5.3.8.1 Initiating Event Definition

The initiating event being considered here is any form of reactivity insertion that can insert sufficient reactivity quickly. In the case of a large reactivity input the amount is large enough to cause damage even if scram occurs. The only causative mechanisms that have been identified are vessel failures which lead to a 2-in (50.8 mm) drop in the core with respect to the control rods. The small reactivity insertion, of less than 0.35\$, can be identified with a substantial rearrangement of material in an experimental assembly.

5.3.8.2 Normal Plant Response and Success Criteria

The normal plant response is to detect power change, scram and remove the decay heat. The success criteria are shown in Table 5.20.

5.3.8.3 Event Trees and Accident Sequences

Fig. 5.27 gives the event tree for a small fast reactivity insertion. In the case of large fast reactivity insertion (RIFL) the event tree of Fig. 5.28 describes the accident sequence. By definition of the initiating event, damage occurs in both the protected and unprotected cases, and this belongs to the P1 category.

Fig. 5.27. Event Tree for Small Reactivity Insertion

EVENT TREE FOR: Fast Reactivity Insertion (> 1.00 \$)

31-MAY-91

Fast Reactivity Insertion (> 1.00 \$)	Sequence	Class	Fuel Damage	Frequency
RIFL				
	RIFL-1	P1	CD	

Fig. 5.28. Event Tree for Large Reactivity Insertion

5.3.9 Event Trees for Loss of Primary Heat Sink

5.3.9.1 Initiating Event Description

This class of initiating events includes all faults on the secondary sodium and balance of plant that can be construed as leading to a significant reduction a total loss of the primary heat sink. Specifically excluded is the steam generator tube rupture, which is treated separately.

5.3.9.2 Normal Plant Response and Success Criteria

In the event of a fault on the balance of plant the operators perform a manual scram. Since the system response to an unprotected loss of heat sink is reduction of power no automatic scram should take place. The decay heat is removed by the shutdown coolers. For those faults which leave partial heat removal capability it is assumed that the operators dump the secondary sodium, others such as sodium leak and loss of feedwater result in mandatory dump of sodium.

5.3.9.3 Event Trees and Accident Sequences

The event trees associated with the loss of primary heat sink are shown in Figs. 5.29 and 5.30. In the event of scram the shutdown coolers must function to remove decay heat; failure to do so leads to protected loss of total heat sink. Failure to scram leads to the unprotected loss of primary or total heat sink and this transient has been shown by direct experimentation to lead to a benign outcome. In fact this then bears upon which automatic protection was removed from the EBR-II scram system.

EVENT TREE FOR: Loss of Secondary System Heat Removal

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Loss of Secondary System Heat Removal	Manual SCRAM Under LOHS Conditions	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
TSEC	MRSS	DHRS	DHRL				
				TSEC-1			
				TSEC-2	P2	CSD	
				TSEC-3	P2	CSD	
				TSEC-4	P3	ND	

Fig. 5.29. Event Tree for Loss of Primary Heat Sink (TSEC)

Fig. 5.30. Event Tree for Loss of Primary Heat Sink (TBOP)

5.3.10 Event Trees for Overcooling

5.3.10.1 Initiating Event Description

This initiating event class includes those means by which the primary system could be cooled down. This represents a threat in the unscrammed case because the cooling of the inlet leads to reactivity insertion. There are three types of events that contribute to this category, but only the steam leak has the potential to lead to core damage. The other two events caused by either primary or secondary pump overspeed only lead to damage through the Loss of Heat Sink.

5.3.10.2 Normal Plant Response and Success Criteria

The response of the system to an overcooling event is that of any other overpower, that is to say the high flux protection should generate a scram signal. The scram signal can direct the accident to a loss of heat sink by one of two paths; either successfully scrambling or by tripping the secondary pump.

5.3.10.3 Event Trees and Accident Sequences

The event tree for the overcooling sequence is exactly the same as that for reactivity insertion except that the additional possibility that the secondary pump fails to trip must be included. The three cases are given in Figs. 5.31, 5.32 and 5.33.

EVENT TREE FOR: Overcooling by Major Steam Leak

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Fig. 5.31. Event Tree for Overcooling (Steam Leak)

Fig. 5.32. Event Tree for Overcooling (Primary Pump Overspeed)

EVENT TREE FOR: Overcooling by Secondary Pump Overspeed

31-MAY-91

Overcooling by Secondary Pump Overspeed	Scram signal during TOP event	Sufficient control or safety rods drop during TOP event	Secondary pump trips	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
OCSP	RSIG	RROD	SPTR	BPHR	DHRS	DHRL	OCSP-1			
							OCSP-2	P2	CSD	
							OCSP-3			
							OCSP-4	P2	CSD	
							OCSP-5	P2	CSD	
							OCSP-6			
							OCSP-7	P2	CSD	
							OCSP-8	P2	CSD	
							OCSP-9	P3	ND	
							OCSP-10	P3	ND	

Fig. 5.33. Event Tree for Overcooling (Secondary Pump Overspeed)

5.3.11 Event Trees for Steam Generator Tube Rupture

5.3.11.1. Initiating Event Description

The EBR-II steam generator transfers heat from secondary sodium on the shell side to water/steam on the tube side. The duplex heat transfer tube provides the barrier between the sodium and water. A leak in the duplex tube would result in the injection of high pressure steam/water into liquid sodium. A failure in the welds which connect the duplex tubes to the duplex tube sheets results in leakage into the airspace which has leak detectors. An undetected failure in the outer-tube weld to its tube sheet and an undetected failure in the inner-tube weld to its tube sheet are required to obtain a sodium/water reaction in the airspace between the two tube sheets. The vigorous sodium/water reaction would produce heat, hydrogen, and sodium compounds. Left unchecked, the size of the original leak would increase due to self-wastage of the failed steam tube, and other steam generator tubes in the vicinity of the leak would also corrode, producing extensive damage to the steam generator [5.38].

Besides possibly producing extensive damage to steam generator components, a steam generator failure accident in EBR-II could result in damage to other secondary and steam system components. Restoration of the secondary and steam systems could entail considerable repair costs and long-term plant unavailability. As regards to the PRA, more important considerations are the consequences of a steam generator failure accident to the integrity of the primary system. A possible consequence of an accident would be the loss of heat transfer capabilities of the IHX. A more remote outcome of an accident would be physical damage to the IHX due to pressure loading by acoustic pressures generated by the reaction or due to corrosive attack by reaction products.

5.3.11.2 Normal Plant Response and Success Criteria

The normal plant response consists of two parts - detection and mitigation. Detection of the steam generator tube rupture accident is accomplished by

activation of the Hydrogen Meter Leak Detector (HMLD) system or by the activation of flow indicators in the sodium pressure relief system. Mitigation of the accident is achieved passively by the sodium pressure relief system or actively by dumping of sodium and water from the secondary system and the steam generator. The accident will be successfully terminated if detection and mitigation is accomplished before the IHX is damaged. If the steam generator tube rupture endangers the integrity of the IHX by pressure loading or corrosive attack, the accident is classified as a P5 plant transient. Fuel damage classification is dependent upon the operating status of the reactor. If the IHX fails, contaminants will pass from the secondary into the primary system. The contaminants could conceivably cause plugging of subassembly coolant channels or control rod drivelines. If the reactor is scrammed before the IHX fails, the possible plugging of reactor coolant channels will not damage the fuel (fuel damage class ND). If the reactor is at power when the IHX fails, reactor coolant channels may become plugged creating local hot spots in the core. In this case, the accident is given a core/structural damage (CSD) classification.

5.3.11.3 Event Trees and Accident Sequences

The event trees for a steam tube rupture failure in a superheater module and an evaporator module are shown in Figs. 5.34 and 5.35, respectively. A tube rupture in a superheater module is different from a tube rupture in an evaporator module because there is a direct pressure relief path from the superheater modules that passes through the IHX, which facilitates the transfer of contaminants to the IHX. A detailed discussion of the top events and the accident sequences is found in Appendix D.3.

EVENT TREE FOR: Superheater Tube Failure

31-MAY-91

Fig. 5.34. Event Tree for A Steam Tube Rupture in A Superheater Module

EVENT TREE FOR: Evaporator Tube Rupture

31-MAY-91

Fig. 5.35. Event Tree for A Steam Tube Rupture in An Evaporator Module

5.3.12 Event Tree for Delayed Loss of Circulation

5.3.12.1 Initiating Event Description

This event includes all causes of detected ruptures and leaks of primary sodium sufficient to uncover the Z-pipe to IHX connection. The concern is that, having detected a sodium leak and having shutdown successfully (there can be many hours to shutdown before events of consequence occur), if the pumps stop operating an argon cover gas bubble forms in the outlet sodium Z-pipe thus stopping natural circulation heat removal.

5.3.12.2 Normal Plant Response and Success Criteria

The definition of the initiating event assumes that the reactor is successfully scrammed. The important question is whether the primary pumps are still operating for one hour after shutdown. Decay-heat removal is carried out by shutdown and shield cooling systems.

5.3.12.3 Event Tree and Accident Sequences

The event tree in Fig. 5.36 shows the sequences for this initiating event. Core damage can be avoided if deenergizing the pumps is delayed long enough after the reactor shutdown. If the loss of pumping power is delayed 1 hour after shutdown, the core will not be damaged however the peak coolant temperature predicted is only slightly less than the sodium boiling temperature. The Emergency Procedures allow the operators to remove the pumping power after a minimum of one hour. This will be assumed to take place at one hour. Therefore, even if decay heat removal is successful, there is potential for damage to experiments.

EVENT TREE FOR: Loss of Circulation - Delayed

31-MAY-91

Loss of Circulation - Delayed	Pumps keep running for 1 hour	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
LCDL	PKRU	DHRS	DHRL				
				LCDL-1	P4	PED	
				LCDL-2	P2	CSD	
				LCDL-3	P2	CSD	
				LCDL-4	P1	CD	

Fig. 5.36. Event Tree for Delayed Loss of Circulation

5.3.13 Event Trees for Support System Failures

5.3.13.1 Initiating Event Description

This category includes all events which lead to shutdown with the decay heat removal function impaired. Included are loss of shutdown cooler (TSDC), loss of shield cooling (TSHC). Loss of instrument air is a more wide ranging transient whose primary effect is loss of feedwater control.

5.3.13.2 Normal Plant Response and Accident Sequences

The normal plant response is to remove the decay heat via the BOP unless the initiating event itself precludes this. In this eventuality the shutdown coolers are expected to carry the load. The success criteria are identical to those for the loss of primary heat sink.

5.3.13.3 Event Trees and Accident Sequences

The event tree for these events is given in Figs. 5.37, 5.38 and 5.39. The accident sequences revolve around either loss of primary heat sink or loss of heat sink.

Fig. 5.37. Event Tree for Transient Loss of Shield Cooling

EVENT TREE FOR: Transient - Unrecoverable Loss of a Shutdown Cooler

31-MAY-91

Transient - Unrecoverable Loss of a Shutdown Cooler	Balance of plant decay heat removal	Decay heat removal short term (1 sd cooler)	Decay heat removal long term (1 sd cooler + shield cooling)	Sequence	Class	Fuel Damage	Frequency	
TSDC	BPHR	DRSC	DRLA					
				TSDC-1	P2	CSD		
				TSDC-2				
				TSDC-3	P2	CSD		
				TSDC-4				
				TSDC-5	P2	CSD		

Fig. 5.38. Event Tree for Transient Loss of Shutdown Coolers

Fig. 5.39. Event Tree for Transient Loss of Instrument Air

5.3.14 Event Trees for Shutdowns

5.3.14.1 Initiating Event Description

This category includes all events which call for normal shutdown, anticipatory shutdown and spurious scram where no safety threat exists. In all cases the threat is that having shutdown, the decay heat removal systems will fail to perform as required and core damage will occur. As before in the loss of primary heat sink description the damage is caused by elevated temperatures throughout the primary system and it is categorized as CSD (Core Structural Damage).

5.3.14.2 Normal Plant Response and Success Criteria

The normal plant response is to remove the decay heat via the BOP unless the initiating event itself precludes this. In this eventuality the shutdown coolers are expected to carry the load. The success criteria are identical to those for the loss of primary heat sink. In the case of the long shutdown an extended mission time is considered.

5.3.14.3 Event Trees and Accident Sequences

The event trees for these events are given in Figs. 5.40 to 5.43. The accident sequences revolve around either loss of primary heat sink or loss of heat sink.

Fig. 5.40. Event Tree for Scram

EVENT TREE FOR: Anticipatory Shutdown

31-MAY-91

Anticipatory Shutdown	Balance of plant decay heat removal	Decay heat removal short term	Sequence	Class	Fuel Damage	Frequency
ANTC	BPHR	DHRS				
			ANTC-1			
			ANTC-2			
			ANTC-3	P2	CSD	

Fig. 5.41. Event Tree for Anticipatory Shutdown

EVENT TREE FOR: Short Shutdown

31-MAY-91

Short Shutdown	Balance of plant decay heat removal	Decay heat removal short term	Sequence	Class	Fuel Damage	Frequency
SHDS	BPHR	DHRS				
			SHDS-1			
			SHDS-2			
			SHDS-3	P2	CSD	

Fig. 5.42. Event Tree for Shutdown Short

EVENT TREE FOR: Long Shutdown

31-MAY-91												
Long Shutdown			Decay heat removal short term		Decay heat removal long term		Sequence	Class	Fuel Damage	Frequency		
SHDL			DHRS		DHRL							
							SHDL-1	P2	CSD			
							SHDL-2					
							SHDL-3	P2	CSD			

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Fig. 5.43. Event Tree for Shutdown Long

5.3.15 Event Trees for Local Faults

5.3.15.1 Initiating Event Description

This initiating event class includes all local faults, specifically, FLTA, complete subassembly inlet or outlet blockage, FLTB, simultaneous failure of large number of pins within a subassembly, MDLF, metal driver subassembly pin failure, MELF, metal experiment subassembly pin failure, OELF, non-metal experiment subassembly pin failure, SFLE, subassembly loading error, and SFPB, subassembly partial blockage. Based on the EBR-II design and analyses, FLTA is not considered to be physically possible and FLTB does not lead to damage consequences, and is therefore not of concern.

5.3.15.2 Normal Plant Response and Success Criteria

The normal plant response is to detect failures using one of the FERD or GLASS signals or by reactivity change, whereupon shutdown is initiated if the operators deem it necessary. Certain conditions require scram, e.g. DN signal > 3000 counts per second.

5.3.15.3 Event Trees and Accident Sequences

The event sequence from breach initiation in a fuel element to large-scale damage is a complex series that involves phenomenological considerations and automatic and operator responses to conditions at specific times. Given the initiating event of an element breach, the next concern is the effect of the breach on neighboring elements, the classical pin-to-pin propagation concern. This is the first branch point considered in the event tree and is a phenomenological consideration. The general top event is No Pin-to-Pin Breach Propagation (NPPP). A positive response to this top event means cladding breach does not cause neighboring elements to breach, thereby leading to a state of no concern in the PRA context.

If the breach of one element leads to a breach in neighboring elements, the nature of the effect leads to the next branch point, is the breach propagation self-limiting or not, another phenomenological consideration. This top event is designated as Self-Limiting Pin-to-Pin Breach Propagation (SLPP). A positive response to this top event means the initial breach would lead to the additional breach of at most three or four neighboring elements and is a state of no concern in the PRA context since it is limited in nature, a situation EBR-II is designed to accommodate.

A negative response to the SLPP top event is the path to continued propagation of cladding breaches within the subassembly. With each new breach, a breach of neighboring elements occurs. The next top event is a response to the proceeding phenomenological events. This top event addresses the detectability of this propagating breach state and is designated Failure Detection (DETK). A positive response means the breaches are detected by the systems included in EBR-II including the Fuel Element Rupture Detector (FERD), the Germanium-Lithium Argon Scanning System (GLASS), and possible reactivity changes.

The consideration of whether a Reactor Shutdown Signal (SHTD) is initiated is the next top event. A reactor shutdown signal may be initiated by the reactor operator or an automatic system. In the case of an element breach initiated local fault, there are few systems that would automatically generate a scram signal. An increase in reactor power to the overpower trip setting (110% nominal) would provide an automatic scram. Other conditions will be governed by operator response and the associated operating and emergency procedures.

The final top event is again one of phenomenological considerations. If unlimited fuel element breach propagation occurs, the determination of subassembly-to-subassembly propagation must be made. The final top event is therefore No Subassembly-to-Subassembly Propagation (NSSP). If there is only a single breach or limited breach propagation, the integrity of the subassembly boundary is maintained.

The event trees for subassembly scale local faults, such as a partial blockage or orificing error and a loading error, are based on those for the single pin breach initiated local faults. The

subassembly fault changes the operating conditions, lower flows or higher temperatures, and thereby changes the element operational lifetime. This change in conditions could then initiate element breaches within the subassembly. The first top event in this tree is therefore Subassembly Fault: No Pin Breach (SNPB). After this element breach consideration, the other top events are identical to the single element breach initiated local fault tree described above. Because the conditions are different than a normal driver or experimental subassemblies, the various branch ratios may be different, but the structure is the same.

The event trees are given in Figs. 5.44 to 5.48. Damage categories are assigned based on the state of the subassembly and its neighbors. Core Damage (CD) results if the subassembly boundary is breached and neighboring subassemblies are involved. If the subassembly boundary remains intact, but there are extensive pin breaches within the subassembly, the state is defined to be Minor Core Damage (MCD). If there are few pin breaches, the state is defined as No Damage (ND) since this is a normal operating state for EBR-II.

EVENT TREE FOR: Local Faults: Metal Driver Breach Initiator

31-MAY-91

Local Faults: Metal Driver Breach Initiator	No Pin-to-Pin Propagation	Self-Limiting Pin Propagation	Detect in Time to Shutdown and Prevent Propagation	Shutdown of Reactor	No Subassembly-to- Subassembly Propagation	Sequence	Class	Fuel Damage	Frequency
MDLF	NPPD	SLPD	DETP	SHTS	NSSP
MDLF-1			
						MDLF-2			
						MDLF-3			
						MDLF-4	P6	CD	
						MDLF-5	P6	MCD	
						MDLF-6	P6	CD	
						MDLF-7	P6	MCD	
						MDLF-8	P6	CD	

Fig. 5.44. Event Tree for Local Fault - Driver Pin Failure

EVENT TREE FOR: Local Faults: Metal Experiment Breach Initiator

31-MAY-91

Local Faults: Metal Experiment Breach Initiator	No Pin-to-Pin Propagation	Self-Limiting Pin Propagation	Detect in Time to Shutdown and Prevent Propagation	Shutdown of Reactor	No Subassembly-to- Subassembly Propagation	Sequence	Class	Fuel Damage	Frequency
MELF	NPPE	SLPE	DETP	SHTS	NSSP				
						MELF-1			
						MELF-2			
						MELF-3			
						MELF-4	P6	CD	
						MELF-5	P6	MCD	
						MELF-6	P6	CD	
						MELF-7	P6	MCD	
						MELF-8	P6	CD	

Fig. 5.45. Event Tree for Local Fault - Metal Experiment Pin Failure

EVENT TREE FOR: Local Faults: Oxide Experiment Breach Initiator

31-MAY-91

Local Faults: Oxide Experiment Breach Initiato	No Pin-to-Pin Propagation	Self-Limiting Pin Propagation	Detect in Time to Shutdown and Prevent Propagation	Shutdown of Reactor	No Subassembly-to- Subassembly Propagation	Sequence	Class	Fuel Damage	Frequency
OELF	NPPO	SLPE	DETP	SHTP	NSSP	OELF-1			
						OELF-2			
						OELF-3			
						OELF-4	P6	CD	
						OELF-5	P6	MCD	
						OELF-6	P6	CD	
						OELF-7	P6	MCD	
						OELF-8	P6	CD	

Fig. 5.46. Event Tree for Local Fault - Oxide Experiment Pin Failure

EVENT TREE FOR: Subassembly Local Fault from Partial Blockage

31-MAY-91

Subassembly Local Fault from Partial Blockage	No Pin Breach	No Pin-to-Pin Propagation	Self-Limiting Pin Propagation	Detected in Time to Shutdown and Prevent Propagation	Shutdown of Reactor	No Subassembly-to- Subassembly Propagation	Sequence	Class	Fuel Damage	Frequency
SFPB	SNPB	NPPB	SLPE	DETS	SHTS	NSSS				
							SFPB-1			
							SFPB-2			
							SFPB-3			
							SFPB-4			
							SFPB-5	P6	CD	
							SFPB-6	P6	MCD	
							SFPB-7	P6	CD	
							SFPB-8	P6	MCD	
							SFPB-9	P6	CD	

Fig. 5.47. Event Tree for Local Fault - Partial Subassembly Blockage

EVENT TREE FOR: Subassembly Local Fault from Loading Error

31-MAY-91

Subassembly Local Fault from Loading Error	No Pin Breach	No Pin-to-Pin Propagation	Self-Limiting Pin Propagation	Detect in Time to Shutdown and Prevent Propagation	Shutdown of Reactor	No Subassembly-to-Subassembly Propagation	Sequence	Class	Fuel Damage	Frequency
SFLE	SNPB	NPPO	SLPE	DETP	SHTS	NSSS	SFLE-1			
							SFLE-2			
							SFLE-3			
							SFLE-4			
							SFLE-5	P6	CD	
							SFLE-6	P6	MCD	
							SFLE-7	P6	CD	
							SFLE-8	P6	MCD	
							SFLE-9	P6	CD	

Fig. 5.48. Event Tree for Local Fault - Loading Error

5.4 Core Damage Analysis

The assignation of damage categories in the previous section has been based upon calculations using a model of EBR-II with the SASSYS code. This enabled a consistent set of damage calculations to be undertaken for core loadings typical of the present loadings.

In the following, details of the SASSYS model of EBR-II are given together with results for various of the important transients.

5.4.1 SASSYS Model of EBR-II

The SASSYS model of EBR-II [5.41] has been extensively validated with the Shutdown Heat Removal Tests (SHRT) [5.41, 5.42, 5.43, 5.44]. Among the transients analyzed with SASSYS are protected LOF experiments (SHRT 8A [5.44] and SHRT 17 [5.42, 5.43]), mild reactor flow and inlet temperature perturbation experiments (SHRT 25 and SHRT 26 [5.43]), an unprotected LOF experiment (SHRT 45 [5.41, 5.43]), and an unprotected loss-of-heat-sink experiment (SHRT B302 [5.41]). The SASSYS model of EBR-II is based on a mixed Mark-II and Mark-III core loading and uses best estimate values for reactivity feedback characteristics [5.33].

The SASSYS model of EBR-II was modified to aid determination of core damage levels during upset conditions. Specifically, channels were modified to represent hot and core-average fuel pins. Hot channels represent the hottest fuel pin and coolant subchannel allowed in any driver-fuel subassembly by EBR-II Technical Specifications [5.24] (peak coolant temperature = $465^{\circ}\text{C} = 1050^{\circ}\text{F}$). Only a few fuel pins operate near Technical Specifications limits in any core loading. If during an accident simulation, thermal conditions in hot channels are severe enough to produce localized boiling, fuel melting, or fuel pin failure, the accident is classified as causing minor core damage, (MCD). Only a few isolated pins would be damaged during the accident. If the accident is severe enough to produce boiling, fuel melting, or pin failure in average channels, the accident is classified as causing core damage, (CD). A core damage accident would result in large numbers of damaged fuel pins, and might result in core relocation.

If hot channels are not damaged, the accident is classified as causing no core damage, (ND). This class of accidents is further divided into three subgroups based on operational considerations as stipulated by the Technical Specifications. If the peak clad temperature reached during the transient is below 710°C (1310°F), the reactor will retain unlimited capability to operate. This temperature is a conservative estimate of the onset temperature of eutectic formation between clad and metal fuel. If the hottest sections of cladding are predicted to reach temperatures within the range between 710°C and 816°C (1500°F) for less than 60 s, the reactor is classified as operating in the limited capability region. If the clad temperature in the hottest pins exceeds 816°C or lies between the eutectic temperature and 816°C for greater than 60 s, the reactor will pass into the unacceptable region of operation. In this case, no damage to driver fuel will occur, however there is the potential for damage to experiments, (PED). In the calculations shown in this section the effects of coolant boiling have been suppressed and the coolant temperature histories shown therefore continue into the boiling regime as if the sodium were still single phase. This is adopted as a convenience to display the severity of the temperature transient.

Core damage states are dependent upon the occurrence of coolant boiling, fuel melting, or fuel pin failure in hot or average channels. Fuel pin failure refers to breach of the cladding caused by eutectic attack, fission-gas pressure induced strain, or a combination of both. Fuel melting is considered to be a damage indicator although it is possible for fuel melting to occur without the subsequent release of fission products via fuel pin failure. In the case of fuel melting without cladding failure, the cladding will still be damaged enough by elevated cladding temperatures and fission-gas pressures to severely shorten the useful lifetime of the fuel pin. Thus although fuel pins may not actually "fail" during an accident that results in fuel melting, accident resolution would require the removal and examination of any subassemblies which are suspected to have experienced fuel melting.

5.4.2 Accident Calculations

5.4.2.1 Loss-of-Flow Events

For calculational purposes, the pump coastdowns have been divided into four groups based on coastdown duration. The first group consists of M/G input faults, which have a pump speed stop time of about 80 s. Pump binding faults are included in this group and are assumed, for simplicity, to have the same coastdown behavior as the M/G input faults. The second group consists of M/G output faults, which have a stop time of about 30-35 s. The third coastdown group consists of clutch faults, which have a stop time of about 23 s. The fourth group consists of pump seizures, which are assumed to have an essentially 0 s stop time. Figure 5.49 shows representative pump speed coastdown curves for the four groups. The response of the reactor shutdown system determines whether the LOF transient is protected (i.e., reactor scram is successful) or unprotected (reactor fails to scram).

In previous studies, the thermal response of cores loaded with Mark-II [5.45, 5.46] and Mark-III [5.47, 5.48] fuel elements to specific loss-of-flow transients has been investigated. Reactor experiments at full power and flow conditions have demonstrated the viability of natural circulation to adequately remove decay heat and reactor feedbacks to shutdown the reactor during protected [5.25] and unprotected [5.10], slow coastdown, LOF transients. In the current investigation, a systematic study of core damage potential for a core predominantly loaded with Mark-III fuel elements was performed for the various pump faults, under protected and unprotected conditions. The model utilizes measurements of primary pump speed made on actual pump coastdown experiments at EBR-II [5.56, 5.57] to simulate the flow coastdowns of all postulated pump faults except for pump seizures. The instantaneous stopping of the pump during a pump seizure is a conservative assumption that is physically impossible because it would require an instantaneous decrease in the momentum of the pump impeller. The auxiliary pump is assumed to be inoperative during the LOF transients. The secondary pump is assumed to be tripped 6 s into the transient. During protected transients, the reactor is assumed to be tripped 1.74 s after initiation of the transient. The control rods are assumed to

Fig. 5.49. Pump Coastdowns in EBR-II

have a 0.252 s drop time. The reactor trip characteristics are typical of those used in previous calculations [5.47].

For the LOF calculations discussed here, representative coolant temperatures are shown in accompanying figures. Plots and data listings of reactor power and primary system flowrate profiles for the LOF calculations may be found in reference [5.49]. It should be noted that a calculational anomaly of SASSYS limits the representation of centrifugal pump coastdown curves to only ten points. This results in uncharacteristically rough temperature response during long coastdown LOF events (e.g. Fig. 5.50 and 5.51).

EBR-II Technical Specifications [5.24] require the operation of the auxiliary pump to guarantee that driver fuel element temperatures remain in the region of unlimited operating capability during an anticipated LOF event, which is defined as a protected, double clutch fault transient. The reactor remains in the unlimited operating capability region as long as the peak driver cladding temperature does not surpass the fuel/clad eutectic temperature limit (710°C). The reactor can operate beyond the unlimited operating capability region (i.e. in the limited and unacceptable operating regions) without incurring fuel damage. Although the auxiliary pump acts to lower peak temperatures during LOF transients, this effect is not large enough to change the damage state of the reactor during any LOF event. Because the PRA is only concerned with the damage state of the driver fuel and not the operational capabilities of the reactor, no credit is taken for operation of the auxiliary pump in the accident analysis calculations. Because the auxiliary pump is assumed to be inoperative, it was hypothesized that the scenario in which a double pump LOF event occurs after the reactor is shutdown might pose a concern. In particular, if a LOF occurs when the reactor is nominally isothermal, natural circulation may take time to develop, during which local core temperatures may reach the sodium boiling point. This scenario, termed a delayed LOF event, has been analyzed (Appendix D.4) and found to not produce significantly elevated temperatures.

Single Pump Fault Events

Single pump fault events are due to mechanical or electrical failures of one of the primary pumps. A special case arises if the reactor is operating on electrical power supplied by the turbine-generator with the EBR-II site disconnected from the INEL loop, i.e. the turbine-generators is supplying the onsite, or hotel, load. In this case, the scram signal generated by the low flow indicators of the reactor shutdown system will generate a turbine trip and a trip of the turbine-generator output breaker. This results in a loss of normal power (2400 V) to the other pump which induces a M/G input fault. Although the M/G input fault of the second pump will occur a few seconds following the failure of the first pump, the second pump trip is conservatively assumed to occur at the same time as the failure of the first pump. A single pump failure with the reactor on hotel load is modeled in the same manner as any double pump fault which includes a M/G input failure of one of the primary pumps.

Calculated core damage levels for protected and unprotected single pump faults are collected in Table 5.23. The columns in the table list the four types of single pump faults. The rows of the table divide the results into four cases, depending on whether the transient is protected or unprotected and whether the reactor is on hotel load or outside power. If the single pump failure occurs while the reactor is on outside power, the other pump continues to operate normally. If the reactor is successfully scrammed, a single pump fault does not produce core damage, even if the reactor is on hotel load. Peak cladding temperatures do not reach the eutectic limit (710°C) during protected single pump faults, unless a pump seizure occurs while the reactor is on hotel load. For a single pump seizure with the reactor on hotel load, the cladding temperature in the hot channel momentarily (< 1 s) surpasses the eutectic temperature limit and reaches a peak temperature of 725°C . If the conservatism inherent in this calculation (instantaneous stopping of the seized pump and simultaneous M/G input failure of the other pump) are replaced by more realistic conditions, the peak cladding temperature would not exceed the eutectic temperature. Figure 5.50 contains plots of the hot pin coolant temperature during a protected single pump seizure with the reactor operating either on hotel load or outside power, clad temperatures are typically $5\text{-}10^{\circ}\text{C}$ hotter than the coolant.

Table 5.23. Core Damage Levels During Single Pump Faults

	Pump Seizure Fault	M/G Clutch Fault	M/G Output Fault	M/G Input Fault
Protected Single Pump Fault, Reactor on Outside Power	No Core Damage/Unlimited Operating Capability	No Core Damage/Unlimited Operating Capability	No Core Damage/Unlimited Operating Capability	No Core Damage/Unlimited Operating Capability
Protected Single Pump Fault, Reactor on Hotel Load	No Core Damage/Limited Operating Capability	No Core Damage/Unlimited Operating Capability	No Core Damage/Unlimited Operating Capability	No Core Damage/Unlimited Operating Capability
Unprotected Single Pump Fault, Reactor on Outside Power	No Core Damage/Unlimited Operating Capability	No Core Damage/Unlimited Operating Capability	No Core Damage/Unlimited Operating Capability	No Core Damage/Unlimited Operating Capability
Unprotected Single Pump Fault, Reactor On Hotel Load	Minor Core Damage	Minor Core Damage	Minor Core Damage	No Core Damage/Unacceptable Operating Region

Fig. 5.50 Hot Pin Coolant Temperature during Protected Single Pump Faults

For unprotected single pump faults, if the reactor is operating on hotel load, minor core damage occurs, unless the single pump failure is due to a M/G input fault when only PED is predicted. Minor core damage occurs because the coolant temperature in the hot channels barely reaches the saturation temperature (see Fig. 5.51). If an unprotected single pump fault occurs while the reactor is operating on normal power, core damage levels will not be reached and the reactor will retain unlimited operating capabilities, for any coastdown type. Figure 5.51 contains plots of the hot pin coolant temperature during the most severe event, an unprotected single pump seizure with the reactor operating either on hotel load or outside power.

It should be noted that the calculated hot pin coolant temperature during an unprotected single pump seizure with the reactor on hotel load momentarily surpasses the sodium saturation temperature. Although SASSYS contains a model for two-phase sodium flow, it was not implemented during these calculations. Instead the coolant continued to be modeled as a single-phase fluid when the saturation temperature was reached. This simplification was taken because the main concern for this analysis was whether the system would reach a failure condition (i.e., boiling), not the effects of a failure on the system. Technically, the temperature plot in Fig. 5.51 should only be drawn up to when the boiling temperature is reached, since calculated temperatures beyond this time are not accurate. Yet calculated coolant temperatures that extend beyond the sodium boiling point are still presented (Figs. 5.51, 5.52, 5.53, 5.54, 5.55, and 5.57), because the degree to which the single phase coolant temperature surpasses the saturation temperature is an indication of the extent of boiling in the system. For instance, although the hot pin coolant at the top of the core reaches the boiling point during an unprotected single pump seizure with the reactor on hotel load, one would expect only localized vapor formation because the single phase coolant temperature rapidly falls below the boiling point.

In conclusion, all single pump faults, protected and unprotected do not in themselves lead to any form of core damage and all lie in the unlimited capability region unless coastdown of the second pump occurs. In the case of protected single pump faults with simultaneous slow coastdown of the other pump the only difference arises in the case of pump seizure where the conservative calculation causes the limited capability region to be entered for a time of < 1 s.

Fig. 5.51. Hot Pin Coolant Temperature during Unprotected Single Pump Faults

The only cases where any form of damage can result are when the single pump fault is combined with an M/G input coastdown of the second pump due to operating on hotel load and failure to scram.

Double Pump Fault Events

In the analysis of initiating event frequencies, double pump faults have been defined as the loss of one primary pump followed by the loss of the second pump within a 6 minute interval. Although the failure of the second pump can occur anytime within the 6 minute interval, it is conservatively assumed to fail simultaneously with the first pump in the transient analysis.

Calculated core damage levels for protected double pump faults are collected in Table 5.24. The rows in the table list the coastdown behavior of one pump while the columns list the coastdown behavior of the other pump. Only a double pump seizure, which is an extremely low probability event (and a very conservative calculation), produces any level of core damage. The damage produced is minor because it is limited to hot channels. Coolant in average driver fuel pins approach saturation temperature (peak coolant temperature is about 880°C). The other protected double pump faults do not produce core damage, but some produce peak cladding temperatures that place the reactor in the limited operating capability region as stipulated by EBR-II Technical Specifications. The peak cladding temperatures in the hot channels during these transients (double clutch faults and pump seizures with clutch, M/G output, or M/G input faults), surpass the eutectic temperature limit (710°C) by a few degrees (<30°C) for a short period of time (<7 s). Calculations indicate that if the auxiliary pump is operating, peak cladding temperatures would not surpass the eutectic temperature limit for any protected double pump fault event, except for double pump seizures. Figure 5.52 contains plots of the peak coolant temperature in the hot pin channels for identical fault, protected double pump failure events (i.e., both pumps experience pump seizure, clutch, M/G output, or M/G input faults).

Table 5.24. Core Damage Levels During Protected Double Pump Faults

	Pump Seizure Fault	M/G Clutch Fault	M/G Output Fault	M/G Input Fault
Pump Seizure Fault	Minor Core Damage	No Core Damage/Limited Operating Capability	No Core Damage/Limited Operating Capability	No Core Damage/Limited Operating Capability
M/G Clutch Fault	No Core Damage/Limited Operating Capability	No Core Damage/Limited Operating Capability	No Core Damage/Unlimited Operating Capability	No Core Damage/Unlimited Operating Capability
M/G Output Fault	No Core Damage/Limited Operating Capability	No Core Damage/Unlimited Operating Capability	No Core Damage/Unlimited Operating Capability	No Core Damage/Unlimited Operating Capability
M/G Input Fault	No Core Damage/Limited Operating Capability	No Core Damage/Unlimited Operating Capability	No Core Damage/Unlimited Operating Capability	No Core Damage/Unlimited Operating Capability

Fig. 5.52. Hot Pin Coolant Temperature during Protected Double Pump Faults

Calculated core damage levels for unprotected double pump faults are collected in Table 5.25. All unprotected double pump faults result in some level of core damage except for a double MG input fault event. Although the double M/G input fault transient does not produce core damage, peak cladding temperatures are high enough to place the reactor in the unacceptable operating region as stipulated by EBR-II Technical Specifications. Figure 5.53 contains plots of the peak coolant temperature in the hot pin channels for identical fault, unprotected double pump failure events (i.e., both pumps experience clutch, M/G output, or M/G input faults). A calculation for an unprotected double pump seizure was not performed since it would undoubtedly lead to core damage given that the less severe unprotected double clutch failure leads to core damage.

The LOF calculations are based on typical measurements of observable quantities such as reactivity feedback and pump speed coastdown characteristics. The core model is based on a mixed Mark-II and Mark-III core loading (Run 148B), except for the inclusion of a technical specifications limited hot pin channel to model hot spot conditions. Although by definition a double pump failure would occur anytime both pumps fail within a 6 minute period, the pumps are assumed to fail simultaneously since this produces the most damaging effects. Although the pump stopping time during a pump binding incident may be anywhere from a few minutes to a few hours, it is modelled as equivalent to the M/G input fault stopping time. Likewise instantaneous pump stoppage is assumed to occur in a pump seizure, because it produces a worst-case. These assumptions reflect the uncertainty inherent in modeling a double pump failure, a pump binding, or a pump seizure incident.

5.4.2.2 Loss of Normal Power

One of the many reactor experiments performed as part of the Shutdown Heat Removal Test (SHRT) program simulated an unprotected LONP in EBR-II, which had the effect of a double M/G input fault. Figure 5.54 provides a basis for comparing the results of the reactor experiment, SHRT 45, with the unprotected double M/G input fault calculation. Three calculations were performed, and representative temperature plots from them are drawn in the figure. The first calculation consisted of the unprotected double M/G input

Table 5.25. Core Damage Levels During Unprotected Double Pump Faults

	Pump Seizure Fault	M/G Clutch Fault	M/G Output Fault	M/G Input Fault
Pump Seizure Fault	Core Damage	Core Damage	Core Damage	Minor Core Damage
M/G Clutch Fault	Core Damage	Core Damage	Minor Core Damage	Minor Core Damage
M/G Output Fault	Core Damage	Minor Core Damage	Minor Core Damage	Minor Core Damage
M/G Input Fault	Minor Core Damage	Minor Core Damage	Minor Core Damage	No Core Damage/Unaccept- able Operating Region

Fig. 5.53. Hot Pin Coolant Temperature during Unprotected Double Pump Faults

Fig. 5.54. Hot Pin Coolant Temperature during an Unprotected Loss-of-Normal-Power Event

fault. This calculation was repeated for a second time, but this time the auxiliary pump was assumed to be operating with power supplied by battery. The second calculation matches more closely the conditions of SHRT 45 because the auxiliary pump was operating on battery during this test. The third calculation is a best estimate reproduction of SHRT 45 test conditions, and was used to validate the SASSYS model of EBR-II [5.41]. For the double M/G input fault calculations the hot pin coolant temperatures are shown in Figure 5.54, while for the SHRT 45 calculation the top-of-core coolant temperature in the instrumented subassembly XX09, which contained average power driver fuel elements, is shown. Also included in Figure 5.55 is the average top-of-core temperature for SHRT 45 as measured by the top-of-core thermocouples in XX09. The coolant temperature in XX09 during SHRT 45 is significantly lower than the coolant temperature in the hot channels of the double M/G input fault calculations. The peak coolant temperature in the hottest subassembly during SHRT 45 (730°C) [5.10] is still significantly lower than the coolant temperature in the hot channels of the double M/G input fault calculations (900°C). The temperature difference occurs because the reactor core during SHRT 45 was specially configured to eliminate hot (near Technical Specifications limits) subassemblies and because the core inlet temperature of SHRT 45 ($343^{\circ}\text{C} = 650^{\circ}\text{F}$) is lower than that of the LOF calculations ($371^{\circ}\text{C} = 700^{\circ}\text{F}$). Also, the SHRT 45 core contained only Mark-II fuel elements, while the current model contains a large fraction of higher-powered Mark-III fuel elements.

5.4.2.3 Loss of Constant Power

Although a double primary pump clutch fault accompanies a loss of constant power (LOCP), the loss of M/G generator field excitation uncouples the M/G sets from the pump motors to produce a coastdown similar to a M/G output coastdown but possibly with some minor breaking effect as the excitation field goes to zero. For this reason the consequences of a LOCP are considered to be the same as a M/G clutch fault as shown in Figs. 5.53 and 5.54, Tables 5.24 and 5.25. However, the branch point probabilities differ due to the response of the scram system. A LOCP causes a loss of power to the scram circuitry which due to its fail-safe design assures the generation of a scram signal (FSIG) with a probability of 1.0. The mechanical operation of the scram (FROD) also has a higher probability of success than other

RSS elements since it is modelled to include the relays which supply power to the control rod clutches. Constant power is supplied to the clutches, hence an LOCP will remove power from the clutches even if the relays fail to open.

5.4.2.4 Reactivity Insertion Events

For purposes of accident sequence quantification the reactivity insertion initiating events (also called the transient overpower or TOP accidents) are divided into two classes. The first class, slow reactivity insertion events, is composed of single or multiple control rod and safety rod insertions. The second class, instantaneous reactivity insertion events, is composed of reactivity insertions which may be caused by structural failures or other rapid events. The division of the reactivity insertion events into two classes is based on the comparison of the rate of reactivity insertion to the response rate of reactor feedbacks. For the slow reactivity insertion events, the rate of reactivity insertion is much slower than the response rate of the dominant reactor feedback characteristics (reactor fuel, sodium, and steel expansion feedbacks) with the result that the system responds quasi-statistically. For the instantaneous reactivity insertion events, the reactivity is inserted into the reactor in a very short time span compared to the typical response time of reactor feedbacks.

The reactor is assumed to be operating at full power and flow conditions at the beginning of the transient. The transient is initiated by adding a time dependant, external source of reactivity to the core neutronics model. To a first approximation, the amount of reactivity inserted during a rod run-in varies linearly with time because the rod worths vary roughly linearly with rod position and the speed of the rod drive is constant [5.50]. Therefore for the slow reactivity insertion calculations, the rate of reactivity insertion is assumed to be constant. For the instantaneous reactivity insertion calculations, the reactivity is inserted as a 0.1 s ramp in order to aid the numerical convergence of the code. The transients described below are unprotected, because for protected transients a reactor scram will quickly halt the reactivity excursion and the undisturbed reactor coolant flow will quickly cool the core to safe conditions.

Slow Reactivity Insertion Events

A safety rod run-in accident was simulated with SASSYS by using the median insertion rate for the accident (2.9×10^{-3} $\$/s$). Figure 5.55, which shows calculated hot pin fuel, cladding and coolant temperatures as functions of time and the amount of inserted reactivity, reveals that reactor temperatures are, to a first approximation, directly proportional to amount of inserted reactivity. As reactor temperatures increase, negative reactivity feedback effects due to thermal expansion of reactor components begin to compensate for the reactivity insertion. As shown in Figure 5.56, reactor feedbacks respond quickly to the reactivity insertion. The feedback response is rapid enough to keep the reactor nearly critical throughout the course of the transient. An analysis of a single control rod insertion accident was simulated by using the estimated median reactivity insertion rate (4.2×10^{-3} $\$/s$) for such an accident. Calculated reactor temperatures for a single control rod insertion accident are plotted in Figure 5.57. The nearly identical behavior of reactor temperatures from the single control rod insertion calculation (Figure 5.57) and the safety rod insertion calculation (Figure 5.55) shows that the effects of slow reactivity insertion accidents are dependent upon the amount of reactivity inserted but independent upon the rate of reactivity insertion. For instance, referring to Figure 5.57, the reactor would remain in a state of no core damage/unlimited operating capability for an insertion of a control rod whose worth is less than 0.23 $\$$. But for reactivity insertions worth more than 0.23 $\$$, the reactor would enter the no core damage/limited operating capability state because the peak cladding temperature would exceed the eutectic temperature limit. A table was constructed (Table 5.26) which lists the minimum amounts of inserted reactivity that will produce the various core damage states for a given rate of reactivity insertion. The table indicates that for a single control rod insertion at the median rate of reactivity insertion (4.2×10^{-3} $\$/s$), a run-in of a rod worth 0.31 $\$$ or more would cause the fuel in the high burn-up hot fuel pins to begin to melt, which is defined as minor core damage. If the control rod is worth more than 0.59 $\$$, the accident would cause the fuel in average fuel pins to begin to melt. During the slow TOP simulations, minor core damage and core damage states are the result of peak fuel temperatures surpassing the fuel solidus temperature (1214°C for U-10 Zr, 1133°C for U) in hot and average fuel pins, respectively. This is in contrast to the LOF simulations, during which core damage states are produced by coolant boiling.

Hot Pin Temperatures during a
Safety Rod Insertion at 0.0029 $\$/s$

Fig. 5.55. Hot Pin Temperatures during a Safety Rod Insertion at 0.0029 $\$/s$

Fig. 5.56. Reactivity Response during a Safety Rod Insertion at 0.0029 $\$/s$

Hot Pin Temperatures during a
Control Rod Insertion at 0.0042 $\$/s$

Fig. 5.57. Hot Pin Temperatures during a Control Rod Insertion at 0.0042 $\$/s$

Table 5.26. Core Damage Levels during Slow Reactivity Insertion Accidents

Reactivity Insertion Rate (¢/s)	Reactivity Worth Needed to Reach No Core Damage/Limited Operation Level (\$)	Reactivity Worth Needed to Reach Minor Core Damage Level (\$)	Reactivity Worth Needed to Reach Core Damage Level (\$)
0.29 (Median Rate Safety Rod Insertion TOP)	0.22	0.31	0.59
0.42 (Median Rate Single Control Rod Insertion TOP)	0.23	0.32	0.59
0.64 (Maximum Rate Single Control Rod Insertion TOP)	0.25	0.34	0.60
0.84 (Median Rate Double Control Rod Insertion TOP)	0.26	0.36	0.62
1.22 (Maximum Rate Double Control Rod Insertion TOP)	0.34	0.39	0.65

Instantaneous Reactivity Insertions

A series of calculations were performed to investigate the effect of an instantaneous insertion of reactivity. An instantaneous TOP accident is a very low probability event which might be produced by a structural failure of the core support apparatus, for example, causing repositioning of the fueled subassemblies with respect to the control rods. Instantaneous TOP accidents were modeled with SASSYS by inserting a given amount of reactivity during the initial 0.1 s of a transient calculation. The external reactivity worth of the core was maintained at a constant, elevated level for the remainder of the calculation. At the beginning of the transient, reactor temperatures increase rapidly in response to the reactivity insertion. As shown in Figures 5.58 and 5.59, which contain plots of the hot pin peak fuel and cladding temperatures during three instantaneous TOP calculations (instantaneous insertions of 0.17 \$, 0.45 \$ and 0.80 \$), reactor temperatures soon begin to level off as prompt reactor feedback effects (reactor fuel, sodium and steel expansion feedbacks) [5.33] respond to the temperature rise. Figure 5.59 shows that for instantaneous reactivity insertions of more than 0.17 \$, hot pin cladding temperatures would surpass the eutectic temperature limit. As with the slow reactivity insertion calculations, core damage levels are reached due to the onset of fuel melting instead of coolant boiling. Fuel melting occurs before cladding failure or the onset of boiling because the energy generated by the reactivity change is deposited directly into the fuel. Figure 5.60, which contains plots of the time to reach core damage conditions as a function of the amount of inserted reactivity, shows that for large reactivity insertions fuel melting would occur within fractions of a second. The figure also shows that fuel melting occurs before cladding failure. The reactor shutdown system will respond quickly, via the neutron power level trip, to a rapid reactivity insertion. But because the control rods require about 0.25 s to drop from the core upon receiving a trip signal [5.51], fuel melting and subsequent fuel pin failure would not be avoided during a large instantaneous reactivity insertion accident, regardless of the action of the reactor shutdown system. Reactor conditions for instantaneous reactivity insertion accidents as functions of the amount of inserted reactivity are listed in Table 5.27. This table shows that for instantaneous reactivity insertions worth less than about 0.3 \$, core damage conditions would not be reached because the peak fuel temperature would not surpass the fuel solidus temperature.

Fig. 5.58. Hot Pin Cladding Temperature during an Instantaneous Reactivity Insertion

Fig. 5.59. Hot Pin Peak Fuel Temperature during an Instantaneous Reactivity Insertion

Fig. 5.60. Time to Reach Core Damage Conditions during Instantaneous TOP Accidents

Table 5.27. Reactor Conditions during Instantaneous Reactivity Insertion Accidents

Inserted Reactivity (\$)	Hot Pin Peak Fuel Temperature (°C)	Hot Pin Peak Cladding Temperature (°C)	Hot Pin Peak Coolant Temperature (°C)	Hot Pin Fuel Melting Time (s)	Average Pin Fuel Melting Time (s)	Core Damage State
0.17	1084	709	663	--	--	No Core Damage/Unlimited Operation
0.20	1124	732	683	--	--	No Core Damage/Limited Operation
0.25	1189	774	719	--	--	No Core Damage/Limited Operation
0.30	1250	812	752	38.7	--	Minor Core Damage
0.40	1367	889	817	10.01	--	Minor Core Damage
0.50	1486	967	883	3.64	--	Minor Core Damage
0.55	1549	1005	916	2.23	--	Minor Core Damage
0.60	1613	1045	949	1.23	29.9	Core Damage
0.70	1746	1123	1015	0.508	14.4	Core Damage
0.80	1884	1202	1080	0.184	5.42	Core Damage
0.85	1953	1241	1113	0.159	3.74	Core Damage
1.00	2167	1359	1210	0.108	1.07	Core Damage

5.4.2.5 Loss of Primary Heat Sink

For the unprotected loss of heat sink sequence the end state is well characterized. In fact this scenario was run as a plant experiment twice in early 1986 as part of the Inherent Safety Demonstration Tests, [5.10]. Fig. 5.61 shows the predicted and observed response for an instrumented subassembly in the second experiment, B302R. The response to the loss of heat sink is an inlet temperature rise which shuts the system down to a uniform elevated temperature. For the technical specifications limited hot-sub channel with an inlet temperature of 700°F the associated temperature rise is given by the PRD ($\leq 35^\circ\text{C}$) and the inlet temperature coefficient of reactivity ($\sim 0.7^\circ\text{C}/^\circ\text{C}$) leading to 50°C rise (90°F) and a final temperature of 790°F.

5.4.2.6 Loss-of-Sodium Events

For purposes of accident sequence quantification the Loss-of-Sodium Accidents (LOS) are divided into two classes, rapid and slow transients. During a rapid LOS, the primary system loses sodium at a rapid rate which results in pump inlet uncover soon after initiation of the transient. During a slow LOS, the primary system loses sodium at a slower rate, which results in pump inlet uncover that occurs a long time after the initiation of the transient. The slow LOS calculations consisted of determining the maximum rate of coolant loss that can be withstood without causing core damage levels to be reached anytime during the transient.

The loss-of-circulation transient calculations are based on the following sequence of events. The reactor, initially at full power and flow conditions, experiences a failure which causes the primary system to lose coolant. A scram signal is generated at the same time as or soon after initiation of the LOS and the reactor scram is assumed to be successful.

The first calculation performed was a simulation of a protected rapid LOS. The transient disrupts the normal flow circuit, limiting natural circulation flow paths to within the reactor vessel and through reactor vessel leakage paths. For simplicity, the speeds of both pumps were

Fig. 5.61. SHRT B302R Top of Core Thermocouple and Reactor Inlet Temperatures

assumed to decay with coastdown characteristics of a M/G clutch failure, which provides a 24 s coastdown period. At the end of the coastdown phase, the pump inlets are assumed to become completely uncovered and sodium flow through the pump is disabled. In the model this is accomplished by giving the pumps an essentially infinite flow resistance. At the same time that flow through the primary pumps is disabled, coolant flow through the Z-pipe is stopped because, as the sodium level drops below the upper section of the Z-pipe, causing the upper section of the Z-pipe to drain in the absence of forced coolant flow. This also causes any heat removal through the IHX to cease.

Following the pump inlet uncover, flow through the core region subassemblies, which are connected to the high pressure plenum, settles to a nearly constant rate that is slightly under 1 % of the nominal core flowrate. The core region flowrate is balanced by a reverse direction flow through the blanket region subassemblies, which are connected to the low pressure plenum. A nearly steady flow circuit is established wherein natural circulation drives coolant flow through the core region subassemblies. The coolant flow then reverses direction and travels downward past the cooler blanket region subassemblies. It then travels through the low pressure plenum and out through the low pressure piping, until it drains into the high pressure piping beneath the pump outlets, through which the flow returns to the high pressure plenum. This flow circuit is augmented by leakage flows which allow cooler sodium from the pool to enter the reactor vessel. Sodium enters the high pressure plenum through leakages in the subassembly adapters. This flow returns to the primary pool via leakages at the reactor cover, after traveling through the core region. The leakage flow accounts for roughly 20% of the total natural circulation flow through the core region following pump inlet uncover. Although the bulk flow through the core region is relatively constant following the pump inlet uncover, flow in individual channels oscillate, as shown in Figure 5.62. This figure shows the normalized coolant flowrate and coolant outlet temperature of the hot pin channel during a protected, rapid LOS. As the flowrate in the channel stagnates following pump inlet uncover, the coolant temperature rises, which results in a larger natural circulation driving force and causes the flowrate to rebound. The rebounding flowrate cools the channel, lowering the temperature and natural circulation driving force. This process repeats resulting in oscillatory behavior within the

Fig. 5.62. Hot Pin Coolant Temperature and Flowrate for a Protected Rapid LOS

channel. Driver fuel subassemblies which have a relatively low flowrate at the start of the transient exhibit a more pronounced oscillatory behavior, because they are more likely to experience flow stagnation. The temperature oscillations during a rapid LOS are great enough to produce boiling and pin failure in core channels, thus the rapid LOS is a direct core damage event, (plant transient category P1), even if protected.

For the delayed LOS calculations the beginning of the pump speed coastdown and the pump inlet uncover time were delayed with respect to the initiation of the LOS and the scram signal. Core damage can be avoided if the pump inlet uncover is delayed long enough after the reactor shutdown. If, during delayed LOS, the pump inlet uncover is delayed 1 hour after initiation of the transient, the core will not be damaged because the peak coolant temperature (918°C) is less than the sodium boiling temperature (927°C). If the reactor has not been shutdown when the pump inlet uncover occurs, reactor damage is unavoidable.

The definition of the delayed loss-of-sodium event is based on the assumption that the reactor will be successfully scrammed. Because the sodium leak may not be detected immediately, the reactor shutdown may be delayed from the initiation of the sodium leak. The period of time between accident initiation and reactor shutdown must be added to the 1 hour delay period between reactor shutdown and pump inlet uncover to determine the time period that the sodium level must remain above the pump inlets. In other words, the sodium leak must be small enough to avoid pump inlet uncover during the period needed to detect the leak, shutdown the reactor, and allow 1 hour of forced circulation cooling of reactor decay heat power.

5.5 Uncertainty Analysis For Reactivity Feedbacks

5.5.1 Introduction

The reactivity feedbacks and their tendency to reestablish a stable state when the reactor is subjected to an accident initiator without scram were described earlier. The likelihood that this characteristic of EBR-II actually prevents core damage is the subject of this failure analysis. It therefore quantifies the conditional probability that reactivity feedbacks are not strong enough to keep temperature rises within safe bounds, given an accident initiator and failure to scram.

In essence the analysis consists of evaluating the ability to predict what the reactor behavior will be for each accident sequence. Computational predictions are used and all the relevant experimental evidence is applied in the evaluation.

The main reason that calculated transient outcomes are uncertain is that the input data are uncertain. The uncertainty in each input parameter can be expressed by constructing a distribution of possible values called a probability density function (pdf). The pdf for the transient outcome quantity of interest, e.g. peak temperature, can be found by sampling from the input parameter pdfs and running transient calculations with these parameter values. The fraction of the peak temperature pdf that lies beyond a safe temperature limit is the passive safety failure probability. This is basically the method described by Mueller and Wade [5.52].

This uncertainty propagation process would be prohibitively expensive to carry out without taking measures to make it computationally efficient. There are several accident initiators to consider, many uncertain parameters on which a transient calculation depends, and many transient calculations (each with a different, randomly selected parameter value) required to propagate each parameter's uncertainty accurately.

Two techniques were used to make the process efficient. One is screening. The procedure is to sort through all possibilities, discarding those that are unimportant. The other

efficiency measure is to use the response surface method [5.53]. The idea is to find a simple function, which is termed a response surface, that approximates the analytically unknown functional relationship between the outcome quantity and the input parameters. The parameter uncertainties can be propagated through this function inexpensively by Monte Carlo sampling, once the function is known.

The analysis method consists of the eight steps listed in Table 5.28. Several steps involve screening or the response surface approximation. EBR-II plant experiments play an important role in Steps 1 and 2, as well as Step 7. Steps are included that proved to be useful in other recent efforts to evaluate uncertainties in the outcome of reactor transients using probabilistic methods [5.54, 5.55].

Table 5.28. Steps in Passive Safety Analysis Procedure

1.	Screen Accident Initiators
2.	Calibrate EBR-II Model in SASSYS Transient Code
3.	Screen Parameters
4.	Quantify Parameter Uncertainties
5.	Develop Experimental Constraints
6.	Compute Response Surfaces
7.	Propagate Parameter Uncertainties Subject to Experimental Constraints
8.	Assess Accuracy of Failure Probabilities

The presentation of the method and results in the next section is organized by step. Some of the descriptions are lengthy because the approach has not been described previously. The casual reader may be satisfied with the summary and discussion provided in Section 5.5.3.

5.5.2 Analysis Method and Results

Step 1 - Screen Initiators

A screening analysis was done for each accident initiator. For initiators where this gives an unambiguous result, no further analysis is needed. The remaining accidents require a detailed analysis. The initiator screening was done principally using a quasi-static reactivity balance [5.56]. EBR-II plant experiments and past safety analyses supplemented the quasi-static evaluation.

The quasi-static reactivity balance can be written as

$$0 = \Delta\rho = A \cdot (P - 1) + B \cdot (P/F - 1) + C \cdot \delta T_{in} + \Delta\rho_{ext} \quad (5.1)$$

Here P is the power (normalized to unity at full power), P/F is the power-to-flow ratio (normalized to unity at normal power and flow), δT_{in} is the deviation from the normal inlet temperature and $\Delta\rho_{ext}$ is the change in externally-imposed reactivity. The coefficients A , B and C reflect the reactivity feedbacks. An accident initiator influences the reactivity balance by changing F , T_{in} and/or $\Delta\rho_{ext}$. Given the changes, Eq. 5.1 can be solved for the asymptotic power level that results when the feedbacks restore a reactivity balance. If the transient is one where worst conditions are reached asymptotically, then comparing the temperatures implied by the asymptotic power level against safe temperature limits tells whether there is damage in the transient. The temperature limits used for screening purposes were the eutectic temperature to represent operational limitations and the sodium boiling temperature to represent radiological release potential.

Experiment-based data were used in the screening to the extent possible. The sum of A and B is the power reactivity decrement from zero to full power (PRD), which is measured in every reactor run. The average from numerous runs, $-32.06 \pm 2.81\%$ was used. The average of four calculated values of the ratio A/B , 0.204 ± 0.048 [5.57], was used to separate A and B where needed. The average of experimental values for C , $-0.693 \pm 0.062\%/^{\circ}\text{C}$, was used.

Measured full power inlet and outlet temperatures provided initial conditions. The difference between inlet and outlet temperatures at any time, when multiplied by a peaking factor, gives the peak temperature rise at that time. Steady state peaking factors, 1.85 for coolant and 2.24 for cladding, were assumed to apply at all times.

For each ATWS event, using the average values of A, B and C above yielded a best estimate of the peak temperature, which implied either a failure or a no-failure prediction. To evaluate the reliability of these predictions, A, B and C were treated as normally distributed probability density functions and the uncertainty quoted above was taken to be the standard deviation of the distribution. Propagating these distribution functions through Eq. 5-1 yielded a distribution of peak temperatures for a given ATWS event. The fraction of this distribution lying above the safe temperature limit gave the failure probability for that event.

The screening results for the ATWS events are as follows:

- Loss Of Heat Sink - The failure probability is totally negligible, even by the eutectic formation criterion. This quasi-static result is consistent with the actual demonstration of the EBR-II plant response to a loss-of-heat-sink initiator [5.58], which had a completely benign outcome.
- Pump Overspeed - The failure probability again is totally negligible. The asymptotic P/F does not exceed unity so the temperatures do not increase. The pump run-up tests in EBR-II confirm this prediction [5.59].
- Loss Of Flow - A detailed analysis is needed. The quasi-static analysis shows that the asymptotic temperatures are benign with a very high probability. However, the peak temperatures in a loss-of-flow accident occurs during a transient peak, which is not addressed in the quasi static analysis. Past safety analyses suggest that the failure probability should be low for long pump coastdowns and high with short coastdowns. The SHRT plant experiments demonstrated that there is no damage if the coastdown time is sufficiently long [5.58].

- Loss Of Normal Power - A detailed analysis is needed. The accident is a long coastdown loss of flow combined with a loss of heat sink. The latter phase is benign and the former should have a low failure probability. The SHRT 45 experiment simulated this accident and there was no damage [5.58].
- Transient Overpower - A detailed analysis is warranted. The probability of eutectic formation appears to be high and fuel melting could occur. This assumes the initiator is a rod run-in with a $\Delta\rho_{\text{ext}}$ that is based on probability tables derived from operating history.

All the remaining steps of the analysis have been carried out for LONP but Steps 6 and 7 need to be completed for loss-of-flow and transient overpower initiators.

Step 2 - Calibrate Models

The model constructed for the consequence analysis was adjusted to yield good agreement with key EBR-II passive safety demonstration experiments [5.58]. For this analysis the model was reevaluated based on data in the literature and additional experiments.

Three modest changes were made. One change was to the nominal values of several feedback reactivity coefficients. New values were chosen based on an evaluation of values reported in the literature [5.60, 5.61, 5.62]. The second change was to increase the metal fuel conductivities to best estimate values [5.63] from three-sigma-low (conservative) values; this had little effect on loss of flow and transient overpower predictions but significant impact on power reactivity decrement (PRD) and rod drop feedback reactivity predictions, which are used in the experimental constraint process.

The third change was to the bowing reactivity feedback model. This is an important feedback because it is the only one that can be positive and it can be more than 40% of net feedback (although it is typically $\sim 15\%$). EBR-II's design is such that bowing is difficult to calculate from first principles and an empirical approach is usually taken. Typically the bowing

feedback for a specific reactor run is inferred from the measurement of PRD vs. reactor power; it is assumed that the calculated nonbowing PRD is correct and any difference between it and the measured PRD is attributed to bowing. The traditional model treats bowing feedback vs. power as zero until some threshold power and then it grows at a constant, generally positive slope [5.63].

Physical reasoning [5.64] and evidence from fifty PRD measurements suggest a somewhat different behavior: the bowing feedback slope is initially negative, changes to a generally positive value around 20 MWt and maintains that value until the vicinity of full power (62.5 MWt). In this vicinity the slope often starts to decrease and eventually it must become negative again. The new model incorporates the first two features but continues the second slope indefinitely beyond full power. This is conservatism prompted by a lack of unambiguous evidence for the location of the second slope change. Nominal values for the new bowing model input parameters were derived from the difference the nominal calculated nonbowing PRD vs. power and the average of the 50 PRD measurements.

The modified EBR-II model in SASSYS yields improved agreement with PRD, rod drop and passive safety demonstration experiments. Results for the SHRT 45 demonstration experiment [5.58] illustrate this. This experiment is directly relevant to unprotected LONP and other loss of flow accidents since it was a planned LONP event without scram. For this experiment, the prediction of peak feedback reactivity by the consequence analysis model is low by 10% and the model improvements reduced this error to less than 3%. There have been no safety demonstration experiments explicitly simulating the other class of accidents requiring a detailed analysis, transient overpower events. However, PRD and rod drop experiments serve to validate the model indirectly since they involve the same feedbacks that control the behavior of the core in other transients.

Step 3 - Screen Parameters

The number of transient calculations necessary for the analysis grows rapidly with the number of uncertain parameters. This provides a strong incentive to minimize the number of parameters that are treated as uncertain.

There are three classes of candidate parameters. One is modeling parameters, such as the number of channels and time step size. Selected values are such that parameters in this class are relatively unimportant. Modeling assumptions also should be considered here. None was identified as being likely to cause a substantially optimistic failure probability estimate. The second class is parameters that affect the predicted material expansions - such things as thermal conductivity and expansion coefficients. The third class is reactivity coefficients of expansion and bowing.

The first measure taken to reduce the number of uncertain parameters was to treat only the third class as uncertain and increase the uncertainty estimates to cover uncertainties that properly belong to the second class. The rationale for this approach is that 1) the feedback effect is the product of the calculated expansion and the reactivity coefficient of expansion, and 2) the reactivity coefficients often are the more uncertain factor.

The reactivity coefficients were screened using importance ranking. The importance of a parameter is the product of the parameter's uncertainty and the sensitivity of the transient outcome to a given change in the parameter. Screening uncertainty estimates for the feedback components were obtained from Ref. 5.52. Sensitivities were deducted from edits of component contributions to feedback reactivity histories in SASSYS calculations of each transient type. Of the ten feedbacks in the SASSYS EBR-II model, four of these were found to be unimportant to all transients relevant to the detailed failure analysis. The feedbacks that are important to a transient or constraining quantity are identified by an "x" in Table 5.29.

Table 5.29. Important Feedbacks

	<u>Core Na</u>	<u>Fuel</u>	<u>Up. Refl.</u>	<u>Rad. Refl.</u>	<u>Control</u>	<u>Bowing</u>
LOF	x		x	x	x	x
TOP	x	x	x	x	x	x
PRD	x	x	x	x	x	x
RDF(1.5)	x	x	x			
RDF(8.0)	x	x	x		x	x

Step 4 - Quantify Parameter Uncertainties

A pdf must be specified for each important parameter. There are not enough data to trace out a distribution for any of the parameters. Rather, a physically reasonable statistical distribution form was assumed and the available evidence was used to determine the distribution parameters (e.g., mean and variance).

A normal distribution was selected for all of the parameters except fuel expansion. The parameters for all the feedbacks except bowing physically cannot change sign. This creates an asymmetry or skewness in the true distribution, which a normal distribution cannot reproduce. However, except for fuel expansion, neglect of this effect introduces only a minor conservatism because this physical limit is more than four standard deviations from the mean. The mean of the fuel expansion parameter is too close to zero (3.0 standard deviations) for a normal distribution to be a reasonable choice. Consequently a log-normal distribution, which reflects this limit, was used for the fuel expansion parameter. Specification of the mean and standard deviation for each parameter is sufficient to complete the uncertainty quantification. Mean values are those developed in the evaluation of feedback parameters in Step 2.

The standard deviation estimates for the feedback components are based on an evaluation of the following sources: uncertainty estimates in the literature [5.61, 5.52, 5.65], the range of feedback values in the literature [5.60, 5.61, 5.62] and evidence from critical experiments [5.66, 5.67]. Reference 5.61 values are smaller than the others, apparently because of an attempt to

be consistent with the range of EBR-II PRD values. The spread in reported EBR-II feedback values primarily reflects the effect of core loading variations and some evolution in analysis methods, which make up only part of the total uncertainty. The C/Es (ratios of calculated to experimental values) are from a number of LMR mockups, none specifically of EBR-II. C/Es from [5.66] are based on more sophisticated neutronics analyses than those used to obtain EBR-II feedbacks and they do not include thermo-mechanical uncertainties. Table 5.30 shows the standard deviation values used here as well as some related quantities from the literature.

Table 5.30. Feedback Uncertainties and Related Quantities

<u>Feedback</u>	<u>Standard Deviation</u>			<u>C/E</u>	
	<u>this work</u>	<u>Ref. [5.61]</u>	<u>Ref. [5.52]</u>	<u>Ref. [5.65]</u>	<u>Ref. [5.66]</u>
Core Na	20%	6%	20%	30%	10%
Fuel	33%	11%	30%	47%→58%	1%→16%
Up. Refl.	30%	7%	20%	30%	
Rad. Refl.	25%		20%	30%	
Control	23%	5%	20%		10%
Bowing	70%	25%	15%→50%		1%→42%

Only a limited accounting for correlations among the uncertain parameters was made. The strongest correlation is among core Na, upper reflector and radial reflector feedbacks because they all are sodium density effects. This correlation was treated by expressing each of these as a product of two factors, one of which is common to all three feedbacks. The common factor was estimated to be uncertain by 18%, implying an uncertainty in the other factor of each feedback of 9%, 24% and 17%, respectively. Other correlations are less clear and were neglected.

Step 5 - Develop Experimental Constraints

Since the uncertain feedback parameters cannot be measured in EBR-II individually, the uncertainty distributions developed in Step 4 are not based on much experimental data specific to EBR-II. The net reactivity from all the feedbacks is measurable, however, and in fact has been monitored routinely over the life of the reactor. These data can be used to force the parameter uncertainty propagation to be consistent with the observed feedback characteristics. In this step the data are used to construct a pdf or a range for the net feedback under the various experiment conditions. These are used as constraints in Step 7.

The experimental constraints should allow the full distribution of feedback reactivities that EBR-II can exhibit but exclude feedback values that are unrealistic. Two types of experiments that reveal feedback properties have been performed often enough over the years to meet these criteria.

One is the PRD measurement, which yields the total feedback reactivity from hot zero power to full power. The PRD includes both prompt and slow feedbacks and has been measured at every subrun (a Technical Specification requirement). Fifty measurements made over the last ten years were selected to construct a probability density function (pdf) for the full-power PRD. The selection spans a wider range of core loadings than is anticipated for the future and includes the most extreme PRD values ever observed. The pdf is displayed in Table 5.31.

The other type of experiment used is the rod drop measurement of prompt feedback reactivity. This activity grows rapidly with time for a little more than a second, then the slope decreases and eventually an asymptotic value is approached. The feedback reactivities at 1.5 s and 8.0 s divided by the magnitude of the rod drop reactivity ($RDF(1.5)$ and $RDF(8.0)$, respectively) were selected as constraining quantities. These two quantities and the PRD each emphasize feedbacks with different time constants.

Table 5.31. Discrete Probability Density Function For The Experimental Full-Power PRD^a

<u>Reactivity Interval (¢)</u>	<u>Probability</u>
-46.8 to -42.3	0.06
-42.3 to -37.8	0.02
-37.8 to -33.3	0.30
-33.3 to -28.8	0.56
-28.8 to -24.2	0.06

^a The mean and standard deviation are 33.3 ± 3.9 ¢

RDF(1.5) and RDF(8.0) values were available from twelve rod drop measurements. (Much more data are archived, but in a form that is very difficult to use.) These data are predominantly from the last three years but go back to 1978. There are not enough data for constructing pdfs but they are sufficient to set limits on acceptable values of RDF(1.5) and RDF(8.0). From the twelve values, a mean and standard deviation were derived for each quantity. The acceptable limits were set at plus and minus three standard deviations from the mean. Some of these quantities are displayed in Table 5.32.

Table 5.32. Rod Drop Feedback Quantities

	<u>Lower Limit</u>	<u>Lowest Observed</u>	<u>Mean</u>	<u>Highest Observed</u>	<u>Upper Limit</u>
RDF (1.5)	0.146	0.165	0.193	0.217	0.240
RDF (8.0)	0.248	0.290	0.316	0.373	0.384

Step 6 - Compute Response Surfaces

A response surface is generated by calculated transient responses for various combinations of input parameters and then fitting the results to a lower order, multivariate polynomial. This was done with the aid of the PROSA-2 code [5.68], which offers several parameter selection and fitting schemes. The single-quadrant scheme was found to be the most computationally efficient [5.69] and was used here. Parameter sets, called knot points, were selected by PROSA-2 for each output quantity. Next SASSYS transient calculations were run using each knot point. Finally the resulting output quantities corresponding to each knot point were supplied to PROSA-2, which used them to compute the response surfaces.

Response surfaces were computed for the LONP-4 sequence and for the PRD and rod drop transients used to constrain the sampling in Step 7. The computational requirements are illustrated by the PRD case. It has 6 uncertain feedbacks with 8 corresponding uncertain input parameters. The single-quadrant scheme identifies 45 knot points and requires 45 SASSYS calculations of the PRD to generate the 8-parameter PRD response surface.

The output quantities must behave smoothly enough to be described well by a quadratic for this approach to be reasonable. The PRD and RDF quantities easily satisfy this restriction. However, it did preclude using clad failure life fraction as an output quantity to monitor failure. Sodium, cladding or fuel temperatures, depending on the accident and damage category, did prove to be good choices of output quantities.

Three damage categories are defined in this PRA. The mildest category is Potential Experiment Damage (PED). This is taken to mean operation in the "Unacceptable Region" (Region 1) defined in Section 2.2 of the Technical Specifications. Cladding temperature exceeding 815°C anywhere in the core at any time was used to signal this "damage" state. This category implies a Technical Specification violation but actual damage is not necessarily implied. Next in severity is Minor Core Damage (MCD), which is indicated by fuel melting, cladding failure or sodium boiling in the hottest, highest-burnup driver subassemblies. Core Damage

(CD), the most severe category, is indicated by fuel melting, cladding failure or boiling in an average driver assembly.

Response surface output quantities were selected for LONP-4 by examining the SASSYS model of EBR-II from the perspective of each damage category definition. The SASSYS code uses the concept of channels to represent different parts of the core. (A channel consists of a fuel pin, clad, coolant annulus and associated structure, and has characteristic physical properties.) The channels representing the hottest driver subassemblies are Channels 1 and 2, which are the same except for their burnups of 15% and 8%, respectively. Accordingly, the peak clad temperature at the top of either of these channels served as the output quantity for PED. Examination of the results of the 28 SASSYS calculations for LONP-4 showed that boiling (925°C) occurs long before cladding failure in Channel 2 and just after cladding failure in Channel 1. (Fuel melting does not occur in this accident.) A sodium temperature of 917°C at the top of Channel 1 (or 2) was determined to be a MCD threshold. Accordingly, the peak sodium temperature over the transient at the top of Channel 1 was used as the MCD output quantity. Average driver subassemblies are represented by Channels 6 and 8. Boiling occurs long before cladding failure at average burnups, and Channel 6 is always hotter than Channel 8. Accordingly, the output quantity used to monitor for CD was the peak sodium temperature at the top of Channel 6, with the boiling point (925°C) being the CD threshold.

Step 7 - Propagate Uncertainties

Once the response surfaces were known, computing transient outcomes corresponding to any set of input parameters (knot point) was simply a matter of evaluating the polynomial with that set of parameters. The input parameter pdfs were sampled about one million times and translated into transient outcomes. This knot point sampling was done randomly, i.e. by Monte Carlo, and yielded the distribution of possible transient outcomes. The fraction of an outcome distribution lying beyond the damage threshold is the failure probability.

The experimental constraints developed in Step 5 were imposed in this sampling process. A rejection technique was added to PROSA-2 such that the knot points retained to determine the

transient outcome distributions reproduced the experimental PRD pdf and satisfied the limits on RDF (1.5) and RDF(8.0) values.

The parameter uncertainties were propagated both with and without the experimental constraints. The resulting conditional damage probabilities from LONP-4 are shown in Table 5.33 and the outcome distributions are shown in Figures 5.63 through 5.65. The abscissa label "Temperature" refers to a different output quantity in each of these figures, as explained under Step 6. The nominal SASSYS calculation of this accident indicated operation in the acceptable region and peak sodium temperature below, but close to boiling. Thus it is not surprising that the passive safety failure probability for PED is essentially unity and for MCD it is as large as 0.14 without experimental constraints. The imposition of the experimental constraints reduced the MCD failure probability by more than an order of magnitude. The RDF constraints added very little to the PRD effect for this accident, perhaps because of the long time scale of the accident.

Table 5.33. Passive Safety Failure Probabilities for LONP-4 Accidents^a

	PED ^b	MCD	<u>CD</u>
No Experimental Constraints	0.98	0.143 ±0.2%	2.5x10 ⁻⁶
Only PRD Constraints	0.99	1.3x10 ⁻²	<2.5x10 ⁻⁶
PRD and RDF Constraints	1.00	1.2x10 ⁻²	<2.5x10 ⁻⁶

^a The uncertainties shown reflect only Monte Carlo statistics.

^b The PED statistical uncertainties are less than 0.1%

Fig. 5.63. LONP-4 Outcome Distribution For Potential Experiment Damage

Fig. 5.64. LONP-4 Outcome Distribution for Minor Core Damage

Fig. 5.65. LONP-4 Outcome Distribution for Core Damage

Very few samples yielded CD outcomes when there were no experimental constraints and none did with the constraints. Consequently, the non-constraint estimate of 2×10^{-6} has a large statistical uncertainty. With none of the million samples exceeding the damage threshold, the explicit observation is simply that the constrained core damage probability is unlikely to be large compared to 10^{-6} . Yet it is apparent from the way the temperature distributions are falling off in Fig. 5.65 that the fully constrained damage probability must be many orders of magnitude smaller than the unconstrained estimate of 2×10^{-6} . The much smaller bounds for the constrained core damage probabilities shown in Table 5.33 were obtained from fits to the Monte Carlo data. The fitting scheme is discussed in [5.70]. Several statistical distributions functions were fit to the data and the damage probability was computed with these functions. This amounts to an extrapolation of the sampling data but, on physical grounds, that is reasonable approach in this case. The goal was simply to determine whether the probability is below the cutoff for consideration in the EBR-II PRA. All estimates are so many orders of magnitude below the 10^{-10} cutoff that core damage can be dismissed.

Step 8 - Assess Accuracy

Sensitivity calculations are done to assess the reliability of the failure probability predictions. These calculations evaluate the error due to parameter screening, the sensitivity to input parameter pdfs and the accuracy of the response surfaces. This analysis is discussed in more detail in [5.71].

A conservative upper bound was found for the error due to screening out parameters from the uncertainty propagation (Step 3). This was done by recomputing the failure probabilities with only the two most important parameters treated as uncertain and comparing these with the reference results (6 certain parameters). The bound for each damage category is 1% for PED, 19% for MCD and one order of magnitude for CD. (It is generally true that errors and uncertainties become a larger percentage of the failure probability as the failure probability gets smaller.) The actual error from the parameter screening is small compared to these bounds, since the four parameters dropped for this test are much more important than the ones screened out in Step 3.

The sensitivity to assumptions about the parameter pdfs was computed with the aim of finding plausible upper bounds on failure probabilities. Assumptions as pessimistic as experimental evidence and common sense would allow were made. First, different statistical distribution functions (log-normal and beta) were tried, without changing the mean and standard deviation. The distributions were skewed to the pessimistic side by postulating a bound (for which there is no evidence) as close as three standard deviations from the mean.

Next, three cases were run where the mean or standard deviation was allowed to change. In one, the standard deviation of the most important parameter was increased by 30% of its original value. In another, all standard deviations were increased by 25% of their original values. Finally, all means were moved one standard deviation in the pessimistic direction. The last case was exposed by the experimental constraining process as being hopelessly in conflict with the PRD and rod drop data.

The following conclusions were drawn about the parameter pdf sensitivities. The PED probability is essentially unity no matter what plausible assumption is made. The uncertainty for the experimentally constrained MCD probability is less than a factor of 2. No CD failures were found in the experimentally constrained Monte Carlo sampling and fits to the data yield failure probabilities below the 10^{-11} cutoff [5.70].

The accuracy of the single-quadrant response surface approximation was investigated by comparing it to more accurate approximations on a simplified version of the LONP problem. The simplification was to treat only the two most important parameters as uncertain, as in the test of parameter screening. This made it feasible to trace out the true functional relationship with many more knot points and use higher order and more flexible polynomials.

The response surface test results are similar to those for parameter pdfs; the PED probability is essentially invariant, the MCD probability could increase by no more than a factor of two, and the experimentally constrained CD probability remains below the 10^{-11} cutoff.

5.5.3 Summary and Discussion

Reactivity feedbacks are a major element of passive safety, bringing heat production from fission into balance with heat removal. The EBR-II feedbacks are well understood and, with one exception, they are relatively easy to predict. There is no close competition between opposing feedbacks and even the sodium density effect, which is by far the strongest feedback, does not suffer from a close balance between opposing spectral and leakage effects. As a result, the best estimate feedback reactivity values and the uncertainty estimates should be reliable. A conservative model was used to deal with bowing, the one feedback that is quite difficult to predict accurately.

A screening analysis of ATWS events (Step 1) determined that reactivity feedbacks virtually guarantee a benign outcome in loss-of-heat-sink and pump-runup accidents. Loss-of-normal-power (LONP), loss-of-flow and transient overpower transients were found to warrant a detailed analysis of the effectiveness of reactivity feedbacks.

In the detailed analysis the probability that the reactivity feedbacks are not strong enough to prevent core damage was computed by propagating parameter uncertainties through transient calculations. Plant transient data were used to calibrate calculational models (Step 2). Parameters were screened using importance ranking to reduce the number of uncertain parameters (Step 3). Transient calculations were run with different parameter values to construct response surfaces, approximate analytic functions describing the dependence of transient outcomes on the input parameters (Step 6). Then transient outcome distributions were found by evaluating the response surface functions for about a million parameter sets selected randomly (Step 7). In this Monte Carlo sampling from theoretical parameter distributions, the values selected were constrained to be consistent with PRD and rod drop experiments. A fitting procedure was used to obtain bounds on probabilities that are small compared to 10^{-6} . Finally, an uncertainty analysis was done to evaluate the reliability of the failure probability estimates (Step 8).

This method was used to analyze the LONP accident without scram. Operation in the Technical Specification unacceptable operating region, where damage to aggressive experimental subassemblies is possible, is a virtual certainty. The best estimate of the minor core damage probability is $1.2 \cdot 10^{-2}$. The uncertainty in this estimate is a factor of 2. The best estimate core damage probability is below the 10^{-11} cutoff for dismissing sequences, and the probability remains below this cutoff when uncertainties in the estimate are included. All of these are conditional probabilities; occurrence of a LONP initiator and failure to scram are given.

The analysis is firmly grounded in experimentally verified reality. Full advantage was taken of the EBR-II experimental database accumulated over its long operating history. The inherent safety demonstration experiments, in particular, provide compelling evidence of passive safety for the events they simulated. And they provided benchmarks for validating the computational tools used to predict responses that were not simulated explicitly. Information from experiments that measured net feedback reactivity made it possible to reduce their minor core damage probability estimate by one order of magnitude and the core damage probability by more than five orders of magnitude.

The reliability of the safety margins from the passive response compares favorably with those from engineered safety features. Since reactivity feedbacks are simply a manifestation of the laws of physics, primarily thermal expansion, there is virtually no chance that the passive response will fail to occur. The only exception is if there is a catastrophic accident initiator that disrupts the core geometry (severe earthquake), where it is reasonable to assume that the damage probability is unity. For the same reason, passive safety has advantages as a complement to primary safety systems. Compared to something like a secondary scram system, there is little chance of some dependent failure defeating both passive safety and the primary safety systems it complements.

The results demonstrate that reactivity feedbacks provide a large safety margin. The conditional damage probabilities would be unity for all the initiators retained in the screening if there were no feedbacks. A large margin is also anticipated for LOF and transient overpower. This margin is in addition to the large safety margin afforded by the scram system.

The substantial safety benefit of EBR-II's reactivity feedback response is not typical of other reactors that have been the subject of PRAs. The vulnerability of LWR plants is such that core damage is generally assumed to occur if there is a failure to scram. Consequently, this is the first PRA to make a probabilistic assessment of reactivity feedbacks. Although this treatment is not a conventional system failure analysis, it is a well founded approach for producing a probabilistic failure quantification.

For application of the results of this analysis to future EBR-II cores, one anticipated reactor modification warrants some discussion. It is the substitution of plutonium for fissile uranium. The effect on reactivity feedbacks of such a substitution has been investigated for a few cores [5.56, 5.72, 5.73] including EBR-II [5.76]. The β_{eff} would decrease by about 25%, which would enhance the feedback reactivities in units of dollars by this amount if there were no other changes. Sodium density reactivity, which is by far the most important feedback in EBR-II, would get less negative in absolute units by a moderate amount - ~ 10%. (The sodium density feedback change is much less in EBR-II than in larger cores because of the large leakage component.) Fuel expansion feedback would be much larger with Mark-V (Pu) fuel and other significant feedbacks should change about 10% in absolute units. Asymptotic transient outcomes depend on reactivities in absolute units and there could be a slight degradation of the passive safety performance for most ATWS events. However, the changes are within the uncertainties used in the analysis. The peak temperature in LONP and LOF events depends on feedback in dollars, and therefore the passive safety performance in these transients should improve modestly with the fuel change. The overall conclusion is that the reactivity feedback reliability estimates presented in this section would be applicable to EBR-II loadings with plutonium driver fuel.

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6.0 SYSTEMS ANALYSIS

This section presents a summary description of the systems analysis methodology used to estimate the probability of the failure of EBR-II frontline and support systems which may lead to core damage states as described by the accident sequences in Section 5.0 of this report. Included are brief descriptions of the systems presented in Appendix C, the mean failure probabilities for each system, possible initiating event occurrences, and the results of sensitivity and uncertainty analyses.

6.1 Systems Analysis Procedures

In the early phases (1989) of this study, the set of systems requiring analysis were selected along the traditional division of reactor systems into (1) frontline systems (i.e., mitigating systems important to safety in terms of core damage) and (2) support systems (systems required to ensure the success of one or more of the frontline systems or support systems). A matrix showing the interfaces and dependencies among the frontline and support systems is shown in Table 6.1.

Following the selection of these systems, a methodology for analysis was created specific to the needs of the EBR-II PRA by utilizing the procedures recommended in a number of PRA reactor studies and references, particularly the USNRC PRA Procedures Guide [6.1] and the IDCOR Technical Report 86.3B1, Appendix B [6.2].

In addition, the collaboration of ANL staff and outside PRA consultants was required to ensure completeness of the technical effort.

The results of the systems analysis effort are documented in system notebooks (Appendix C) which contain the following information (as recommended in ref. 6.2):

- System function

Table 6.1. Front Line and Support Systems Interface/Dependency Matrix

<div> <div>Support Systems</div> <div>→</div> </div> <div> <div>Front Line Systems</div> <div>↓</div> </div>	Thimble Cooling/Stack Exhaust System	Secondary Sodium System	Balance-of-Plant (BOP) Systems	Electrical Systems	Instrument Plant Air System	Water Supply Systems	Argon System
Reactor Shutdown System (RSS)			X	X			
Primary Pumps	X			X			X
Primary Pumps Motor-Generator Sets				X		X	
Shutdown Cooling System				X		X	
Shield Cooling System	X			X	X	X	
<div>Support Systems</div> <div>↓</div>							
Thimble Cooling/Stack Exhaust System				X	X		
Secondary Sodium System			X	X		X	X
Balance-of-Plant (BOP) Systems		X		X	X	X	
Electrical Systems	X	X	X		X	X	X
Instrument and Plant Air System	X		X	X		X	
Water Supply Systems		X	X	X	X		
Argon System		X		X			

- System description
- System operation
- Performance during accident conditions
- Location within the plant
- Operating experience
- Logic models (fault trees)
- Initiating event review
- Quantification
- Summary of key findings
- Summary of review comments
- References

This summary section presents information from the system notebooks for the frontline and support systems with emphasis on the quantification results as pertains to the core damage frequency.

The systems analysis procedures included several phases:

- Plant familiarization, including several walkdowns
- System modeling and fault trees
- Integration (and critique) of the data base
- Quantification - failure frequency

Concomitant with these phases was the evolution of a set of guidelines and system descriptors particularly oriented to computer storage and documentation requirements as outlined in the Quality Assurance Plan [6.3].

Following the plant familiarization task, which included the accumulation of information from safety documents, current drawings, maintenance and operating procedures and extensive interaction with EBR-II management and operating personnel, the system modeling and fault tree development proceeded under the following rules:

- The system models reflect the plant configuration as of October, 1989.
- For reactor operational data, there are three (3) 100-day runs per year (capacity factor $\sim 82\%$).
- The truncation values for the fault tree models are a maximum of 10^{-11} failure probability and 7 terms of the minimal cutsets.
- System models include random failures, mandated operator actions, test and maintenance unavailability, and human error.
- System models do not include errors of commission or recovery actions, with some special exceptions.
- Events in the final fault trees are of two types: basic events and house events.
- The database (see Section 7.0) used for quantification is derived from generic data, the CREDO LMR database, and EBR-II plant-specific data.
- Failures due to intersystem dependencies, effects of initiating events and common-cause failures are modeled explicitly in the fault trees.

6.1.1 System Analysis Naming Conventions

The construction of fault trees requires naming conventions for identification of main systems, support systems, components, failure modes, component locations, gates (ANDs and ORs), basic events, and house events. Figure 6.1 shows the formats for gates and basic/house events. The sections presented here (6.1.1) are intended to demonstrate the depth and completeness of the systems analysis task. (For specific examples of the usage of naming conventions, refer to the system notebooks in Appendix C).

Gate Names:

X n n n n

Basic Events:

X c c d d d l m

House Events:

X O O n n s 0 0

(Note: See Appendix C for specific examples.)

Fig. 6.1. Gate and Basic/House Event Formats for Fault Tree Inputs

6.1.1.1 Gate Names

As shown in Figure 6.1, gate names are formatted as Xnnnn, where X is the system identifier and nnnn is a unique four-character (alphanumeric) set designated by the system analyst. Where feasible, nnnn is numeric and these numbers are in ascending order as the gates appear more deeply nested in the fault trees.

6.1.1.2 Basic Event Names

As shown in Figure 6.1, basic event names are formatted as Xccdddlm, where X is the system identifier, cc is the component type identifier, ddd is a component identifier specific to EBR-II plant terminology, l is the location code, and m is the failure mode for the basic event.

6.1.1.3 House Event Names

As shown in Figure 6.1, house events are formatted as XOOnns00, where X is the system identifier, OO is the component identifier for a conditional event (identical for all house events), nn is a two-digit (numeric) house event sequence identifier, s assumes a Z, A, or B value dependent upon pairing, and 00 are fixed numeric zeros.

6.1.1.4 Identifiers for Main and Support Systems

The main systems and the support systems are designated in the fault trees by a one-character alpha identifier as listed in Table 6.2. This identifier always appears as the first character of a gate name, a basic event name, or a house event name.

6.1.1.5 Identifiers for Mechanical and Electrical Components

The system component type is designated in the fault trees by a two-character alpha identifier as listed in Table 6.3. This identifier always appears as the second

Table 6.2. System Identifiers

1.	MAIN SYSTEMS	
	Auxiliary Pump	A
	Balance of Plant	B
	Condensate	Z
	Steam	V
	Control Rods	C
	Fuel and Experimental Pins and Subassemblies	F
	Primary Pumps	P
	Reactor Shutdown System	R
	Safety Rods	S
	Secondary Pump	N
	Shield Cooling	H
	Shutdown Coolers	D
2.	SUPPORT SYSTEMS	
	Argon	O
	Containment Isolation	K
	Cover Gas	G
	Cover Gas Cleaner	M
	DC Power	L
	Emergency Power	E
	Instrument Air	I
	Normal Power	J
	Plant Cooling Water	Q
	Reactor Cooling Water	W
	Primary Sodium and Siphon Break	X
	Thimble Cooling	T
	Uninterruptible Power Supply (UPS)	U
	Stack Exhaust System	Y

Table 6.3. Mechanical and Electrical Component Identifiers

<u>IDENTIFIER</u>	<u>MECHANICAL COMPONENT</u>	<u>IDENTIFIER</u>	<u>ELECTRICAL COMPONENT</u>
AU	Accumulator	AC	AC Power
CA	Cable	AM	Amplifier
CD	Safety/Control Rod Drive Unit	AN	Annunciator
DL	Diesel	BY	Battery Charger
FL	Filter or Strainer	BS	Bus
GB	Gas Bottle	CA	Cable
HE	Heat Exchanger	CB	Circuit Breaker
IG	Igniter	CL	Clutch
OR	Orifice	DC	DC Power Supply
PF	Flexible Hose	IV	Inverter (Solid State)
PP	Pipe	LT	Light
PE	Expansion Joint	LS	Limit Switch
PC	Pipe Cap	MO	Motor
PV	Pressure Vessel	ND	Neutron Detector
PT	Pump, Turbine-Driven	DE	Diode or Rectifier
PM	Pump, Motor-Driven	RC	Recorder
TK	Tank	LA	Light Arrester
CR	Cooler	GS	Ground Switch
TG	Tubing	RE	Relay
CV	Valve, Check	RT	Resistor, Temperature Device
XV	Valve, Manual	PS	Switch, Process
MV	Valve, Motor-Operated	VR	Voltage Regulator
RV	Valve, Relief	PI	Process Indicator
SV	Valve, Solenoid	GC	Governor Valve Controller
DV	Valve, Diaphragm	ST	Solid State Device
AV	Valve, Pneumatic		
HV	Valve, Hydraulic	HR	Heater
GV	Valve, Regulator	SW	Switch, Manual
VV	Valve System	KS	Switch, Lock-out
FE	Flow Element	AT	Automatic Transfer Switch
SD	Steam Drum	DP	Distribution Panel
MF	Motor Fan (Blower)	IN	Instrumentation
FN	Fan	CO	Connector
RD	Rupture Disc	FU	Fuse
CH	Chiller	GE	Generator
CM	Compressor	HT	Heat Tracing
CN	Condenser		
LC	Control Logic		
SL	Seals		
TV	Turbine		

Table 6.3. (Continued)

<u>IDENTIFIER</u>	<u>MECHANICAL COMPONENT</u>	<u>IDENTIFIER</u>	<u>ELECTRICAL COMPONENT</u>
BE	Bearing	TR	Transformer, Power
GG	GN Generator	XT	Transformer, Voltage
FM	Flow Meter	SC	Speed Controllèrs
BO	Boiler	WR	Wire
GE	Generator	CC	Capacitor Gear
DM	Damper	RS	Resistor
TP	Power Turbine	TZ	Position Transmitter
HU	Human Error	EX	Exciter (of Motor Pump Generator)
SZ	Position Sensor	SG	Switch Gear
TE	Temperature Element	OO	Conditional Event
ME	Meter	TC	Thermocouple
DC	DC Power	CZ	Electrical Contacts
LE	Level Sensor	BI	Bistable
PR	DC Power	VI	MVI Converter
LE	Level Sensor	BA	Battery
PR	Pressure Sensor	TS	Pressure Transmitter
TM	Tachometer		
DR	Dryer		
RR	Receiver		
LK	Linkage		
CP	Coupling		
IP	Impeller		
LB	Lubrication Subsystem		
AR	Actuator		
SR	Safety/Control Rod		
TW	Tower/Chimney		
HA	Primary Tank Hangers		
BD	Containment Building		
SB	Fuel Storage Basket		
DK	Primary Tank Deck		
NA	Not Available, Other		
OO	Conditional Event		

and third characters of a basic event name. The system component identifier is an alphanumeric abbreviation derived from EBR-II terminology as used on drawings, etc. This identifier always appears as the fourth, fifth, and sixth characters of a basic event name.

6.1.1.6 Component Location Identifiers

System component locations are designated in the fault trees by one-character alpha identifiers as listed in Table 6.4. This identifier always appears as the seventh character of a basic event name.

6.1.1.7 Failure Mode Identifiers

Failure modes for system components are designated in the fault trees by a one-character alpha identifier as listed in Table 6.5. This identifier always appears as the eighth character of a basic event name.

Table 6.4. Component Location Identifiers

<u>IDENTIFIER</u>	<u>BUILDING</u>	<u>LOCATION</u>
R	Reactor	Reactor/Reactor Top, Tank
A	Reactor	Batteries; Control Rod Instrument Air Tank
F	Reactor	Fuel Handling Machine Area; Argon Cell; CGCS Area
L	Reactor	RSLC; Electrical Equipment
S	Reactor	Subbasement Area
W	Boiler	Boiler Wing
N	Boiler	Sodium Storage Tank
E	Boiler	Electrical Equipment Room
I	Boiler	Electrical Equipment Room
J	Boiler	Operations Area
P	Power Plant	Power Plant Corridor
B	Power Plant	Battery Room
V	Power Plant	2400 V / 480 V Switch Gear
G	Power Plant	100 kW Diesel Generator
H	Power Plant	400 kW Diesel Generator, Boilers
Y	Power Plant	Rest of First Floor Power Plant Building
T	Power Plant/ Reactor	Cable Tunnel
D	Power Plant	DAS Room
Q	Power Plant	Cable Routing Room
X	Power Plant	DAS Expansion Room
K	Power Plant	Condenser, Pumps, Rest of Mezzanine Floor Power Plant
C	Power Plant	Control Room
Z	Power Plant	Turbine, 13.8 kV Switch Gear, Rest of Operating Floor Power Plant
M	Power Plant	Diesel Oil Day Tanks Area
1	Power Plant	Auxiliary Cooling Tower (Roof)
2	-	Main Cooling Tower
O	-	Outside Main Buildings (Shutdown Coolers, Piping, CGCS Building...)
U	-	Other

Table 6.5. Failure Mode Identifiers

<u>IDENTIFIER</u>	<u>FAILURE MODE (ACTIVE)</u>	<u>IDENTIFIER</u>	<u>FAILURE MODE (PASSIVE)</u>
K	Does Not Reclose	A	Short to Power
L	Latent Human Error	B	Open Circuit
M	Calibration	C	Short to Ground
N	Does Not Close	D	Shorts
O	Does Not Remain Closed	E	Plugged
P	Does Not Open	F	Leakage/Rupture
Q	Does Not Remain Open- Fails Closed	G	Distortion/Binding
R	Does Not Continue to Run	H	Leak
S	Does Not Start	I	Dirt/Foreign Material
T	Multi-Position Switch/Valve in Wrong Position	J	Vibration
U	Common Cause Failure		
V	Grip Failure		
W	Loss of Function		
X	Maintenance Unavailability		
Y	Test Unavailability		
Z	Dynamic Human Error		

6.2 Analysis of Frontline Systems

The frontline systems include the Reactor Shutdown System (RSS), the Primary Pumps, the Primary Pumps Motor-Generator Sets, the Shutdown Cooling System, and the Shield Cooling System.

6.2.1 Reactor Shutdown System (RSS)

This section presents the analysis used to determine the probability of scram failure of the RSS in the unlikely events of loss-of-flow (LOF) or transient-over-power (TOP). Either of the two events is assumed to occur at any reactor steady power level. The RSS and the fault trees for the LOF and TOP events are described in detail in Appendix C.1.

6.2.1.1 System Function and Description

The EBR-II Reactor Shutdown System (RSS) is designed to automatically shut down the reactor in the event of equipment or operator malfunctions or environmental problems that could damage the reactor, injure personnel, or release contamination to the environment. The shutdown system will cause the control rods (and in some cases, the safety rods) to automatically scram to their least reactive position to shutdown reactor operation, if any of the selected plant parameters exceed established limits.

In the event of an LOF or TOP, an automatic scram signal normally opens a sufficient number of contacts in shutdown strings A and B (Figure 6.2) to deenergize the corresponding Control Power relays (CP-relays) which in turn deenergize the clutches of the individual control rods and thus allow rapid insertion of all the control rods. Each control rod has its own release and scram mechanism and drops into the core by gravity assisted by a pneumatic force acting on a piston at the top of the rod. Scram can also be initiated manually by the reactor operators. The operators are instructed first to scram the control rods and if the shutdown can not be verified, the operators should then scram the safety rods. The two safety rods share the same

Fig. 6.2. Reactor Shutdown System (RSS) Simplified Schematic

scram mechanism and are laterally connected so that they can only move together vertically. The safety rods drop into the core by gravity assisted by a torsional spring force.

The EBR-II core can be brought subcritical from any authorized power level by fully inserting a single high worth control rod or the two safety rods. Therefore, a failure to scram the reactor would arise when all the high worth control rods (there are currently nine high worth rods in the core) fail to drop in the core and also the two safety rods fail to drop. Specifically, a failure to scram can take place due to one of three reasons: (i) failure to send a scram signal to the shutdown strings, (ii) failure to release the control rods and safety rods, and (iii) failure to insert the control rods and safety rods. All potential causes and mechanisms that would lead to the above scram failures have been delineated in the fault trees.

6.2.1.2 Support System Interfaces and Effects

The electrical systems (Appendix C.9) provide the only direct interface system for the RSS. In particular the Continuous Power Supply (CPS) provides power to the electrical buses RCL-1 and RCL-2 for both shutdown strings. There are a number of interfaces with the Balance-of-Plant (BOP) systems that provide reactor scram signals.

6.2.1.3 Fault Tree Development

The fault tree analyses have been developed based on the background material provided in the RSS notebook (Appendix C.1). Some of the highlights of these fault trees are presented in this subsection. Since a scram failure occurs only when all the high-worth control rods fail to drop, most of the independent mechanical failure modes of the control rods have been eliminated from the fault trees. Earlier analyses have shown that these independent failure modes have insignificant contribution to the overall scram failure probability. As system understanding developed, more emphasis was placed on the mechanical common cause failures of the control rods and common cause failures of similar electrical components in redundant channels and strings.

The mechanical failure of the control rods are represented in terms of five different failure modes which cover potential causes and locations of binding. Distinct failure modes, for instance, are assigned to foreign materials in sodium entering the assemblies and in sodium exiting the assemblies. Such a detailed description is useful for understanding the system and how its components interact with each other and also for developing future data base. As a result of this detailed representation, two failure modes are found to have potential common cause mechanisms between the control rods and safety rods. The two modes are: failure of the rods to move within their thimbles as a result of foreign material in sodium entering the assemblies and as a result of cross section distortion and bowing of the assemblies relative to their thimbles.

6.2.1.3.1 LOF Fault Tree

The LOF event considered in the fault tree analysis reduces the primary sodium flow below the specified set points of the low-flow channels. There are presently four low-flow channels with scram capability. The four channels are designated according to the location of the flow sensor as follows: HPPF-2 with a flow sensor at the high pressure plenum of pump No. 2, LPPF-2 with a sensor at the low pressure plenum of pump No. 2, TSOF with a sensor at the reactor coolant outlet pipe, and UPFB with a sensor at the reactor upper plenum. Channels HPPF-2 and LPPF-2 use magnetic type of flow sensors whereas channels TSOF and UPFB use pressure difference type of sensors in conjunction with NaK-pressure and strain-gauge signal transducers. In an LOF event that could be detected by all the four sensors, the flow channels in a 2-out-of-4 trip logic would initiate a scram signal by eliminating the voltage across shutdown strings A and B. Because of the locations of the flow sensors, not all types of LOF can be detected by all the four sensors. For example, shaft binding of primary pump No. 1 would be directly detected by TSOF and UPFB channels and indirectly by LPPF-2 channel. The HPPF-2 channel, because of its sensor location, can not detect such an LOF condition and thus the trip logic of the flow channels becomes 2-out-of-3. To accommodate this type of LOF events, two mutually-exclusive house events are used with the HPPF-2 channel to indicate at the outset whether this channel can detect the LOF condition under consideration.

In addition to the automatic scram, the reactor operators are trained and instructed to quickly initiate a manual scram when they realize that an LOF condition exists. The human action in this case will rely on annunciators, alarms and indicators that are independent of the low-flow channels and SOT channels. The operators continuously monitor the reactor delta-T along with the primary system flow. The reactor delta-T has two annunciators that should alert the operators in the event of an LOF. With regard to the primary flow condition, there is an annunciator for Pump No. 1 or No. 2 low rpm and an annunciator for Pump No. 1 low discharge pressure. In addition there are DAS alarms for reactor power, speed of the two primary pumps and array of bulk sodium temperatures and upper plenum probe temperatures.

6.2.1.3.2 TOP Fault Tree

For the purpose of assessing the reactor shutdown capability, the TOP can be described simply as an abnormal event that increases the neutron flux level above the specified set points of the wide-range nuclear channels. There are presently three independent wide-range nuclear channels, designated channels A, B, and C, with scram capability. In the event of a TOP, the wide-range nuclear channels in a 2-out-of-3 trip logic would initiate a scram by eliminating the voltage across shutdown strings A and B. Each of the wide-range channel also trips at loss of DC voltage to the nuclear detector and when one of the non-operate conditions exists in the channel.

In addition to the automatic scram capability, the reactor operators can initiate a manual scram when they realize that a TOP condition exists and that the automatic scram has failed. Reactor delta-T with its two annunciators should alert the operators of the abnormal core condition. Digital indicator of the compensated ion chambers (channel 7 or 7a) should confirm that a TOP has actually occurred. As in the case of LOF, the DAS provides an array of temperature alarms as well as alarms from channels 7 and 7a.

6.2.1.3.3 Contribution of SOT Channels

There are four independent SOT channels that provide a redundant and diversified automatic scram capability in the event of LOF, TOP, or any core overheating. The SOT channels scram has a 2-out-of-4 trip logic in both reactor shutdown strings. The thermocouples used in the SOT channels are located in the reactor vessel upper plenum above driver assemblies where they sense the temperature of sodium as it exits these assemblies. The modeling of the SOT channels covers the failure to initiate an automatic scram signal in terms of the failures of the electrical components and a common human error in setting the trip points of all the channels. This SOT branch is then combined with the failure to initiate a scram signal by the low-flow channels in the LOF fault tree and with the wide-range nuclear channels in the TOP fault tree. It is important to mention that there are a number of components that are similar in design and made by the same manufacturer that are used in the

SOT channels and the low-flow channels. Similar MV/I converters, trip bistables, and trip relays are used in both types of channels. The SOT channels and TOP wide-range nuclear channels, however, only have similar trip relays. Common cause failure between these similar components are considered in the fault trees.

6.2.1.4 Component Failure Data, Common Cause Failures, Human Reliability

The components failure data used in the fault tree analyses are given in Section 7 and the system notebook (Appendix C.1). The common cause failures of the electrical and mechanical components and human errors have been treated in detail. Generally, conservatism is applied when the failure data is selected either from generic data sources or derived from plant operating records. For instance, two failure modes are considered for each relay and its contacts, one failure per demand of $3.0 \cdot 10^{-4}$ and another failure per hour of operation equal to $2.0 \cdot 10^{-7}$. The latter failure mode is particularly important as it simulates the effect of extended run times. Also, component failure rate corresponding to all modes are used when failure rates for individual modes can not be determined. Generic data [6.4] and EBR-II CREDO [6.5] data are used in conjunction with more recently available EBR-II specific data.

In almost all parts of the EBR-II RSS, redundancy has been provided in order to reduce scram failure probability. As a result, common cause failures (CCFs) of similar components, rather than independent failures, are expected to dominate the probability of scram failure. There are few mechanisms, however, that can lead to CCFs among the similar and independent components in the RSS. For instance, the detection channels rely in their normal operation on AC voltage supplied from the UPS system. But a failure of the UPS would directly lead to an automatic scram as a result of deenergizing the magnetic clutches of all the control rods as well as the magnetic clutch of the two safety rods. Also, the RSS of EBR-II has been successfully operating for more than 25 years and any design or maintenance procedure defects are excluded as CCF mechanisms. One of the mechanisms of CCF that is postulated is a hardware failure among similar components made by the same manufacturer. This CCF can arise as a result of extended operations of components or simultaneous replacement of old components with new ones that are either defective or are not compatible with other components.

CCFs among similar components are considered in the fault tree analyses in terms of the beta-factor method [6.6] by two types of analyses. Analysis "A" utilizes relatively large and fixed beta-factors, since the emphasis is on common causes among components that share the same location and operate within the same channel or string. For instance, all the CP-relays on string A that affect both the control and safety rods are assumed to fail in a common mode with a relatively large beta-factor since the relays are located in the same cabinet. Analysis "B" utilizes beta-factors that depend on the number of redundant components in the system (cf. Section 7.4). In both types of analyses, each redundant component is represented by two basic events: an independent component failure and common failure among all the redundant components. The probability of the CCF is set equal to the independent failure probability multiplied by the beta factor. To account for the number of redundant components in the system, a beta-factor of 0.10 is used for less than four redundant components, 0.02 for four or five redundant components, and 0.008 for six or more redundant components. As an example, in the wide-range nuclear channels the beta-factor for the three detectors is 0.10, whereas the beta-factor for the six high-flux bistables is 0.008. Comparing the results of the two types of analyses illustrates the sensitivity of scram failure probability to CCFs.

Human errors are addressed in the fault trees in two general areas: errors that would prevent the automatic scram channels from detecting the abnormal transient events and human errors that would prevent utilization of the manual scram capability. (Details of the human reliability analysis, which is based on the methodology of Ref. [6.7], are given in Section 8.) Human errors in the first area include errors in calibrating the electrical components, in setting the specified trip points, and in performing channel operability tests. The potential for common errors in performing each of the above tasks has been taken into account. Such a common human error is found to be the dominant failure mode in the case of the SOT channels. The same type of common error has some effect on the wide-range nuclear channels but insignificant effect on the low-flow channels.

In the second area, two different human errors are considered: (i) reactor operators failure to manually scram the reactor when the automatic scram fails, and (ii) reactor operators failure to scram the safety rods when the control rods fail to drop for mechanical reasons. With

regard to the manual scram simulation in the fault trees, the approach taken is that the operators should be able to diagnose the abnormal transient condition and realize the need to initiate a scram based on annunciators and alarms independent of the automatic scram channels. It is important to mention that the DAS, which is usually in operation with the reactor at steady power more than 99% of the time, provides a useful and fast means for alerting the operators in the event of LOF or TOP. The DAS, however, has alarms for more than 900 parameters and these parameters appear on the terminal, up to 20 alarming channels at one time, according to the order that the limits are exceeded without any channel priority system. The display of scram related alarms in the middle of a list of less important parameters could delay the operator response.

Human reliability analysis of the second operator error, failure to scram the safety rods, shows that the operators in this situation will be aware of the need to scram and would normally follow the manual-scram emergency procedure. The probability of failure to perform this task is found to be small. However, as a conservative approach, the fault tree modeling accounts for possible dependencies between the two types of human errors.

6.2.1.5 Quantification - Results

The probabilities of scram failure in the event of LOF and TOP are obtained for a set of analyses that cover different transient scenarios and equipment testing intervals and illustrate the effect of potential common cause failures. The results of these analyses are summarized in Tables 6.6 and 6.7. Each table shows the mean probability of scram failure in the abnormal event for a total of eight cases. The probability values listed in the tables include correction for uncertainty coupling. To demonstrate the effect of the SOT channels, the results of the first four cases do not include the automatic scram capability of the SOT channels, whereas the second four cases include the capability of the SOT channels. The probability of scram failure is presented in terms of the three major contributors to failure: signal, mechanical, and electrical. The signal failure represents the failure to send a scram signal to the shutdown strings by either the automatic scram channels or the manual scram channels.

Table 6.6. Scram Failure Probabilities for LOF Events

Case ⁽¹⁾	Total	Signal Failure ⁽²⁾			
		A/S & M/S	A/S Only	Mechanical Failure	Electrical Failure
<u>Without SOT Channels</u>					
I-A	2.53 10 ⁻⁷	7.00 10 ⁻¹⁰	1.40 10 ⁻⁸	2.26 10 ⁻⁷	2.63 10 ⁻⁸
I-B	4.97 10 ⁻⁶	4.24 10 ⁻⁷	8.48 10 ⁻⁶	2.26 10 ⁻⁷	4.32 10 ⁻⁶
II-A	6.70 10 ⁻⁷	4.18 10 ⁻⁷	8.35 10 ⁻⁶	2.26 10 ⁻⁷	2.63 10 ⁻⁸
II-B	5.35 10 ⁻⁶	8.04 10 ⁻⁷	1.61 10 ⁻⁵	2.26 10 ⁻⁷	4.32 10 ⁻⁶
<u>With SOT Channels</u>					
I-A	2.52 10 ⁻⁷	1.75 10 ⁻¹¹	3.50 10 ⁻¹⁰	2.26 10 ⁻⁷	2.63 10 ⁻⁸
I-B	4.72 10 ⁻⁶	1.74 10 ⁻⁷	3.48 10 ⁻⁶	2.26 10 ⁻⁷	4.32 10 ⁻⁶
II-A	2.63 10 ⁻⁷	1.05 10 ⁻⁸	2.09 10 ⁻⁷	2.26 10 ⁻⁷	2.63 10 ⁻⁸
II-B	4.73 10 ⁻⁶	1.84 10 ⁻⁷	3.68 10 ⁻⁶	2.26 10 ⁻⁷	4.32 10 ⁻⁶

Notes:

- (1) Case I: 2/4 channel trip configuration
Case II: 2/3 channel trip configuration without channel HPPF-2
Case A: Limited CCFs are considered
Case B: More extensive CCFs are considered

- (2) A/S: Automatic Scram
M/S: Manual Scram; $HEP = 5.0 \cdot 10^{-2}$

Table 6.7. Scram Failure Probabilities for TOP Events

Case ⁽¹⁾	Total	Signal Failure ⁽²⁾		Mechanical Failure	Electrical Failure
		A/S & M/S	A/S Only		
<u>Without SOT Channels</u>					
I-A	7.64 10 ⁻⁵	7.62 10 ⁻⁵	7.62 10 ⁻⁴	2.26 10 ⁻⁷	2.63 10 ⁻⁸
I-B	1.69 10 ⁻⁴	1.64 10 ⁻⁴	1.64 10 ⁻³	2.26 10 ⁻⁷	4.32 10 ⁻⁶
II-A	8.35 10 ⁻⁷	5.83 10 ⁻⁷	5.83 10 ⁻⁶	2.26 10 ⁻⁷	2.63 10 ⁻⁸
II-B	1.18 10 ⁻⁵	7.26 10 ⁻⁶	7.26 10 ⁻⁵	2.26 10 ⁻⁷	4.32 10 ⁻⁶
<u>With SOT Channels</u>					
I-A	2.16 10 ⁻⁶	1.91 10 ⁻⁶	1.91 10 ⁻⁵	2.26 10 ⁻⁷	2.63 10 ⁻⁸
I-B	9.10 10 ⁻⁶	4.55 10 ⁻⁶	4.55 10 ⁻⁵	2.26 10 ⁻⁷	4.32 10 ⁻⁶
II-A	2.67 10 ⁻⁷	1.46 10 ⁻⁸	1.46 10 ⁻⁷	2.26 10 ⁻⁷	2.63 10 ⁻⁸
II-B	4.98 10 ⁻⁶	4.34 10 ⁻⁷	4.34 10 ⁻⁶	2.26 10 ⁻⁷	4.32 10 ⁻⁶

Notes:

- (1) Case I: Test Interval of HF Channel Components = 100 days
Case II: Test Interval of HF Channel Components = 7 days
Case A: Limited CCFs are considered
Case B: More extensive CCFs are considered

- (2) A/S: Automatic Scram
M/S: Manual Scram; HEP = $1.0 \cdot 10^{-1}$

The mechanical failure represents the failure to release or insert all the control rods and the two safety rods. It has a fixed magnitude of $2.26 \cdot 10^{-7}$ in all the LOF and TOP cases with a major portion of $2.0 \cdot 10^{-7}$ due to the CCF of the control and safety rods. The other portion represents the probability of the control rods mechanical failure (including the failure of magnetic clutches) multiplied by the human error of failing to scram the safety rods. The electrical failure represents the failure of the CP-relays to open and prevent deenergizing the clutches of the control and safety rods. This part of the scram failure depends heavily on the extent of the common cause failure that one postulates and models in the analysis.

As an independent scram system, the SOT channels would normally have a mean failure probability that ranges from $1.0 \cdot 10^{-5}$ to $2.5 \cdot 10^{-2}$ per demand based on the beta-factors assumed among similar components, test interval of the components, and human error in setting the trip points of the channels. The upper value of $2.5 \cdot 10^{-2}$ is determined by a human error in setting the trip points of all the four channels higher than the specified temperatures in the reactor-run plan. The effect of including the automatic scram capability of the SOT channels can be simply viewed in Tables 6.6 and 6.7 as multiplying the failure probability of the signal part by $2.5 \cdot 10^{-2}$. In addition, analysis B takes into consideration potential common cause failures between similar components in the SOT channels and either LOF or TOP channels. The mechanical and electrical parts remain the same since the SOT channels can only affect transient detection and scram signal initiation.

The results of the LOF and TOP analyses show that the modeling of the CCFs has a very large effect on the electrical contribution to the scram failure probability. The electrical failure probability increases by almost two orders of magnitude to $4.32 \cdot 10^{-6}$ in analysis B with more extensive CCFs. This effect dominates the total failure probability and thus minimizes the effect of any improvement in the detection capability introduced by the SOT channels for example.

The LOF results in Table 6.6 show the effect of the number of channels that can detect the transient event by considering two cases. Case I assumes that all the low-flow channels can detect the transient and case II assumes that the sensor of channel HPPF-2 cannot detect the transient. The effect of number of channels is more evident in the signal analysis without the

SOT channels, particularly in analysis A. The results of analysis B, however, converge to almost the same value of about 5.0×10^{-6} irrespective of the number of channels. The essentially identical results of analysis B with the SOT channels indicates that the failure probability is largely determined by the mechanical and electrical failures and by the CCFs among similar components in the low-flow channels and SOT channels.

The TOP results in Table 6.7 show the effect of the test interval of the electrical components by considering two cases. Case I has all the components of the wide-range channels tested every reactor run (100 days) without any monitoring, and case II has all the components tested every week. Case I conservatively models the channels based on the current operation and maintenance procedures. Case II is an alternative that would require weekly operability checking of the channels. The failure probability of the signal part drops from 1.6×10^{-4} to 7.3×10^{-6} , in analysis B without the SOT channels, as a result of decreasing the test interval. This significant improvement in the failure probability of the signal part is an incentive for periodic testing of the wide-range nuclear channel components. It is important, however, to assess any system improvement in terms of the total failure probability which shows significant improvement in both analyses A and B without the SOT channels. Such an improvement becomes much less significant when the analyses take into consideration the SOT channels as the effect of the signal contribution diminishes. The results of Case I are used in the current analysis.

6.2.1.6 Conclusions and Special Insights

- The mean probability of scram failure in the event of LOF ranges from $2.5 \cdot 10^{-7}$ to $5.0 \cdot 10^{-6}$ per demand when all the four low-flow channels can detect the low flow condition. The probability of failure ranges from $2.5 \cdot 10^{-7}$ to $5.4 \cdot 10^{-6}$ per demand when channel HPPF-2 can not detect the LOF condition. The probability of the scram failure varies based on the extent of the CCFs and whether the capability of the SOT channels is taken into account. The higher failure probability in both cases is determined by the CCF among all the CP-relays in the shutdown strings.
- The mean probability of scram failure in the event of TOP ranges from $2.6 \cdot 10^{-7}$ to $1.7 \cdot 10^{-4}$ per demand based on the assumptions used in the analysis with regard to CCFs, test interval of components and whether the redundant capability of SOT channels is included. The higher failure probability is determined by the signal-failure contribution with a test interval of 100 days. This limit is reduced to $9.1 \cdot 10^{-6}$ by taking into account the scram capability of SOT channels.
- Any one of five CP-relays in shutdown strings A and B can release the latches of four or more control rods, allowing the control rods to drop into the core, bringing the reactor core subcritical. Similarly, any one of the two CP-relays on the safety-rod branch of string A can release the clutch of the two safety rods. A common cause failure among all the seven CP-relays sets a lower limit on the scram failure probability of about $4.3 \cdot 10^{-6}$ per demand.
- The failure rate due to mechanical common modes between all the control rods is estimated at $2.0 \cdot 10^{-6}$ per demand and the mechanical failure rate of the two safety rods is estimated at $3.0 \cdot 10^{-4}$ per demand. A common mode failure is postulated between the two types of rods and is estimated at $2.0 \cdot 10^{-7}$ failure per demand. These failure rates are based on a generic single-rod failure rate of $3.0 \cdot 10^{-5}$ per demand and supported by plant specific data for 25 years of reactor operation and a daily test of each control rod that is capable of detecting any potential rod binding.

- The effect of human error in initiating a manual scram in the event of an automatic scram failure is significant. In the absence of scram annunciators, the DAS alarms (not always online) can be a useful means to alert the operators to the possibility of LOF or TOP, but such scram alarms are not currently distinguishable from other causes of DAS alarms (see Section 14).
- Human errors during calibration and trip point setting are found to have small effect on the LOF and TOP channels. Human error in setting the trip points of the SOT channels based on the present startup procedure is relatively large and as a result limits the effectiveness of the SOT channels (see Section 14).

6.2.2 Primary Pumps

6.2.2.1 System Function and Description

Two single-stage centrifugal pumps circulate primary sodium to cool the reactor core. The two pumps are identical and operate in parallel. The pumps are designed to operate in a liquid sodium environment and meet several functional criteria:

- (1) Flow and differential pressure requirements are sufficient to provide cooling for the reactor.
- (2) Leak tightness to prevent leakage of the radioactive primary blanket gas to the building atmosphere, and liquid sodium into the motor.
- (3) Effective biological shielding to contain radiation within the primary tank.

- (4) Accessibility. The assembly is physically arranged to submerge the pump in the sodium. The driver motor is located at the operating floor level, completely out of the radiation field. Removal is accomplished with a simple vertical lift.
- (5) Bearing reliability. The hydrostatic pump bearing is reliable and does not require an external pressure source. Motor bearings support imposed thrust and static loads.
- (6) Flow variation. Pump flow rate is variable from 10 to 100%, with specified acceleration/deceleration rates, during steady-state operation, speed is constant within 0.2% of rated speed.

During full-power operation, 62.5 megawatts of heat is continuously removed from the reactor. To remove this heat, each main pump supplies sodium coolant at about 4500 gpm (0.284 m³/s). Both pumps must operate against the total pressure drop of the primary cooling system, which is 56 psi (0.37 MPa) (from pump outlet to heat exchanger outlet at maximum flow).

Each pump (Figure 6.3) is a vertically mounted structure consisting of a centrifugal sodium pump, a stilling baffle assembly, a shield plug, a hermetically-sealed 350hp (262 kW) drive motor, and a pump shaft. The entire assembly is approximately 27 ft 7 inches (8.4m) long and weighs 27 tons (24,545 Kg). The pump shaft transfers power from the drive motor at the top of the structure to the pump. This shaft passes through the shielding and baffle sections, and connects to the pump impeller.

Each pump is a four-volute 27-inch (0.67m) centrifugal pump designed for liquid sodium service. The 26-inch (0.66m) pump impeller is of conventional design, similar to water pumps of equal capacity. Type 304 stainless steel is used for impeller material (as well as all other

Fig. 6.3. Main Primary Coolant Pump

parts submerged in sodium) due to its compatibility with the high-temperature coolant. The hydrostatic bearing surfaces are faced with Colmonoy to reduce wear and galling that could occur during startup and low-speed operation.

Further details of the pump assemblies are contained in Appendix C.2.

6.2.2.2 Support System Interfaces and Effects

Two support systems are required for proper pump operation:

- (1) Electrical systems - the pump motors require a 480V input from the MG sets which operate from the 2400V switchgear.
- (2) Argon systems - the low pressure argon system provides argon flow along the pump shafts to prevent sodium vapor or sodium oxide from migrating up the pump shaft.

6.2.2.3 Fault Tree Development

The top event for the pump system is loss of both primary pumps. The tree is developed from a failure mode and effects analysis (FMEA). This detailed analysis is discussed in Appendix C.2. The results of the FMEA are presented in Table 6.8 shown below.

6.2.2.4 Component Failure Data, Common Cause, Human Reliability

The data for the pumps is extremely limited; there have been no severe failures of the EBR-II primary pumps and the unique pump design precludes the use of generic data (e.g., from LWRs). Some shaft binding problems have occurred. Hence, the FMEA described above was used as the basic input failure data. Common cause data is not available for similar reasons.

Table 6.8. FMEA Analysis - Loss of Pumps

<u>Basic Event</u>	<u>Failure Rate</u>	<u>Error Factor</u>
Loss of electrical power to pump 1	$9.4 \cdot 10^{-1}*$	3.
Loss of electrical power to pump 2	$9.3 \cdot 10^{-1}*$	3.
Pump 1 motor fails to run	$2.0 \cdot 10^{-5}/h$	3.
Pump 2 motor fails to run	$2.0 \cdot 10^{-5}/h$	3.
Pump 1 shaft binding	$2.9 \cdot 10^{-5}/h$	2.6
Pump 2 shaft binding	$2.9 \cdot 10^{-5}/h$	2.6
Pump 1 shaft broken or seized	$1.9 \cdot 10^{-6}/h$	8.5
Pump 2 shaft broken or seized	$1.9 \cdot 10^{-6}/h$	8.4

*annual frequency

6.2.2.5 Quantification - Results

The top level fault tree (Appendix C.2) has been quantified for loss of both pumps, including the MG sets. The probability of losing both primary pumps in a 7200 hr (300 day) period is estimated at 6.5×10^{-2} with an error factor of 4.0.

6.2.2.6 Initiating Events and Special Insights

The loss of one or both primary pumps can be considered an initiating event in the loss-of-flow (LOF) category discussed in Section 5.

6.2.3 Primary Pumps - Motor Generator Sets

6.2.3.1 System Function and Description

Each of the two primary cooling pumps is powered by an associated motor-generator (MG) set (see Figure 6.4). The MG sets provide power to the primary pumps and are used to control the pump speed through the clutch connecting the motor and generator. Failure of the MG sets to provide power to the primary pumps is an initiating event for a loss of cooling accident.

6.2.3.2 Support System Interfaces and Effects (Table 6.9)

In the event of loss of normal power, specifically the 2400 V supply to the motors of the MG sets, both pumps will coast down with a time determined by the inertia of the entire MG set. This result is identical to that for motor malfunction. A worse case initiating event is the failure of one or both clutches. If coupling is lost, only the inertia of the generator will determine the coastdown period. Clutch coupling has been lost in the past, but only for one pump at a time. The shortest coastdown initiating event caused by failure of the MG set is loss of electrical output from both of the generators. In this case only the inertia of the pump motor and impeller is available to promote flow.

Table 6.9. Motor Generator Support System Dependencies

<u>Component</u>	<u>Support System</u>	<u>Support System Failure Mode</u>	<u>Effect on MG System</u>
Motor	2400 VAC	Loss of voltage	Motor coastdown reactor shutdown
	125 VDC	Loss of voltage	No control of motor input breakers
Clutch	480 VAC	Loss of voltage	Cooling water sump pumps stop
	Cooling water	No water flow	Clutch overheats, alarm condition anticipatory reactor shutdown
Clutch Power Supply	CPS	Loss of voltage	Loss of HP power supply output, loss of MG coupling, pump coastdown
Clutch Controller	CPS	Loss of voltage	No primary pump speed control
Instrumentation	CPS	Loss of Voltage	No alarms for off-normal conditions

In addition to electrical systems needed for monitor and control instrumentation, the primary support systems associated with the MG sets are the 2400 V power connection, Continuous Power Supply (CPS) connections, the 125 V DC supply, and the reactor auxiliaries cooling water supply for the clutch. As described above, loss of 2400 V (or the motor exciter circuits) causes both pumps to stop and causes a reactor scram. Loss of cooling water to the clutches causes an alarm condition, which if not corrected results in an anticipatory shutdown of the reactor. Failure of either one of the two CPS panels results in loss of clutch coupling for the associated MG set, and a dual CPS panel failure causes a dual-pump loss.

Under other accident conditions, the pumps are expected to function normally, provided power is available for the pumps and the clutch control systems, and provided the sodium level in the reactor remains above the pump inlets.

6.2.3.3 Fault Tree Development

The fault tree analysis evolved from the information and data collected and organized in the system notebook in Appendix C.3.

Table 6.10 lists the three major failure modes for each MG set. The top event is defined as a loss of power to the primary pump. For each pump, loss of power results from either loss of motor output, loss of clutch coupling, or a failure associated with the generator. Motor output is lost only if the input power (2400 V) is lost or the motor subsystem has failed. The clutch output is lost either because the motor output is lost or a failure has occurred in the clutch system. The third failure mode is loss of generator output even though the motor and clutch are operating normally.

Assuming that the reactor is operating at full power, and that both primary pumps are operating normally, the longest coastdown time occurs if a motor fails, either because of a power loss or mechanical problem. In this case, the inertia of the motor, generator and pump generally leads to the longest pump stopping time, depending on operating conditions at the time

Table 6.10. Primary Pump Failure Modes/Coastdown Times

	50% Coastdown (s)	100 % Coastdown (s)
Loss of motor power	7	81-84
Loss of clutch coupling	7	23
Loss of generator output	2.5-3	23

of failure. If the coupling between the motor and generator is lost, the pump will coastdown in a shorter time since the motor inertia is no longer in the system. The shortest case occurs when only the inertia of the primary pump is available to extend the coastdown time, leading to the potentially shortest pump stop time, depending upon conditions at the time of failure. Table 6.10 lists the approximate times to reach 50% flow and full pump coastdown for the three cases described above. In the development of the fault trees, it is assumed that the reactor is operating at normal power, that the clutch excitation key switch is locked in the "on" position, and that all necessary support systems are functioning normally.

6.2.3.4 Component Failure Data, Common Cause Failures, Human Error

A review of the fault trees shows that each MG set is for the most part independent of the other. However, several common cause failures can occur which will lead to a coastdown of both pumps. Loss of the 2400 V bus, which powers the motors, causes both MG sets to stop, in addition to being an automatic scram condition. A loss of cooling water would cause both clutches to overheat. A high temperature alarm or low water pressure alarm calls for an anticipatory reactor shutdown but does not lead to an initiating event. Failure of the Module Power Supply which powers both clutch controller modules causes both controllers to automatically switch to bypass mode. Failure of both I/O power supplies causes a loss of pump control, but the HP power supplies remain at the set level, not causing a loss of pump output condition. Failure of the Constant Power Supply (both panels 1 and 2) causes power to be lost to the controller modules, the DCSs and the HP power supplies, resulting in loss of clutch coupling on both MG sets.

6.2.3.5 Quantification/Results

The failure rate data for various components of the EBR-II motor-generator systems as identified in the fault trees of Appendix C.3 were selected from standard PRA references and refined to reflect the operating experience at EBR-II.

The annual frequency of failure of MG Set 1 through any of the three failure modes is 0.94; for MG set 2, the value is 0.93. The difference reflects the commonality of feeder circuits to the I/O and Module power supplies. Analysis of the fault trees for each of the three failure modes results in the data shown in Table 6.11.

If the two individual MG set fault trees are combined, the annual frequency of one of the two pumps failing according to the MODULE analysis is 1.842 with an error factor of 2.8 while the annual frequency of both MG sets failing at the same time is 0.04 with an error factor of 5.

6.2.3.6 Initiating Events and Special Insights

Loss of output from one or both of the MG sets is an accident initiator. In the event of loss of normal power, specifically 2400 V supply to the motors of the MG sets, both pumps will coast down with a time determined by the inertia of the entire MG set. This is the same as if the motor malfunctions. A worse case initiating event is the failure of one or both clutches. If coupling is lost only the inertia of the generator will determine the coastdown period. Clutch coupling has been lost in the past, but only for one pump at a time. The worst initiating event is loss of electrical output from one or both of the generators. In this case only the inertia of the pump motor and impeller is available to slow the pump coastdown.

Table 6.11. Annual Frequencies for MG Sets Failure Modes

<u>Event</u>	<u>Gate</u>	<u>Point Mean Value</u>	<u>Error Factor</u>
Loss of MG Set 1 Motor Output	P0010	0.238	3
Loss of MG Set 1 Clutch Coupling	P0020	0.560	6
Loss of MG Set 1 Generator Output	P0040	0.194	3
Loss of MG Set 2 Motor Output	P0110	0.238	3
Loss of MG Set 2 Clutch Coupling	P0120	0.555	6
Loss of MG Set 2 Generator Output	P0140	0.194	3

6.2.4 Shutdown Coolers

6.2.4.1 System Function and Description

The shutdown cooling system removes heat from the primary sodium tank through two natural convection NaK loops to the atmosphere. The primary purpose of the system is the removal of decay heat from the primary sodium when the secondary sodium system is inoperable. The system is one of the EBR-II "sensitive" systems, i.e., it is critical to the integrity of the primary tank and internals and therefore is an engineered safety feature (ESF). In addition to its function as an ESF, the system is also used to accelerate the reactor cooldown rate in conjunction with the main sodium and BOP systems.

The shutdown cooling system consists of two identical sub-systems. Each of the cooling sub-systems consists of a shutdown cooler plug, a shutdown cooler nozzle, a NaK-air heat exchanger, and associated piping and equipment (Fig. 6.5). Flow occurs in a closed loop between the heat exchangers and individually mounted L1 and L2 nozzles extending into the primary tank. The heat exchanger cores are mounted in two box assemblies on the outside wall of the reactor building. The natural convection flow of NaK in the closed loop systems is completely independent of any electrical power source and functions whenever the bulk sodium in the primary tank is hotter than the outside air. A small expansion tank blanketed with argon gas and part of the box assembly permits thermal expansion of the system. The transfer of decay heat from the NaK coolant to the atmosphere is accomplished by the natural convection of air past a finned-tube heat exchanger. The rate of air flow through the fins is controlled by dampers positioned above and below the heat exchanger and by a sheet metal chimney located above the upper damper. A detailed description of the system is found in Appendix C.4.

6.2.4.2 Support System Interfaces

The shutdown cooling system under normal conditions requires only two support systems, instrument air system (Appendix C.10), and the continuous power system (Appendix C.9). It should be noted that the shutdown cooling system does not rely on any

Fig. 6.5. Shutdown Cooling System

sources of power for its operation during an emergency cooling system situation. However, during normal operation when the shutdown coolers are in standby, it is required that both instrument air and continuous power supplies be available to the system to prevent excessive heat loss from the bulk sodium.

6.2.4.3 Fault Tree Development

The top event (Appendix C.4) for the system is entitled "Any Shutdown Cooling System Malfunction" has two major branches: (1) undercooling by either shutdown cooler, or (2) overcooling (excess heat removal). Each cooler removes 165 kW of power and system failure is defined by either the inability to remove 330 kW or an excessive cooling rate for the current reactor operation.

6.2.4.4 Component Failure Data, Common Cause Failures, Human Reliability

With the exception of the damper actuator mechanism, no failures of the shutdown cooler system have occurred in the lifetime of the reactor. Generic data and some EBR-II experimental data (e.g., the actuators) are used for calculations. Common cause data is considered only for the damper actuators. Because of the passive nature of system operation, proper operation has a low vulnerability to human error. In fact, human error is possible only if the instrument air supply valve is manually closed or if the temperature and flow off-normal alarms are ignored.

6.2.4.5 Quantification - Results

The failure rates and method of calculating unavailabilities can be found in Appendix C.4. Five cases were quantified, as shown in Table 6.12.

Table 6.12. Shutdown Cooler System - Failure Probabilities in a One-Year Period

<u>Case:</u>	<u>System</u>	<u>Undercooling</u>		<u>Overcooling</u>	
		<u>Both</u>	<u>1 of 2</u>	<u>1 of 2</u>	<u>Both</u>
Point mean value:	$3.10 \cdot 10^{-1}$	$6.95 \cdot 10^{-2}$	$1.45 \cdot 10^{-1}$	$1.60 \cdot 10^{-1}$	$1.65 \cdot 10^{-1}$
Error factor:	2.	5.	2.	3.	3.

For the overall failure study the assumption that both overcooling and undercooling could occur is used. The most restrictive of the subtrees is used. One cooler is needed for success in the undercooling case and one cooler is considered to cause excess heat removal.

The calculations are based on a 2400 hour (100 day) reactor run length and 3 runs per year.

6.2.4.6 Initiating Events and Special Insights

Neither failure of the shutdown coolers nor occurrence of any of the basic events in the fault tree is considered to be an initiating event, provided corrective action occurs within ~ 8 hours. As shown by the analysis in Appendix C.4, the technical specification limits on primary tank temperature would not be exceeded within ~ 8 hours, if no other heat transfer mechanism is available. Beyond the 8 hour limit, an anticipatory reactor shutdown is required, which is an initiating event (see Section 5).

6.2.5 Shield Cooling System

6.2.5.1 System Function and Description

The system cools the reactor shielding and recirculates air through the air-baffle tank surrounding the primary tank inside of the blast shield and draws in air around openings near the nozzles in the rotating plugs and boot seals of the nozzles of the primary tank cover. The shield cooling system also maintains the reactor building pressure at slightly less than atmospheric by exhausting air from the reactor building through the plant suspect exhaust system.

The shield cooling system removes heat lost from the primary system bulk sodium before it reaches the shielding which surrounds the primary tank. This cooling is necessary to prevent the concrete shielding from losing its water content resulting in a loss of structural strength and effective shielding. Another purpose of the shield cooling system is to provide circulation of air from the reactor building main floor through the depressed area and the space above the

primary tank. Shield cooling also forms an integral part of the reactor building ventilation system.

Secondary functions of the shield cooling system are to provide a cross connected, partial backup for the thimble cooling system in the event of a failure in the thimble cooling system, to provide a means of air removal from the exhaust hood over the shield plug at the storage pit workstation.

The shield cooling system consists of two recirculation fans, two exhaust fans, three refrigeration systems, manual and automatic dampers, and inter-connecting duct work. An overview of the system is shown in Figure 6.6. The exhaust fans draw approximately 5000 cfm of air into the system from the reactor building atmosphere. This air enters twelve manual dampers located on the rotating plugs. Six of the dampers control flow through the small plug and the other six control flow through the large plug. The air being drawn through these plugs provides cooling around the rotating plug nozzles and boot seals in the primary tank cover. The combined flow from both plugs mixes in the annulus outside the large plug and cools the large plug freeze seal trough. Ducts adjacent to the primary auxiliary pumps busbars direct cooling air along the busbars and into the pump nozzle cooling annuli. There are also six inlet dampers above the biological shields to direct air to cool the area under the operating floor deck plates. Each of these inlets has an inlet filter and a four inch manual damper. A single large inlet filter and 10 inch manual damper is located in the depressed area near the ACS vapor trap which helps cool the depressed area by removing 750 cfm of air. Another 300 cfm of air is normally drawn from the exhaust hood over the shield plug at the storage pit workstation. The manual damper located just downstream of the hood controls the amount of exhaust flow to the system between zero to 530 cfm. In addition to the 5000 cfm of air being drawn into the system there is an additional 15,000 cfm being recirculated through the air baffle tank which surrounds the primary tank. These two flows mix in a common header; the flow then divides and approximately 5000 cfm of air from the return duct passes through the absolute filters which are arranged in two banks. The filter bank consists of eight two inch thick cleanable filters mounted in interlocking frames in a straight bank arrangement. The resistance to air flow of a clean filter does not exceed 0.175 inches w.g. at an air velocity of 628 feet per minute. Only one bank of filters is

Fig. 6.6. Shield Cooling System Pictorial Flow Diagram

in service and the second bank remains in standby to be used in the event of an excessive pressure differential across the operating bank filters caused by build up of dirt and other foreign particles. The operation of the filters is such that usually the operating filters need only be changed one time per year.

Cooling of the circulating air in the shield cooling system is provided by freon refrigeration systems. The normal refrigeration system consists of a 40-ton compressor, a water cooler condenser, alternate air cooling condensing coils, and two normal cooling units with two cooling coils in each unit. The standby refrigeration system consists of two 20-ton units similar in design to the 40 ton unit. The two smaller units serve a dual function and are normally used for a space cooling of the reactor building basement and sub-basement. Each one has two sets of cooling coils, normal and standby. The normal sets of cooling coils for the 20-ton unit are located in the basement and sub-basement space cooling units. The standby sets of cooling coils are located in a branch duct of the shield cooling recirculation line. The water-cooled condensers are located outside of the reactor building in order to minimize the possibility of sodium water reactions. In addition all of the freon lines penetrating the reactor building containment may be isolated by the containment isolation system.

Heat is removed from the refrigeration system by one of two methods: (1) by condensers in the reactor auxiliary cooling water system; or (2) by air cooled condensing coils located in the reactor building inlet air supply. During the cooler months of the year the refrigerant can be lined up to be condensed by air cooled condensing coils. This recycles the heat removal by the shield cooling system back to the supply air to the reactor building and power plant building, saving energy needed by plant systems to heat the supply air.

When the ambient humidity in the reactor building is high, moisture collects on the normal shield cooling coils, the standby shield cooling coils, the basement space cooling unit coils, and the sub-basement space cooling unit coils. This condensate is drained from below the coolant coils to a collection tank in the sub-basement. In order to minimize the volume of water present within the reactor building this collection tank is limited in volume to between one and two liters. A condensate pump activated by a float switch discharges the condensate from the

collection tank to the suspect exhaust tank via the thimble cooling system exhaust duct. The piping from pump to the thimble cooling duct is maintained at 220°F by trace heaters to vaporize water prior to entering the exhaust duct. These heaters aid in the evaporation of condensate in the thimble cooling air flow.

6.2.5.2 Support System Interfaces

The shield cooling system requires electrical systems (to 480 V) and the instrument air system.

The electrical system powers the exhaust and recirculating fans, the normal and standby refrigeration compressors, the motor-operated dampers, the shield cooling condensate pump and line heater, auxiliary and corridor panels for I/C and the annunciators, and the building refrigeration and full isolation relays.

The reactor building instrument air operates the automatic outlet dampers on the exhaust fans.

6.2.5.3 Fault Tree Development

The top level event in the shield cooling system is "Failed or Reduced Heat Removal by Shield Cooling". For purposes of constructing the fault tree (Appendix C.5), the system contains two major subsystems:

- (1) cooling of the rotating plug and nozzle area
- (2) cooling of the primary tank walls and bottom and the biological shield using recirculating heat removal.

The only part of the system which the two subsystems share is a common section of ducting which contains no valves or other active components. The failure of this common ducting is

excluded in this analysis. This failure of either part of the shield cooling system would not adversely affect the operation of the other part of the shield cooling system. In fact, if fans were not operating in one section there may be some beneficial, residual flow in the other section but this has not been completely verified and no credit is taken in this analysis.

6.2.5.4 Component Failure Data, Common Cause Failures, Human Reliability

The failure rates (as listed in Appendix C.5) are selected from standard PRA literature (IREP, CRBRP-4, generic) and refined to reflect EBR-II experience.

6.2.5.5 Quantification - Results

The system has been quantified for the system as a whole, and also for plug and nozzle area cooling and for recirculation cooling as shown in Table 6.13. These results assume a 100 day reactor run cycle, a 72 hour mission time to allow reactor cooldown, and 3 cycles per year.

6.2.5.6 Initiating Events and Special Insights

No initiating events can be caused by the independent failure of the shield cooling system, provided the failure is recoverable within ~ 30 hours. As shown by the analysis in Appendix C.5, the thermal inertia of the reactor shielding is large enough so that ~ 30 hours are available to initiate corrective actions prior to exceeding technical specification limits. Beyond the 30 hour limit, an anticipatory reactor shutdown is required, which is an initiating event (see Section 5).

6.3 Analysis of Support Systems

The support systems include the Thimble Cooling/Stack Exhaust System, the Secondary Sodium System, the Balance-of-Plant (BOP) Systems, the Electrical System, the Instrument and Plant Air System, Water Supply Systems, and the Argon System.

Table 6.13. Shield Cooling-Probability of Failure in a One-Year Period

	<u>Point Mean Value</u>	<u>Error Factor</u>
Shield Cooling System	0.519	5.
Recirculation Cooling	0.236	8.
Plug & Nozzle Cooling	0.285	6.

6.3.1 Thimble Cooling/Stack Exhaust

6.3.1.1 System Function and Description

The instrument thimble cooling system provides cooling for the reactor neutron detectors and the FERD system Benelex neutron moderator. The thimble cooling system also aids in maintaining reactor building pressure, controlled by balanced operation of the reactor building ventilation supply system and the reactor building exhaust. The thimble cooling system exhaust and the shield cooling system exhaust provide the total reactor building exhaust flow of about 7000 cfm. Therefore the thimble cooling system is maintained in continuous operation. If the reactor building ventilation supply or the shield cooling exhaust or thimble cooling system shuts down, the reactor building must be partially isolated to control reactor building pressure.

The instrument thimble cooling system is designed: (1) to cool nuclear detectors that monitor reactor power level, (2) to provide about 2000-cfm reactor building exhaust ventilation flow, (3) to provide cooling for the Fuel Element Rupture Detector (FERD) System Benelex neutron moderator, and (4) to provide low vacuum flow for the primary pumps shield-plug dehydration system.

The thimble cooling system consists of the following major components: Eight instrument thimbles, two turbocompressors, thimble intake dust filters, in-line absolute filters, sound absorbers, and air flow regulating valves. The thimble intake dust filters and the upper ends of the instrument thimbles are located just under the operating floor level of the reactor building; the remainder of the major system components are located in the reactor building subbasement.

There are eight instrument thimbles designated J1, J2, J3, J4, O1, O2, O3, and O4. The thimbles extend from the reactor building operating floor downwards to the reactor core radial centerline and are submerged in 700°F (371°C) primary bulk sodium. Refer to Fig. 6.7, "Instrument Thimble Cooling System Flow and Equipment Diagram," and detailed figures in Appendix C.6.

Fig. 6.7. Instrument Thimble Cooling System Flow and Equipment Diagram

Thimble cooling system flow is provided by either of two identical turbocompressors. One compressor operates continuously with the other in automatic standby. Air enters the O-1, O-2, O-3, and O-4 thimbles through filters located under the reactor building deck plates. The air inlet for thimbles J-1, J-2, J-3, and J-4 is located at grates in the deck plates, where the air passes through double filters. For all the thimbles the air flow is channeled downwards through the center annulus of the thimble to provide cooling for the instruments such as the nuclear detectors. The air then flows up the outer annulus and out through branch A and B piping which routes the air exhaust through individual sound absorbers before entering a common bank of absolute filters, located in the subbasement. The filter bank is monitored by a beta/gamma radiation monitor which indicates and alarms locally and in the main control room. Air flow continues into the operating compressor, discharges through another sound absorber and continues into the reactor building ventilation exhaust system. Thimble cooling system flow is controlled by motor operated manually-positioned valves located at the inlet of each compressor.

The following is a tabulation showing which nuclear detectors are located in each thimble.

J-1	Channel "A" WR Fission Chamber/Channel 7 (CIC)
J-2	Empty
J-3	Channel "B" WR Fission Chamber/Channel 7A (CIC)
J-4	Channel "C" WR Fission Chamber/DAS (CIC)
O-1	Empty
O-2	Empty
O-3	Empty
O-4	Empty

If the thimble cooling compressors fail, the system can be cross-connected to the shield cooling exhaust system. The shield cooling exhaust system can draw about 500-cfm total flow through the thimbles. This prevents the nuclear detector cable temperatures (most sensitive component) from exceeding temperature limits for at least 8 hours.

Since the thimble cooling system provides cooling for the reactor neutron detectors and the FERD system Benelex neutron moderator, the thimble cooling system is maintained in continuous operation. Reactor building pressure is controlled by balanced operation of the reactor building ventilation supply system and the reactor building ventilation air exhaust flow. Exhaust flow is provided by the shield cooling exhaust and thimble cooling systems; this exhaust flow is channeled to the suspect stack through the motor operated suspect stack dampers.

6.3.1.2 Support System Interfaces

The instrument thimble cooling system depends on the electrical systems 480 V power for operation of the turbocompressors and the (inlet) flow regulating valves; the system also requires CPS power for the panel instrumentation in the control room. The system also requires the instrument air system for operation of the automatic discharge dampers.

6.3.1.3 Fault Tree Development

The top level event for thimble cooling system failure is "Thimble Cooling Less Than Adequate". The three major branches to the top level event correspond to inadequate cooling in any one of the three thimbles currently in use, i.e., J1, J3, J4. The cooling for the FERD system Benelex neutron moderator is not included in the tree since the FERD is not part of the reactor safety circuits.

Each thimble has an isolation valve between the thimble and its exhaust line. There is also a bypass tap which allows some form of forced thimble cooling. The bypass is modeled as mode of failure to "short circuit" air flow around the thimble but recovery using the bypass is not modeled. Each thimble has a filter at its inlet. These filters are not the HEPA type filters used in the exhaust section but rather commercial air filters used to prevent dirt and debris from entering the thimbles. Clogging of individual filters as well as common cause clogging is considered. There are two branches ("legs") for exhaust flow.

For this system, "recovery" and "no recovery" are considered since the shielding cooling system can be used as backup and therefore shield cooling failure modes are considered in the fault tree construction.

6.3.1.4 Component Failure Data, Common Cause Failures, Human Reliability

The component failure data is based on standard PRA literature (CRBRP-4, IREP, and generic), combined with EBR-II operating experience. A common cause failure is considered for failure of the commercial inlet filters to the thimbles.

6.3.1.5 Quantification

The thimble cooling system has been quantified for the individual thimbles and for the system as a whole. Due to the symmetry of the system there is no difference in numerical failure values for any of the individual thimbles when recovery by cross-connecting to the shield cooling system is not considered. If the cross-connect is considered, the J-3 thimble as a flow path through two additional valves which will make its failure rate slightly higher than that of the other two thimbles. The results summarized in Table 6.14 are for the no recovery case only. The recovery case is analyzed when the shield cooling and stack exhaust trees are linked with the thimble cooling system (see Section C.5, "Shield Cooling System").

The results assume a 100 day reactor run cycle, an 8 hour mission time to allow reactor cooldown, and 3 cycles per year.

6.3.1.6 Initiating Events and Special Insights

The potential for the thimble cooling system to be an initiating event is considered in that the loss of thimble cooling for an extended period of time would result in

Table 6.14. Thimble Cooling-System Failure Probabilities in a One-Year Period

	<u>Point Mean Value</u>	<u>Error Factor</u>
Thimble cooling system failure (any of three thimbles)	$3.07 \cdot 10^{-1}$	6.
Failure of J-1 thimble	$2.71 \cdot 10^{-1}$	6.
Failure of J-3 thimble	$2.71 \cdot 10^{-1}$	6.
Failure of J-4 thimble	$2.71 \cdot 10^{-1}$	6.

the neutron detector cable and connector temperatures increasing beyond the maximum allowable limit of 230°F. If this condition is undetected there is a possibility of erroneous signals from the neutron detectors. Since EBR-II uses a variety of scram functions, this could only lead to a problem if the reactor was being controlled, either manually or automatically, based on the high flux channel readings. To prevent this, the thimble cooling system is designed to alarm on several abnormal conditions including high temperature. Emergency procedure EP 3-7 of the operating procedures outlines the operator action required in the event of abnormal thimble cooling. Thus a failure of the thimble cooling system has to be combined with a failure of the thimble cooling alarms system or operator error to fail to respond correctly to the alarm condition, and a condition of controlling on the flux level instruments in order to lead to an initiating event.

6.3.2 Secondary Sodium System

The failure of the secondary sodium system is one of the loss-of-heat-sink (LOHS) accident sequences (see Section 5). Direct plant data was used to estimate failure frequencies for initiation event quantification.

6.3.3 Balance-of-Plant (BOP)

A BOP failure is one of the LOHS accidents (Section 5).

6.3.4 Electrical System

6.3.4.1 System Function and Description

The electrical distribution system provides power to all components and equipment of the EBR-II facility. The EBR-II substation steps down the 138 kV power from the Idaho National Engineering Laboratory (INEL) loop to 13.8 kV and matches the 13.8 kV turbine generator output voltage to the 13.8 kV from the main power transformers. Circuit breakers, motor starters and disconnect switches located in switchgear cubicles, motor control

centers and distribution panels provide individual feeder controls. Numerous relays guard against over-voltage, under-voltage and over-current to provide protection for most circuits. Figure 6.8 shows a summary single-line diagram of the EBR-II electrical distribution system.

EBR-II is a power producing reactor, and the turbine generator has a power output capability of 20 MW at 13.8 kV. The nominal generator output is 20 MW, but is also given on certain diagrams and some manuals as 25600 kVA. The electrical distribution system is designed to utilize the turbine-generator-produced power to satisfy power requirements of the EBR-II facility when the reactor is operating. When the turbine generator is producing more power than needed by the ANL-W facility, the excess 13.8 kV power is stepped up by the substation power transformers and fed to the INEL power loop.

The electrical distribution system is divided essentially into the following systems.

- 138 kV Distribution/Substation Equipment

- 13.8 kV Distribution System

- 2400 V Distribution System

- 480 V Main Distribution System

- 480 V Emergency System

- Continuous Power Supply System

- 125 V DC Distribution System

The detailed system analysis (Appendix C.9) considers each of the subsystems separately and calculates a subsystem failure rate for seven top fault tree levels. The general system description that follows indicates the major functions of each subsystem.

Under normal operating conditions, power utilized in the EBR-II facility is distributed from the 13.8 kV switchgear. The 2400 V distribution system supplies 2400 V power for water pumps, induction heaters, sodium pump motor generator sets, and other facilities at the ANL-W site.

Fig. 6.8. Single Line Electrical Diagram EBR-II Electrical Power Distribution

The 480 V distribution system is divided into a normal power distribution system and an emergency power distribution system. The system supplies power to all EBR-II equipment and components requiring 480, 208, or 120 V for operation. The 480 V emergency system furnishes power at 480 V from emergency diesel generators to components that must continue to operate if the normal source of power fails and a reactor shutdown is to be avoided.

The continuous power supply system supplies constant regulated 120 V AC power to the nuclear and process instrumentation and control equipment even after normal power and the emergency diesel generators fail. The continuous power system utilizes two uninterruptible power supplies (UPS) to power two major distribution panels, Continuous Power Panels 1 and 2.

The 125 V DC Distribution System provides control power for electrically operated air circuit breakers, isolation valves, emergency lighting, and essential turbine generator auxiliaries and controls.

The 2400 V system is directly connected with the control power chain of the reactor. An undervoltage condition will cause the 2400 V undervoltage relay to operate contacts in string A of the reactor shutdown system, causing string A to de-energize, scrambling the reactor in anticipation of a loss of primary cooling flow. The reactor primary pumps enter into a coastdown mode.

The loss of all 480 V AC systems, including the emergency power from the diesel generators, does not cause a reactor scram. If 480 V AC feeder power is lost to the charging systems of the auxiliary pump battery, station battery or continuous power supply battery, an anticipatory reactor shutdown is required before required functions are jeopardized by excessive battery discharge.

Personnel safety is ensured without availability of the emergency diesels. The emergency diesels are required only for protection of equipment, particularly the radiological and blast shields. If the radiological and blast shields are overheated, reactor operation would have to be

delayed until necessary analysis and/or modifications could be completed. Normal and backup means of supplying power are considered satisfactory to avoid such damage.

The core and primary tank are adequately cooled without the availability of the diesels via convective cooling of the reactor core and the shutdown coolers providing cooling to remove decay heat, maintaining bulk sodium temperatures below limits associated with the primary tank.

The loss of either Continuous Power Panel will cause a reactor scram and also loss of clutch control power to the primary pump connected to the Continuous Power Panel that is lost. The scram occurs in anticipation of a reduction of primary cooling flow and is caused by a control power interruption because of various fail-safe scram instrumentation systems on each Continuous Power Panel and, in the case where Continuous Power Panel No. 1 fails, also with the loss of the control rod latch power bus. To ensure availability of continuous power in the event of an outage of normal power during reactor operation, one of the two continuous power supply batteries and its associated UPS system must be operable and normal feeder power must be available to at least the operable UPS unit. If normal feeder power to both units is lost or if neither UPS unit is operable, an anticipatory shutdown of the reactor must be performed. Normally, loss of a UPS does not cause a reactor scram, since the two Continuous Power Panels are then fed from the operable unit or normal power.

The simultaneous loss of both Continuous Power Panels is an unlikely but significant initiating event, since it results in a total loss of primary flow and requires a reactor scram to limit fuel damage.

Loss of any 125 V DC bus does not cause a reactor scram. During the total loss of 125 V DC power, several critical loads will lose power. These include the control power for the 13.8 kV, 2400 V, and 480 V switchgears, support systems for the turbine generator, and selected motor control center functions. If the station battery 125 V DC system fails or normal feeder power to the station battery charger is lost, an anticipatory reactor shutdown is required.

The major concerns with a complete loss of 125-V DC are:

1. The turbine generator loses automatic transfer control, field rheostat control, remote control of the governor and the load limit, the DC-powered seal oil pump, electrical trip protection for the turbine stop valve, and trip protection for the generator output breaker and the field breaker. The turbine generator cannot be tripped from the control room and must be shut down manually. To prevent possible damage to the generator armature due to overvoltage, the generator field breaker must be opened immediately following opening of the generator output breaker.
2. There is no trip protection for any of the breakers in the 13.8-kV and the 2400-V switchgears. Most 480-V breakers have thermal overload and instantaneous short circuit protection, but all other fault protection is lost, including the ability to automatically switch to emergency diesel power.
3. Power to the DC-powered seal-oil pump is lost. If the seal-oil pressure to the 13.8-kV generator is lost, hydrogen will leak out and create an explosion hazard and fire hazard in the power plant building. Normal AC power or the 400-kW diesel generator provide power to the AC seal-oil pump and the turning-gear oil pump. The AC-powered pump will provide sufficient oil to the hydrogen seals of the generator to prevent loss of hydrogen if the 125 V DC-powered seal-oil pump is lost.
4. The reactor-building-isolation relay deenergizes causing reactor building isolation.

The electrical distribution systems are expected to function normally under postulated accident conditions, unless physical damage interrupts service. External causes such as fire or seismic events can cause loss of various electrical systems.

6.3.4.2 Support System Interfaces

The main support system interface is with the BOP, essentially at the turbine generator. The generator output is connected to the 13.8 kV switchgear through an air circuit breaker. The generator operates in parallel with the INEL system supplying the power to operate the EBR-II site and also feeding additional power to the INEL loop.

6.3.4.3 Fault Tree Development

The top events of the electrical subsystems are as follows:

- (1) Failure of the 138 kV Distribution/Substation Equipment
- (2) Loss of 13.8 kV Bus Power
- (3) Loss of 2400 V Bus Power
- (4) Loss of 480 VAC - Main Distribution System (10 buses)
- (5) Loss of 480 VAC - Emergency System (8 buses)
- (6) Loss of Continuous Power Panels (No. 1 and No. 2)
- (7) Loss of 125 V DC Distribution System (8 buses)

6.3.4.4 Component Failure Data, Common Cause Failures, Human Reliability

Failure rate data for the system fault trees were selected from standard PRA references (CRBRP-4, IREP, generic) and refined to reflect EBR-II operating history. Throughout the electrical system fault trees, a number of common mode failures are considered.

6.3.4.5 Quantification

As a first step, each of the seven major subsystems (and the buses) was analyzed for mean failure probability, as shown in Table 6.15.

6.3.4.6 Initiating Events and Special Insights

Loss of the electrical power subsystems can produce several potential initiating events:

- (1) Loss of 13.8 kV power produces a reactor scram.
- (2) Loss of the 2400 V distribution system causes a reactor scram and immediately puts the primary pumps into a coastdown mode.
- (3) Loss of all 480 V AC systems does not cause a reactor scram. However, the EBR-II Technical Specifications require some of the 480 V electrical equipment to be available for continued reactor operation. The required actions and anticipatory shutdowns are detailed in Appendix C.9.
- (4) Loss of either Continuous Power Panel (120 V AC) will cause a reactor scram and loss of control power to the primary pumps. Details of UPS requirements and anticipatory reactor shutdown requirements are found in Appendix C.9.

Table 6.15. Failure Probabilities for Electrical Systems in a One-year (300 day) Period

<u>System</u>	<u>Failure Rate Per Year</u>	<u>Error Factor</u>
138 kV Distr./Substation Equipment	$5.1 \cdot 10^{-1}$	2
13.8 kV Bus Power	$2.2 \cdot 10^{-3}$	4
2400 V Bus Power	$4.2 \cdot 10^{-3}$	3
480 V Main Distr. System (10 buses)	$4.2 \cdot 10^{-3} - 1.1 \cdot 10^{-2}$	3
480 V Emergency System (7 buses)	$6.0 \cdot 10^{-3} - 1.6 \cdot 10^{-2}$	3-4
Continuous Power System (Panels 1 and 2)	No. 1 - $5.4 \cdot 10^{-2}$	5
	No. 2 - $5.4 \cdot 10^{-2}$	5
125 V DC Distr. System (8 buses)	$5.1 \cdot 10^{-2} - 9.6 \cdot 10^{-2}$	3-4

- (5) The loss of 125 V DC power does not cause a reactor scram. However, the 125 V DC system buses provide power for several critical loads of EBR-II. The detailed requirements prior to anticipatory reactor shutdown are found in Appendix C.9.

6.3.5 Instrument and Plant Air

6.3.5.1 System Function and Description

The instrument air system is the primary air system for EBR-II components and systems and provides a source of cool, dry, filtered and oil-free service air to all control equipment via the instrument air distribution system and the plant air distribution system under normal operations.

The plant and instrument air systems are cross-connected. Figure 6.9 shows a diagram of the combined systems. Under normal conditions, the plant air compressor is shut down and isolated and one of the two instrument air compressors furnishes air to both systems. The two instrument air compressors are operated as one on-line compressor and one on standby (alternated weekly).

The plant compressor is used as a high quality backup system when both instrument air compressors are inoperative, although the plant compressor cannot provide enough air to maintain the entire system, i.e., some portion must be isolated.

In addition, there is a reactor building emergency air compressor (air-cooled) which automatically activates when the reactor building is isolated or when there is a loss of air pressure to the reactor building. This system also supplies air to the ACS during fuel handling operations. (See Appendix C.10 for detailed descriptions.)

Fig. 6.9. Instrument and Plant Air Systems

6.3.5.1.1 Instrument Air Compressors

The instrument air compressors are Joy Class WNOL-112B type compressors. They are two-cylinder, two-stage units constructed for continuous heavy duty service. The cylinders are double acting and operate from a common single-throw crankshaft. Each compressor is rated 398 cfm at 5000 ft elevation, 110 psig discharge pressure, while operating at normal speed (570 rpm).

The compressor cylinders are water cooled and require no lubrication. Teflon piston rings are used to ensure clean, oil-free air is furnished to the instrument air system.

6.3.5.1.2 Plant Air Compressor

The plant air compressor is a vertical Ingersoll Rand unit belt-driven by a 50 HP motor. The compressor is rated 200 cfm at 100 psig. Teflon piston rings are used to eliminate the requirement for cylinder oil lubrication. This serves to minimize possible air system oil contamination. Chassis oil lubrication is provided by a gear pump off the compressor crankshaft. The compressor and aftercooler are normally cooled by the raw water system.

6.3.5.2 Support System Interfaces

In addition to the cross-connection between the two air systems, a limited amount of air can be supplied to both systems from the FCF and the HFEF facilities via a normally closed valve in the FCF facility basement.

The Raw Water System is used to provide the normal cooling water to the plant and instrument compressors. The Plant and Reactor Auxiliary Cooling Water Systems can be lined up as backup cooling sources to both compressor systems. The electrical system supplies power to both systems and the reactor building emergency air compressor.

6.3.5.3 Fault Tree Development

Three fault trees are developed in Appendix C.10. The top level events are:

- (1) Loss of Plant Air Distribution
- (2) Loss of Instrument Air System Output
- (3) Loss of Emergency Air System Output

For the plant air system, the analysis assumes that the system is operating in normal mode, with the plant air compressor shut down and the plant air distribution system supplied by the instrument air system. It is assumed that valves used for maintenance, such as drains, have been returned to their operating positions, and that manual valves do not fail from the operational positions.

For the instrument air system, the analysis assumes that the system is operating normally, with one instrument air compressor running with the controller in the MAN position, and the other in the AUTO position. Since the plant air compressor cannot provide a complete backup of the instrument air system, it is not included as a backup in the fault trees. It is assumed that valves used for maintenance have all been returned to their operating position, and that normal valves will not fail from their operational position. The distinction between the primary and standby compressor systems is artificial, since the roles are reversed on a regular basis. The analysis is identical for both combinations of primary/standby assignments except for the 480 V emergency power circuit. Since the two circuits have nearly the same failure probability, little difference in the final result will be seen between the two configurations.

The reactor building emergency air distribution system is considerably simpler than the normal instrument air system, but its failure modes are comparable. The fault tree assumes that

the emergency compressor controller is in the STANDBY mode and is started by a low pressure signal.

6.3.5.4 Component Failure Data, Common Cause Failures, Human Reliability

For all three air systems, the failure rate data was selected from standard PRA references (CRBRP-4, IREP, generic) and refined to reflect EBR-II operating experience.

6.3.5.5 Quantification

6.3.5.5.1 Instrument Air System

Because of the dual-compressor nature of the system, the mission time for the primary (i.e., operating) system was set to one week. Since the roles of operating and standby compressors are routinely reversed each week, this is a more accurate assumption than requiring one compressor to run for the entire reactor run time. Similarly, a conservative estimate of one week was used for the standby system. On the average, it is expected that the standby system would be called on to function one-half of the mission time of the primary system, and using the full mission time produces a conservative failure probability.

The point mean probability for the loss of instrument air distribution in a 300-day period is 0.216 with an error factor of 8.

6.3.5.5.2 Plant Air System

Since the plant air system is not normally in operation, it is difficult to assign a mission time for failure calculations. The system is tested each week, and is used only as a backup to the instrument air system, which cycles compressors on a one-week basis. It is assumed that the plant air system would have to function for no longer than one week before the instrument air system is restored to operational status.

The point mean probability for the loss of plant air distribution in a 300-day period is 0.321 with an error factor of 5.

6.3.5.5.3 Emergency Air System

The emergency air system is used only on a temporary basis in the event of reactor building isolation. The duration of its average mission time has been set to one day.

The point mean probability for loss of the emergency air system in a 300-day period is 0.026 with an error factor of 3.

6.3.5.6 Initiating Events and Special Insights

For the three air systems, no initiating events can be caused by system failures.

It should be noted that the conditions following certain initiating events can be mitigated or degraded by the presence or absence of the instrument air system (see System Interconnections in Appendix C.10). Total failure of the instrument air system would necessitate manual cross-connections to the plant and could require an anticipatory reactor shutdown.

6.3.6 Cooling Water System

6.3.6.1 System Function and Description

The cooling water system supplies water for heat transfer to various components located in the power plant, the sodium boiler building and the reactor building. After heat transfer occurs, the cooling water is returned to a cooling tower where the water is cooled through evaporation and conduction and again recirculated through the system.

The cooling water support system consists of the reactor auxiliaries cooling water system and the plant cooling water system. These systems are supported by the raw water system and two cooling towers.

The reactor auxiliaries cooling water system provides the required cooling for various components associated with the primary coolant systems. The system may be cross-connected to supply the plant cooling water system, but only if sufficient cooling is available to the reactor auxiliaries components. Interconnection of the two systems is made by opening the appropriate manually operated cross-connection valves.

The plant cooling water system furnishes cooling water to the power plant building, sodium boiler building, FCF, laboratory and office building and the fuel assembly and storage building. This system can be cross-connected as backup to the reactor auxiliaries cooling water system.

The raw water system provides make-up water for all systems and can provide a backup cooling water source for the reactor auxiliaries and plant cooling water systems via a backflow preventer assembly (emergency cross-connect). The backflow preventer blocks inadvertent contamination of the raw water system by the chemically treated plant or reactor auxiliary cooling water systems. System details are provided in Appendix C.11.

6.3.6.1.1 Reactor Auxiliaries Cooling Water System

The reactor auxiliaries cooling water system consists of the auxiliary cooling tower, two cooling water pumps, a Cuno filter, piping and valves. Figure 6.10 shows a schematic of the system. This system provides heat removal for the following components:

- Primary pumps MG eddy current clutches (2)
- Primary purification silicone/water heat exchangers (2)
- Reactor shield cooling freon condenser (40-ton unit)
- Reactor building basement air freon condenser (20-ton unit)
- Reactor building sub-basement air freon condenser (20-ton unit)
- Reactor building ventilation freon condenser
- Plant and instrument air compressors (backup supply)

In addition, the system may be cross-connected to supply the plant cooling water system, but only if sufficient cooling is available to the reactor auxiliaries components. The system also provides a backup, via a check valve, to the auxiliary boiler feedwater pump clutches.

Cooling water is pumped from the auxiliary cooling tower basin by either of two 275 gpm centrifugal reactor auxiliary cooling water pumps. The water is then discharged at approximately 55 psig and 65° F through a Cuno filter in the power plant pump pit to a distribution pipe. Several parallel flow paths are provided at this point through the primary systems components and coolers. Cooling water is then collected in a return pipe which goes directly back to the auxiliary cooling tower.

6.3.6.1.2 Plant Cooling Water System

The major elements of the plant cooling water system are the two cooling water pumps, miscellaneous valves, the Cuno filter and the auxiliary

cooling tower. Figure 6.11 shows the plant cooling water system configuration. Cooling water is supplied from the auxiliary cooling tower basin by either of two 1080 gpm centrifugal plant cooling water pumps. The water is then discharged through the Cuno filter at approximately 70 psig and 65° F into a distribution header located in the power plant building. Several parallel flow paths are provided at this point to various components and coolers. Cooling water is then collected in a return pipe which goes directly back to the auxiliary cooling tower.

The plant cooling water system provides heat removal for the following EBR-II components:

(1) Sodium boiler building:

- Secondary sodium main EM pump
- Dowtherm/water heat exchanger
- Sodium recirculating pumps (2)
- Dowtherm pumps (2)

(2) Power plant building:

- Sample station sample coolers (5)
- Feedwater pump no. 1
- Feedwater pump no. 2
- Blowdown after-cooler
- Turbine-generator lube oil Coolers (2)
- Generator hydrogen coolers (4)
- Instrument air compressor no. 1 - alternate supply
- Instrument air compressor no. 2 - alternate supply
- Plant air compressor - alternate supply
- Freon condenser - control room air conditioning
- Freon condenser - DAS room
- Freon condenser - room 14 (Computer Development Lab.)
- Feedwater heater level alarm panel cooler

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Fig. 6.11. Plant Cooling Water System

6.3.6.2 Support System Interfaces

Both of the above systems are supported by the raw water system, the instrument air system and the electrical system.

6.3.6.3 Fault Tree Development

6.3.6.3.1 Reactor Auxiliaries Cooling Water System

The top level event for this system is "Loss of Reactor Auxiliaries Cooling Water". The two main branches of the fault tree reflect the loss of the centrifugal pumps output or blockage of the Cuno filters.

It is assumed that the reactor auxiliaries cooling water system is operating normally using one operating auxiliary cooling water pump, and the other pump is in standby mode. Since cross connections to the plant water or raw water systems are manually activated and are not the normal operating procedure, these systems are not included as backups to the reactor auxiliaries system. Manual valves that are in their correct operating position are assumed to remain so, and no human errors of commission in changing valve positions are taken into account.

6.3.6.3.2 Plant Cooling Water System

The top level event for this system is "Loss of Plant Cooling Water". The two main branches of the fault tree reflect the loss of centrifugal pumps output or blockage of the Cuno filters.

It is assumed that the plant cooling water system is operating normally using one operating plant cooling water pump, and the other pump is in standby mode. Since cross connections to the reactor auxiliaries water or raw water systems are manually activated and are not the normal operating procedure, these systems are not included as backups to the plant cooling water system. Manual valves that are in their correct operating position are assumed

to remain so, and no human errors of commission in changing valve positions are taken into account.

6.3.6.4 Component Failure Data, Common Cause Failures, Human Reliability

The component failure data were selected from standard PRA references (CRBRP-4, IREP, generic) and refined to reflect EBR-II operating experience.

6.3.6.5 Quantification - Results

Because of the dual-pump nature of both systems, the mission time for each of the primary (i.e., operating) pumps was set to one week. Since the roles of operating and standby pumps are routinely reversed, this is a more accurate assumption than requiring one pump to run for the entire reactor run time. Similarly, a conservative estimate of one week was used for the standby pump system. On the average, it is expected that the standby system would be called on to function one-half of the mission time of the primary pump, and using the full mission time produces a conservative failure probability.

The point mean probability for the loss of the reactor auxiliaries cooling water system in a 300-day period is 0.203 with an error factor of 9.

The point mean probability for the loss of the plant cooling water system in a 300-day period is 0.202 with an error factor of 9.

6.3.6.6 Initiating Events and Special Insights

No initiating events can be caused by the failure of either the reactor auxiliaries cooling water system or the plant cooling water system. In the event of a total failure of either system without rapid cross-connection to backup supplies, the event would require an anticipatory reactor shutdown.

Section 6.0 References

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7.0 DATA BASE DEVELOPMENT

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7.0 DATA BASE DEVELOPMENT

7.1 Discussion

7.1.1 Sources of Data

The development of a data base for the EBR-II PRA presents some difficulties because, in common with other DOE Class A facilities, EBR-II has a large number of unique components. The data which can be derived from EBR-II records, even assuming impeccable record keeping, is limited to 25 reactor years of operation. This position can be improved somewhat because EBR-II FFTF and the Japanese LMR program have been contributing data to a LMR data base, CREDO, based at Oak Ridge National Lab [7.1], which can be used to augment this database. The component data used in this PRA have been developed as follows:

(1) Establish an initial data estimate for a given event using generic data [7.2], the IREP data base [7.3] and the data given in the Clinch River Breeder Reactor PRA [7.4]. This estimate will be the only data gathered in most cases where the fault is not specific to EBR-II or there are no identified concerns about the relative performance of the EBR-II equipment.

(2) CREDO data was used to improve the estimates for LMR-specific equipment. This data was confirmed against EBR-II records for the failures deemed to be particularly important.

(3) For the RSS system, a detailed survey of plant specific data was undertaken because of the importance of the system and the sensitivity of the results to common mode failures in this system.

Many of the systems analyses performed were used to evaluate initiating event frequencies and in some of these cases the results could be directly compared to the historical frequency of faults derived from a survey of EBR-II UORs. In general the frequencies derived

from plant data have been used, the insight gained from the comparison being incorporated into the fault tree quantification as plant specific data.

The basic data in the following tables can therefore be characterized as plant specific or generic. In addition, there are human errors identified and quantified, as discussed in Section 8, and dependent failures which are discussed in the following sections.

7.1.2 Failure Probability Calculation Methodology

Both generic and plant specific data normally provide information on failure rates or demand failure probabilities for selected components. In addition, plant records provide information on test, repair, and maintenance periods and frequencies. Fault tree analysis, however, requires input data in the form of failure probability for operating components, and unavailability for standby components. Failure probabilities and unavailabilities must be calculated using the raw data on failure rates and test and maintenance information.

Different models can be used in calculating the failure probability or unavailability of a component, depending upon the component type and its operating condition. Table 7.1 lists the different models available in the EBR-II PRA. The calculation models apply as follows:

1. Type 0 - this model applies to components that can fail on a failure per demand basis. The raw data for such components is already in the form of a probability of failure per demand, and no further correction is needed when several demands are required of the component during an accident sequence. This input mode can also be used to input probability and error factor results from external calculations which fall outside these models.
2. Types 1 and 2 - this is the standard failure probability model for an operating component without repair. The failure rate is known from generic sources or it has been estimated from plant data. Two types based on the same model are provided to differentiate between the operating time of the component: type 2

Table 7.1. Failure Data Calculation Models

Type	Description	F1	F2	F3	Probability Model
DT					
O	Probability Standby component, failure per demand	P δ_d	EF EF	- (d)	$P = \delta_d * (d)$
1	Operating component without repair	δ_o	EF	-	$P = 1 - \exp(-\delta_o \text{TMIS})$ or $P = \delta_o \text{TMIS}$
2	Operating component without repair	δ_o	EF	-	$P = 1 - \exp(-\delta_o \text{TRUN})$ or $P = \delta_o \text{TRUN}$
3	Operating component with repair	δ_o	EF	τ	$P = (1 - \exp(-(\delta_o + 1/\tau) \text{TMIS})) * \delta_o \tau / (1 + \delta_o \tau)$ or $P = \delta_o \min(\text{TMIS}, \tau)$
4	Operating component with repair	δ_o	EF	τ	$P = (1 - \exp(-(\delta_o + 1/\tau) \text{TRUN})) * \delta_o \tau / (1 + \delta_o \tau)$ or $P = \delta_o \min(\text{TRUN}, \tau)$
5	Standby component, failure in time	δ_o	EF	T_t	$P = 1 + (\exp(-\delta_o T_t) - 1) / \delta_o T_t$ or $P = \delta_o T_t / 2$
6	Standby component, test/repair outage	τ_t	EF	T_t	$P = \tau_t / T_t$
7	Standby component, maintenance outage	τ_m	EF	T_m	$P = (\tau_m / T_m) / (1 + \tau_m / T_m)$
T	True house event	-	-	-	$P = 1.0$
F	False house event	-	-	-	$P = 0.0$

Definitions:	P	Basic Event Probability	τ	Average repair time, hr
	EF	Error factor	δ_o	Standby failure rate, hr ⁻¹
	δ_d	Failure rate per demand	T_t	Test interval, hr
	d	Number of demands	τ_t	Average test/repair time, hr
	δ_o	Operating failure rate, hr ⁻¹	τ_m	Maintenance time, hr
	TMIS	Mission time, hr	T_m	Average maintenance interval, hr
	TRUN	Reactor Run time, hr		(inverse of maintenance frequency)

Notes: An optional third variable can be entered for types 1 and 2, to be used instead of TMIS or TRUN. If F3 is 0 or is not provided, then TMIS and TRUN are used in the formulae.
The rare event approximations for types 1, 2, 3, 4 and 5 are used when δt is smaller than 0.01.

applying to components in continuous operation during a reactor run, and type 1 for those components that are only in operation during off-normal conditions.

3. Types 3 and 4 - these models are the unavailability models of types 1 and 2 corrected for the possibility of repair of the component. Plant data supplies the information on repair times. Again the distinction is made between components operating in normal or off-normal conditions.
4. Type 5 - this model provides the unavailability of a standby component that can fail in time. The standby component failure is undetected until the component is demanded or tested. Generic or plant-specific data are used in calculating the standby failure rate, while plant specifications supply information on test intervals.
5. Type 6 - this is the model to estimate the unavailability of a component due to test or repair outage. In the case of test outage, T_t is the average test outage time, while for repair outage unavailability, T_t is interpreted as:

$$T_t = T_R F_R,$$

where T_R is the average component repair time, and F_R is the expected repair frequency per test period.

Plant data and records must be used in estimating all the variables involved in this model.

6. Type 7 - model for the unavailability of a component due to maintenance outage. Again, plant data provides the necessary information.

7. Types T and F - these two models simply set the failure probability of an event to 0 or 1, for its use as a house event in a fault tree to trigger a branch of a fault tree on or off.

Data types 1, 2, 3, 4, and 5 can be simplified to their rare event approximation when the product of the failure rate and the corresponding operating or standby time is smaller than 10^{-2} . The applicable rare event approximations are:

- for types 1 and 2:
 $\delta_0 T$, where T is TMIS or TRUN, respectively
- for types 3 and 4:
 $\delta_0 t$, where t is the minimum of t and the minimum time (TMIS or TRUN)
- for type 5:
 $1/2 \delta_s T_s$.

7.2 Data

7.2.1 Data Used in Quantification

The data used in the quantification of the fault trees is given in the various system notebooks and the tables of data repeated here system by system, Tables 7.2 - 7.21 and 7.24.

7.2.2 Plant Specific Data

Plant-specific data relates to such data as test and maintenance, and specific pieces of equipment. There is almost no test and maintenance related unavailability of safety significant equipment to be considered. This arises because of the EBR-II operating philosophy. Maintenance is performed with the reactor shutdown and all safety significant systems are tested

Maintenance is performed with the reactor shutdown and all safety significant systems are tested before returning to power. Test and maintenance unavailabilities are summarized in Table 7.22.

Plant specific component failure data was generated in areas where the system possesses attributes which did not have direct analogies in the generic database or was deemed to be significantly important. The plant specific data used in the EBR-II PRA are summarized in Table 7.23.

7.2.3 Special Data Considerations

Several top events in the event trees were not developed into fault trees. In certain cases they represent subjectively evaluated probabilities related to the occurrence of different physical phenomena. This includes most of the evaluation of the local fault and steam generator tube rupture event trees. In addition, there is one important undeveloped top event whose quantification can be related to plant data - this is the probability that the system is operating on hotel load. These special data are listed in Table 7.24. A number of initiating event frequencies, particularly those relating to very rare events, are evaluated as special cases using engineering judgement. These events are discussed on a case-by-case basis in the treatment of initiating events.

Table 7.2. Reactor Shutdown System Data

PAGE 1

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
RAM2A1QW	Current Amplfr Reduces Signal to Trip unit	1.0E-6/h [1]	10	1.0E-6/h [2]	10			5.0E-6/h [3]	20
RAM2B1QW	Current Amplfr Reduces Signal to Trip unit	1.0E-6/h [1]	10	1.0E-6/h [2]	10			5.0E-6/h [3]	20
RAM2C1QW	Current Amplfr Reduces Signal to Trip unit	1.0E-6/h [1]	10	1.0E-6/h [2]	10			5.0E-6/h [3]	20
RAM2Z1QU	CCF - All Current Amplfr Reduce Signals to Trip unit	1.0E-7/h [4]	10						
RBI1A0QW	Trip Unit of Strings A & B Fails	5.0E-7/h [1]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10
RBI1B0QW	Trip Unit of Strings A & B Fails	5.0E-7/h [1]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10
RBI1C0QW	Trip Unit of Strings A & B Fails	5.0E-7/h [1]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10
RBI1D0QW	Trip Unit of Strings A & B Fails	5.0E-7/h [1]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10
RBI1Z0QU	CCF - All Trip Units in All Channels Fail	4.0E-9/h [6]	10						
RBI2A0QU	CCF - All Bistables of Channel A Fail	1.0E-7/h [7]	10						
RBI2A1QW	Bistable of String A Fails to dtct High Flux	5.0E-7/h [8]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10
RBI2A2QW	Bistable of String B Fails to dtct High Flux	5.0E-7/h [8]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10
RBI2A3QW	Bistable of String A Fails to dtct Low Voltage	5.0E-7/h [8]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10
RBI2A4QW	Bistable of String B Fails to dtct Low Voltage	5.0E-7/h [8]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10
RBI2B0QU	CCF - All Bistables of Channel B Fail	1.0E-7/h [7]	10						
RBI2B1QW	Bistable of String A Fails to dtct High Flux	5.0E-7/h [8]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10
RBI2B2QW	Bistable of String B Fails to dtct High Flux	5.0E-7/h [8]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10
RBI2B3QW	Bistable of String A Fails to dtct Low Voltage	5.0E-7/h [8]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10
RBI2B4QW	Bistable of String B Fails to dtct Low Voltage	5.0E-7/h [8]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10
RBI2C0QU	CCF - All Bistables of Channel C Fail	1.0E-7/h [7]	10						
RBI2C1QW	Bistable of String A Fails to dtct High Flux	5.0E-7/h [8]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10
RBI2C2QW	Bistable of String B Fails to dtct High Flux	5.0E-7/h [8]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10

Table 7.2. Reactor Shutdown System Data (Cont'd)

PAGE 2

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
RBI2C3QW	Bistable of String A Fails to detect Low Voltage	5.0E-7/h [8]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10
RBI2C4QW	Bistable of String B Fails to detect Low Voltage	5.0E-7/h [8]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10
RBI2Z1QU	CCF - All HF Bistables of All Channels Fail	4.0E-9/h [6]	10						
RBI2Z2QU	CCF - All LV Bistables of All Channels Fail	4.0E-9/h [6]	10						
RBI3C1QW	SOT Trip Unit of Strings A & B Fails	5.0E-7/h [1]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10
RBI3D1QW	SOT Trip Unit of Strings A & B Fails	5.0E-7/h [1]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10
RBI3F1QW	SOT Trip Unit of Strings A & B Fails	5.0E-7/h [1]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10
RBI3H1QW	SOT Trip Unit of Strings A & B Fails	5.0E-7/h [1]	10	1.0E-6/h [2]	10	3.0E-7/d	10	3.0E-7/h [5]	10
RBI3Z0QU	CCF - All Trip Units in All SOT Channels Fail	1.0E-8/h [9]	10						
RBS004QP	REACTOR SHUTDOWN SYSTEM FAILURE	1.0E-3/d	10						
RBS007QP	REACTOR SHUTDOWN SYSTEM FAILURE	1.0E-4/d	10						
RCA2A0RW	Cable and Connector of Channel A Fail	4.0E-7/h [8]	10	1.0E-7/h	10	3.0E-7/h	10	1.0E-7/h [10]	10
RCA2B0RW	Cable and Connector of Channel B Fail	4.0E-7/h [8]	10	1.0E-7/h	10	3.0E-7/h	10	1.0E-7/h [10]	10
RCA2C0RW	Cable and Connector of Channel C Fail	4.0E-7/h [8]	10	1.0E-7/h	10	3.0E-7/h	10	1.0E-7/h [10]	10
RCA2Z0RU	CCF - All Cables and Connectors Fail	4.0E-8/h [4]	10						
RCZ1A0QU	CCF - Contacts of both Strings Fail Closed	6.0E-5/d [7]	10						
RCZ1A1QP	Contact 3,4 or 5 in String A Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ1A2QP	Contact 3,4 or 5 in String B Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ1B0QU	CCF - Contacts of both Strings Fail Closed	6.0E-5/d [7]	10						
RCZ1B1QP	Contact 3,4 or 5 in String A Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ1B2QP	Contact 3,4 or 5 in String B Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ1C0QU	CCF - Contacts of both Strings Fail Closed	6.0E-5/d [7]	10						
RCZ1C1QP	Contact 3,4 or 5 in String A Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10

Table 7.2. Reactor Shutdown System Data (Cont'd)

PAGE 3

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
RCZ1C2QP	Contact 3,4 or 5 in String B Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ1D0QU	CCF - Contacts of both Strings Fail Closed	6.0E-5/d [7]	10						
RCZ1D1QP	Contact 3,4 or 5 in String A Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ1D2QP	Contact 3,4 or 5 in String B Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ1Z0QU	CCF - All Trip Contacts of both Strings Fail Closed	2.4E-6/d [6]	10						
RCZ2A0QU	CCF - Contacts of both Strings Fail Closed	6.0E-5/d [7]	10						
RCZ2A1QP	Contact 7 or 8 in String A Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ2A2QP	Contact 7 or 8 in String A Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ2A3QP	Contacts of K3 Relay Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ2A4QP	Contacts of K4 Relay Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ2A5QP	Contacts of K5 Relay Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ2A6QP	Contacts of K6 Relay Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ2A9QU	CCF - Contacts of All K-Relays Fail Closed	6.0E-5/d [7]	10						
RCZ2B0QU	CCF - Contacts of both Strings Fail Closed	6.0E-5/d [7]	10						
RCZ2B1QP	Contact 7 or 8 in String A Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ2B2QP	Contact 7 or 8 in String A Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ2B3QP	Contacts of K3 Relay Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ2B4QP	Contacts of K4 Relay Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ2B5QP	Contacts of K5 Relay Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ2B6QP	Contacts of K6 Relay Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ2B9QU	CCF - Contacts of All K-Relays Fail Closed	6.0E-5/d [7]	10						
RCZ2C0QU	CCF - Contacts of both Strings Fail Closed	6.0E-5/d [7]	10						

Table 7.2. Reactor Shutdown System Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
RCZ2C1QP	Contact 7 or 8 in String A Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ2C2QP	Contact 7 or 8 in String A Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ2C3QP	Contacts of K3 Relay Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ2C4QP	Contacts of K4 Relay Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ2C5QP	Contacts of K5 Relay Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ2C6QP	Contacts of K6 Relay Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ2C9QU	CCF - Contacts of All K-Relays Fail Closed	6.0E-5/d [7]	10						
RCZ2Z0QU	CCF - All Trip Contacts of both Strings Fail Closed	2.4E-6/d [6]	10						
RCZ2Z8QU	CCF - All Contacts of LV K-Relays Fail Closed	2.4E-6/d [6]	10						
RCZ2Z9QU	CCF - All Contacts of HF K-Relays Fail Closed	2.4E-6/d [6]	10						
RCZ3C1QP	Contact 7 or 8 in String A Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ3C2QP	Contact 7 or 8 in String B Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ3D1QP	Contact 7 or 8 in String A Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ3D2QP	Contact 7 or 8 in String B Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ3F1QP	Contact 7 or 8 in String A Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ3F2QP	Contact 7 or 8 in String B Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ3H1QP	Contact 7 or 8 in String A Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ3H2QP	Contact 7 or 8 in String B Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ401QP	Relay UV Contact 1 in String A Fails Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZ402QU	CCF - UV Trip Relay Contacts Fail Closed	3.0E-5/d [4]	10						
RCZC00QU	CCF - Contacts of All CP-Relays Fail Closed	2.4E-6/d [6]	10						

Table 7.2. Reactor Shutdown System Data (Cont'd)

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
RCZC01QU	CCF - Contacts of All String A Relays Fail Closed	9.0E-5/d [11]	10						
RCZC02QU	CCF - Contacts of All String B Relays Fail Closed	9.0E-5/d [11]	10						
RCZC81QP	Contacts Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZC82QP	Contacts Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZCC1QP	Contacts Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZCP2QP	Contacts Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZCPAQP	Contacts Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZCPYQP	Contacts Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZSROQP	Contacts Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RCZSRXQP	Contacts Fail Closed	3.0E-4/d [10]	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d [10]	10
RDC1C1AW	Transmitter DC Power Supply Fails	3.0E-6/h [3]	10					3.0E-6/h [3]	10
RDC1D1RW	Transmitter DC Power Supply Fails	3.0E-6/h [3]	10					3.0E-6/h [3]	10
RDC1D2QW	Bistable DC Power Supply Fails	3.0E-6/h [3]	10					3.0E-6/h [3]	10
RDC2A1QW	+15 V DC Power Supply Fails Low	3.0E-6/h [8]	10					3.0E-6/h [3]	10
RDC2A2QW	High Volt DC Power Supply Fails Low	3.0E-6/h [8]	10					3.0E-6/h [3]	10
RDC2B1QW	+15 V DC Power Supply Fails Low	3.0E-6/h [8]	10					3.0E-6/h [3]	10
RDC2B2QW	High Volt DC Power Supply Fails Low	3.0E-6/h [8]	10					3.0E-6/h [3]	10
RDC2C1QW	+15 V DC Power Supply Fails Low	3.0E-6/h [8]	10					3.0E-6/h [3]	10
RDC2C2QW	High Volt DC Power Supply Fails Low	3.0E-6/h [8]	10					3.0E-6/h [3]	10
RDC2Z1QU	ALL +15 V DC Power Supplies Fail Low	3.0E-7/h [4]	10						
RDC2Z2QU	CCF - All High Volt DC Power Supplies Fail Low	3.0E-7/h [4]	10						
RFM1A0RW	Flow Sensor Fails High or Fails to Respond	3.0E-6/h [12]	10	3.0E-5/h [13]	10	3.0E-6/h [13]	10	1.0E-6/h [10]	10
RFM1B0RW	Flow Sensor Fails High or Fails to Respond	3.0E-6/h [12]	10	3.0E-5/h [13]	10	3.0E-6/h [13]	10	1.0E-6/h [10]	10
RFM1C0RW	Flow Sensor Fails High or Fails to Respond	3.0E-6/h [12]	10	3.0E-5/h [13]	10	3.0E-6/h [13]	10	1.0E-6/h [10]	10
RFM1D0RW	Flow Sensor Fails High or Fails to Respond	3.0E-6/h [12]	10	3.0E-5/h [13]	10	3.0E-6/h [13]	10	1.0E-6/h [10]	10

Table 7.2. Reactor Shutdown System Data (Cont'd)

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
RFM1Z1RU	CCF - Electro-Magnetic Flow Sensors Fail	3.0E-7/h [4]	10						
RHU001CL	Opertr Error - Failure to Push SCRAM Buttons	5.0E-2/d [14]	10						
RHU002CL	Opertr Error - Failure to Push SCRAM Buttons	1.0E-1/d [14]	9						
RHU006CL	Operator Failure to Push SCRAM Buttons w/LOHS Conditions	1.0E-6/d [15]	10						
RHU00ZCL	Opertr Error - Failure to Push 3MPB SCRAM Button	7.5E-3/d [16]	10						
RHU201QL	CCF Human Error - Failure to Set HF Trip Points Correct	1.5E-6/d [16]	10						
RHU202QL	CCF Human Error - Failure to Set LV Trip Points Correct	1.5E-6/d [16]	10						
RHU302QL	CCF - Human Error - SOT Trip Points Set Too High	2.5E-2/d [16]	10						
RMO001RR	Primary Pump 1 Motor Fails to Run	2.0E-5/h	3	1.0E-5/h	10			3.0E-5/h	3
RMO002RR	Primary Pump 2 Motor Fails to Run	2.0E-5/h	3	1.0E-5/h	10			3.0E-5/h	3
RND2A0RW	Grded Fission Chamber in J1 Thimble Fails	3.0E-6/h [1]	10					6.0E-6/h [3]	3
RND2B0RW	Grded Fission Chamber in J3 Thimble Fails	3.0E-6/h [1]	10					6.0E-6/h [3]	3
RND2C0RW	Grded Fission Chamber in J4 Thimble Fails	3.0E-6/h [1]	10					6.0E-6/h [3]	3
RND2Z0RU	All Grded Fission Chambers Fail	3.0E-7/h [4]	10						
ROO1AA00	Flow Sensor Can Not Detect LOF Condition (Location)	0.0	1						
ROO1AB00	Flow Sensor Can Detect LOF Condition (Location)	1.0	1						
ROO99A00	Reactor Operation Permitted	1.0	1						
ROO99B00	Reactor Operation Not Permitted	0.0	1						
RRE1A0QU	CCF - Trip Relays of both Strings Fail to Move	4.0E-8/h [7]	10						
RRE1A1QP	String A Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE1A2QP	String B Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE1B0QU	CCF - Trip Relays of both Strings Fail to Move	4.0E-8/h [7]	10						

Table 7.2. Reactor Shutdown System Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
RRE1B1QP	String A Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE1B2QP	String B Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE1C0QU	CCF - Trip Relays of both Strings Fail to Move	4.0E-8/h [7]	10						
RRE1C1QP	String A Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE1C2QP	String B Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE1D0QU	CCF - Trip Relays of both Strings Fail to Move	4.0E-8/h [7]	10						
RRE1D1QP	String A Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE1D2QP	String B Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE1Z0QU	CCF - All Trip Relays Fail to Move	1.6E-9/h [6]	10						
RRE2A0QU	CCF - Trip Relays of both Strings Fail to Move	4.0E-8/h [7]	10						
RRE2A1QP	String A Relay Fails to Move to Open Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE2A2QP	String B Relay Fails to Move to Open Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE2A3QP	K3 Relay Fails to Move to Open Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE2A4QP	K4 Relay Fails to Move to Open Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE2A5QP	K5 Relay Fails to Move to Open Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE2A6QP	K6 Relay Fails to Move to Open Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE2A9QU	CCF - All K-Relays of Channel A Fail to Move	4.0E-8/h [7]	10						
RRE2B0QU	CCF - Trip Relays of both Strings Fail to Move	4.0E-8/h [7]	10						
RRE2B1QP	String A Relay Fails to Move to Open Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE2B2QP	String B Relay Fails to Move to Open Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE2B3QP	K3 Relay Fails to Move to Open Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE2B4QP	K4 Relay Fails to Move to Open Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	

Table 7.2. Reactor Shutdown System Data (Cont'd)

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
RRE2B5QP	K5 Relay Fails to Move to Open Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE2B6QP	K6 Relay Fails to Move to Open Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE2B9QU	CCF - All K-Relays of Channel B Fail to Move	4.0E-8/h [7]	10						
RRE2C0QU	CCF - Trip Relays of both Strings Fail to Move	4.0E-8/h [7]	10						
RRE2C1QP	String A Relay Fails to Move to Open Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE2C2QP	String B Relay Fails to Move to Open Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE2C3QP	K3 Relay Fails to Move to Open Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE2C4QP	K4 Relay Fails to Move to Open Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE2C5QP	K5 Relay Fails to Move to Open Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE2C6QP	K6 Relay Fails to Move to Open Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE2C9QU	CCF - All K-Relays of Channel C Fail to Move	4.0E-8/h [7]	10						
RRE2Z0QU	CCF - All Trip Relays of both Strings Fail to Move	1.6E-9/h [6]	10						
RRE2Z8QU	CCF - LV K-Relays of All Channels Fail to Move	1.6E-9/h [6]	10						
RRE2Z9QU	CCF - HF K-Relays of All Channels Fail to Move	1.6E-9/h [6]	10						
RRE3C1QP	String A Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE3C2QP	String B Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE3D1QP	String A Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE3D2QP	String B Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE3F1QP	String A Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE3F2QP	String B Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE3H1QP	String A Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE3H2QP	String B Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	

Table 7.2. Reactor Shutdown System Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
RRE401QP	Relay UV (Device 21-RRI) Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRE402QU	CCF - UV Trip Relays (Devices 27-A & -C)	2.0E-8/h [4]	10						
RREC00QU	CCF - All CP-Relays on Strings A & B Fail	1.6E-9/h [6]	10						
RREC01QU	CCF - All Relays on String A Fail	6.0E-8/h [11]	10						
RREC02QU	CCF - All Relays on String B Fail	6.0E-8/h [11]	10						
RRECB1QP	CPB-1 Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRECB2QP	CPB-2 Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRECC1QP	CPC-1 Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRECP2QP	CP-2 Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRECPAQP	CPA Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRECPYQP	CPY Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRESROQP	SRO Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRESRXQP	SRO-X Relay Fails to Change Position	2.0E-7/h [8]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h [5]	
RRT300AU	CCF - All Ref Junctions Fail High	4.0E-8/h [9]	10						
RRT3C0AW	T/C 503C Ref Junction Fails High	2.0E-6/h [14]	10					3.0E-6/h [10]	10
RRT3D0AW	T/C 503D Ref Junction Fails High	2.0E-6/h [14]	10					3.0E-6/h [10]	10
RRT3F0AW	T/C 503F Ref Junction Fails High	2.0E-6/h [14]	10					3.0E-6/h [10]	10
RRT3H0AW	T/C 503H Ref Junction Fails High	2.0E-6/h [14]	10					3.0E-6/h [10]	10
RTC300RU	CCF - All T/C's Fail Low	2.0E-8/h [9]	10						
RTC3C0RW	T/C 503C Fails Low - Short/Open	1.0E-6/h [10]	10	1.0E-6/h	10			1.0E-6/h [10]	10
RTC3D0RW	T/C 503D Fails Low - Short/Open	1.0E-6/h [10]	10	1.0E-6/h	10			1.0E-6/h [10]	10
RTC3F0RW	T/C 503F Fails Low - Short/Open	1.0E-6/h [10]	10	1.0E-6/h	10			1.0E-6/h [10]	10
RTC3H0RW	T/C 503H Fails Low - Short/Open	1.0E-6/h [10]	10	1.0E-6/h	10			1.0E-6/h [10]	10
RTS1C1AW	Pressure Transmitters Fail	2.0E-5/h [14]	10					3.0E-5/h [3]	12
RTS1D1RW	Pressure Transmitters Fail	2.0E-5/h [14]	10					3.0E-5/h [3]	12
RV11A1QW	MV/I Converter Fails	1.0E-6/h [8]	10	1.0E-6/h [2]	10	3.0E-6/h [2]	10	1.0E-6/h [3]	10

Table 7.2. Reactor Shutdown System Data (Cont'd)

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
RVI1A2QW	DMV/I Converter Fails	1.0E-6/h [8]	10	1.0E-6/h [2]	10	3.0E-6/h [2]	10	1.0E-6/h [3]	10
RVI1B1QW	MV/I Converter Fails	1.0E-6/h [8]	10	1.0E-6/h [2]	10	3.0E-6/h [2]	10	1.0E-6/h [3]	10
RVI1C1QW	MV/I Converter Fails	1.0E-6/h [8]	10	1.0E-6/h [2]	10	3.0E-6/h [2]	10	1.0E-6/h [3]	10
RVI1Z0QU	CCF - All MV/I Converters Fail	8.0E-9/h [6]	10						
RVI3C0QW	MV/I Converter Fails	1.0E-6/h [8]	10	1.0E-6/h [2]	10	3.0E-6/h [2]	10	1.0E-6/h [3]	10
RVI3D0QW	MV/I Converter Fails	1.0E-6/h [8]	10	1.0E-6/h [2]	10	3.0E-6/h [2]	10	1.0E-6/h [3]	10
RVI3F0QW	MV/I Converter Fails	1.0E-6/h [8]	10	1.0E-6/h [2]	10	3.0E-6/h [2]	10	1.0E-6/h [3]	10
RVI3H0QW	MV/I Converter Fails	1.0E-6/h [8]	10	1.0E-6/h [2]	10	3.0E-6/h [2]	10	1.0E-6/h [3]	10
RVI3Z0QU	CCF - All MV/I Converters in SOT Channels Fail	2.0E-8/h [9]	10						

- Notes:
1. Based on Plant-Specific Data
 2. Low power solid state device value
 3. IEEE Standard 500, 1984
 4. 0.1 x Independent Failure Value
 5. Taken from Table 7-3, SAND-87-7138
 6. 0.008 x Independent Failure Value
 7. 0.2 x Independent Failure Value
 8. Based on Plant-Specific Data and Engineering Judgement
 9. 0.02 x Independent Failure Value
 10. Recommended value tables, EGG-SSRE-8875, Feb. 1990
 11. 0.3 x Independent Failure Value
 12. CREDO Data for EBR-II Flow Sensors, 7-1-64 to 12-21-89
 13. Instrumentation in general
 14. Based on Engineering Judgement
 15. Directed by Procedures; Long Time Available
 16. Estimate provided by D. Sherman, SAIC, Idaho Falls

Table 7.3. Reactor Control System Data

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
CARCRPRD	LIFTING PLATFORM RAISED 1/4 IN	6.0E-4/d [1]	5						
CCD10GRU	CCF - Bowing and Binding / Inside Tank	1.3E-6/d [2]	20						
CCD20WRU	CCF - Bowing and Binding/Above Rotating Plug	1.0E-7/d [3]	20						
CCD30IRU	CCF - Forces due to Foreign Materials in Sodium	3.0E-7/d [3]	20						
CCL00PRU	CCF - Magnetic Clutches Fail to Release CRs	1.6E-6/d [4]	20						
CHU0P1CL	OPERATOR DECIDES TO COMPENSATE WITH 2nd ROD	1.0E-1/d [5]	5						
CRECSUQP	CU OR SU CONTACT FAILS TO REOPEN	6.6E-7/d [6]	8						
CRECSUQQ	CU OR SU RELAY CLOSES SPONTANEOUSLY	1.4E-6/h [6]	8						
CREDUMQI	RELAY EVENT MUST OCCUR WITHIN 0.1 h	1.4E-5/d [7]	1						
CREKD7QP	K7 RELAY FAILS TO REOPEN	3.0E-6/d [8]	2						
CREKD7QQ	K7 RELAY CLOSES SPONTANEOUSLY	5.0E-7/h [9]	10						
CSR10GRU	CCF - Forces due to Bowing and Binding	1.0E-7/d [3]	20						
CSR30IRU	CCF - Forces due to Foreign Materials in Sodium	1.0E-7/d [3]	20						
CSTMC1QD	CONTROLLER SENDS UP-VOLTAGE SPONTANEOUSLY	4.0E-7/h [10]	4						
CSTMC2QD	CONTROLLER FAILS AFTER SWITCH OR RELAY	4.0E-7/h [10]	4						
CSW100CP	1MPB Mech Parts Fail	2.0E-6/d [5]	10						
CSW10ACP	1MPB Switch of String A Fails to Open	1.0E-5/d [9]	5	1.0E-4/d	10	3.0E-5/d	10	1.0E-5/d [9]	5
CSW10BCP	1MPB Switch of String B Fails to Open	1.0E-5/d [9]	5	1.0E-4/d	10	3.0E-5/d	10	1.0E-5/d [9]	5
CSWC0NCZ	CONSOLE ACCESS RESTRICTION VIOLATED	5.0E-2/d [11]	5						
CSWDUMCI	SWITCH EVENT MUST OCCUR WITHIN 0.1 h	1.4E-5/d [7]	1						
CSWMS1CL	OPERATORS FAIL TO FREE MSW FRM UNAUTHORIZED PERSON	5.0E-2/d [11]	2						
CSWMS1CQ	MSW-1 CLOSES SPONTANEOUSLY	7.0E-8/h [8]	2						
CSWMS1CZ	MSW SWITCH HELD UP BY UNAUTHORIZED PERSON	2.1E-1/d [6]	2						

Table 7.3. Reactor Control System Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
CSWMS2CL	OPERATORS FAIL TO RELEASE MSW WITHIN 1 MIN AFTER ALARM	1.0E-3/d [11]	5						
CSWMS2CP	INTERLOCK BETWEEN MSWs FAILS	1.0E-4/d [5]	10						
CSWMS2CQ	MSW-2 CLOSES SPONTANEOUSLY	7.0E-8/h [8]	2						
CSWMS2CZ	OPERATOR HOLDS UP MSW UNTIL ALARMS	2.0E-2/d [6]	8						
CSWMS3CZ	OP. CLOSES RELAY, FAILS TO COMPENSATE W/CONTROLLING ROD	1.1E-1/d [11]	3						
CSWMS4CZ	OPERATOR HOLDS MSW UP INSTEAD OF DOWN	2.5E-2/d [11]	2						
CSWMS5CL	MSSW NOT SET TO NEUTRAL POSITION	1.7E-4/h [11]	3						
CSWMSWCP	MSW STAYS UP WHEN RELEASED	1.4E-6/d [6]	3						
CSWMSWCZ	OPERATOR FAILS TO FREE STUCK SWITCH	3.0E-1/d [5]	2						
CSWPSBCL	OPERATOR FAILS TO PUSH PSB2	1.7E-4/h [11]	3						
CSWPSBCP	PSB2 SWITCH FAILS TO OPEN	1.0E-5/d [9]	5						
CSWPSBCQ	PSB2 SWITCH CLOSES SPONTANEOUSLY	8.5E-8/h [8]	10						
CSWSS2CD	2-ROD SHORT OF SELECTOR SWITCH	1.5E-6/h [8]	1						
CSWSS3CD	MANY-ROD SHORT OF SELECTOR SWITCH	1.8E-8/h [5]	10						

- Notes:
1. From Fault Tree Analysis for the CRD Platform
 2. Incl. Prob. of 1.2E-6/d due to RV Cover Tilting
 3. Based on combined CCF Prob of 7.0E-7/d for all CRs
 4. 0.008 x Independent Failure Value
 5. Based on Engineering Judgement
 6. Based on Plant Data
 7. Based on a 6-minute Window
 8. IEEE Standard 500, 1984
 9. Recommended value tables, EGG-SSRE-8875, Feb. 1990
 10. IEEE Standard 500 plus Engineering Judgement
 11. Based on NUREG/CR-1278 and Engineering Judgement

Table 7.4. Primary Pump and Primary Pump M/G Set Data

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
PCB001PO	MG Set 1 Output Breaker Fails Open	3.0E-7/h	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
PCB002PO	MG Set 2 Output Breaker Fails Open	3.0E-7/h	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
PCB012PU	MG Set 1 and 2 Output Breaker Fails Open, Common Cause	3.0E-8/h	10						
PCL001PW	MG Set 1 Clutch Failure	5.0E-5/h	30	3.0E-4/d	3			3.0E-4/d	3
PCL002PW	MG Set 2 Clutch Failure	5.0E-5/h	30	3.0E-4/d	3			3.0E-4/d	3
PCZCP6QB	Contacts 6 of Relay CPC-2 Fail Open	1.0E-7/h	10	1.0E-7/h	10				
PCZCP7QB	Contacts 7 of Relay CPC-2 Fail Open	1.0E-7/h	10	1.0E-7/h	10				
PCZPB1PB	Contacts 1 of Relay PBS Fail Open	1.0E-7/h	10	1.0E-7/h	10				
PCZPB8PB	Contacts 8 of Relay PBS Fail Open	1.0E-7/h	10	1.0E-7/h	10				
PCZX15PB	Contacts 5 of Relay EX1 Fail Open	1.0E-7/h	10	1.0E-7/h [1]	10				
PCZX24PB	Contacts 4 of Relay EX2 Fail Open	1.0E-7/h	10	1.0E-7/h [1]	10				
PCZX25PB	Contacts 5 of Relay EX2 Fail Open	1.0E-7/h	10	1.0E-7/h [1]	10				
PDC001PB	HP Power Supply No. 1 Failure	3.0E-6/h	10	1.0E-6/h [2]	10	3.0E-6/h [2]	10	3.0E-6/h [2]	10
PDC002PB	HP Power Supply No. 2 Failure	3.0E-6/h	10	1.0E-6/h [2]	10	3.0E-6/h [2]	10	3.0E-6/h [2]	10
PDC006PB	Exciter No. 1 Power Supply Failure	3.0E-6/h	10	1.0E-6/h [2]	10	3.0E-6/h [2]	10	3.0E-6/h [2]	10
PDC007PB	Exciter No. 2 Power Supply Failure	3.0E-6/h	10	1.0E-6/h [2]	10	3.0E-6/h [2]	10	3.0E-6/h [2]	10
PDC012PU	HP Power Supply No. 1 and 2 Fails, Common Cause	3.0E-7/h	10						
PEX001PW	MG Set 1 Exciter Failure	3.0E-6/h	10	1.0E-6/h [2]	10	3.0E-6/h [2]	10	3.0E-6/h [2]	10
PEX002PW	MG Set 2 Exciter Failure	3.0E-6/h	10	1.0E-6/h [2]	10	3.0E-6/h [2]	10	3.0E-6/h [2]	10
PGE001PW	MG Set 1 Generator Failure	3.0E-6/h	10	1.0E-6/h [3]	10	3.0E-6/h [3]	10		
PGE002PW	MG Set 2 Generator Failure	3.0E-6/h	10	1.0E-6/h [3]	10	3.0E-6/h [3]	10		
PKS001PB	Primary Pump Key Switch Fails Open	1.0E-7/h	10	1.0E-7/h	10				
PMO001PW	MG Set 1 Motor Failure	3.0E-5/h	3	1.0E-5/h	10			3.0E-5/h	3
PMO002PW	MG Set 2 Motor Failure	3.0E-5/h	3	1.0E-5/h	10			3.0E-5/h	3
PPM001RG	Primary Pump 1 Shaft Binding	2.9E-5/h	2						
PPM001RR	Primary Pump 1 Shaft Broken or Seized	1.9E-6/h	8						

Table 7.4. Primary Pump and Primary Pump M/G Set Data (Cont'd)

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
PPM002RG	Primary Pump 2 Shaft Binding	2.9E-5/h	2						
PPM002RR	Primary Pump 2 Shaft Broken or Seized	1.9E-6/h	8						
PPM012RU	Primary Pump 1 and 2 Shaft Binding, Common Cause	2.9E-6/h	10						
PPM021RU	Primary Pump 2 Shaft Binding, Common Cause	8.0E-7/h	30						
PREEX1PB	Relay EX1 Coil Fails Open	5.0E-7/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
PREEX2PB	Relay EX2 Coil Fails Open	5.0E-7/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
PREP2AQB	Relay CPC-2 Coil Fails Open	5.0E-7/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
PREPBSPB	Relay PBS Coil Fails Open	5.0E-7/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
PSC001PB	MG Set Controller 1 Overspeed Cutoff	3.0E-6/h	10	1.0E-6/h [4]	10	3.0E-6/h [4]	10	2.0E-6/h [4]	10
PSC002PB	MG Set Controller 2 Overspeed Cutoff	3.0E-6/h	10	1.0E-6/h [4]	10	3.0E-6/h [4]	10	2.0E-6/h [4]	10
PSC01BPW	MG Set 1 Controller Faulty Demand Signal	3.0E-6/h	10	1.0E-6/h [4]	10	3.0E-6/h [4]	10	2.0E-6/h [4]	10
PSC01CPW	DCS No. 1 Fails with Low Demand Signal	3.0E-6/h	10	1.0E-6/h [4]	10	3.0E-6/h [4]	10	2.0E-6/h [4]	10
PSC02BPW	MG Set 2 Controller Faulty Demand Signal	3.0E-6/h	10	1.0E-6/h [4]	10	3.0E-6/h [4]	10	2.0E-6/h [4]	10
PSC02CPW	DCS No. 2 Fails with Low Demand Signal	3.0E-6/h	10	1.0E-6/h [4]	10	3.0E-6/h [4]	10	2.0E-6/h [4]	10
PST001PW	No. 1 Feedback Signal Circuit Fails With High Signal	3.0E-6/h	10	1.0E-6/h [4]	10	3.0E-6/h [4]	10	2.0E-6/h [4]	10
PST002PW	No. 2 Feedback Signal Circuit Fails With High Signal	3.0E-6/h	10	1.0E-6/h [4]	10	3.0E-6/h [4]	10	2.0E-6/h [4]	10
PSW012CU	Control Room STOP Switch Short, Common Cause Loss of CPS	1.0E-8/h	10						
PSW012PU	Primary Panel STOP Switch Short, Common Cause Loss of CPS	1.0E-8/h	10						
PSW01APB	Primary Panel STOP Switch Fails Open, Pump 1	1.0E-7/h	10	1.0E-7/h	10				
PSW01APD	Primary Panel STOP Switch Short, Loss of CPS, Pump 1	1.0E-7/h	10	1.0E-7/h	10				
PSW01BPB	Primary Panel STOP Switch Fails Open, Pump 2	1.0E-7/h	10	1.0E-7/h	10				
PSW01BPD	Primary Panel STOP Switch Short, Loss of CPS, Pump 2	1.0E-7/h	10	1.0E-7/h	10				
PSW02ACB	Control Room STOP Switch Fails Open, Pump 1	1.0E-7/h	10	1.0E-7/h	10				

Table 7.4 Primary Pump and Primary Pump M/G Set Data (Cont'd)

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
PSW02ACD	Control Room STOP Switch Short, Loss of CPS, Pump 1	1.0E-7/h	10	1.0E-7/h	10				
PSW02BCB	Control Room STOP Switch Fails Open, Pump 2	1.0E-7/h	10	1.0E-7/h	10				
PSW02BCD	Control Room STOP Switch Short, Loss of CPS, Pump 2	1.0E-7/h	10	1.0E-7/h	10				
PWR012PU	Pumps 1 & 2 START/STOP Circuit Wiring Failure Common Cause	3.0E-7/h [5]	10						
PWR01APB	Primary Pump 1 START/STOP Circuit Wiring Fails Open	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
PWR01BPB	Primary Pump 2 START/STOP Circuit Wiring Fails Open	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
PWR022PU	Wiring between STOP Switches Fails Open, Pumps 1 & 2 Common	3.0E-7/h [5]	10						
PWR022YB	CPS Feeder to HP Power Supply No. 1 Fails Open	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
PWR022YC	CPS Feeder to HP Power Supply No. 1 Short Circuit	3.0E-7/h	10	1.0E-7/h [6]	10	3.0E-7/h [6]	10	3.0E-7/h [6]	10
PWR02APB	Wiring between STOP Switches Fails Open, Pump 1	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
PWR02BPB	Wiring between STOP Switches Fails Open, Pump 2	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
PWR032QU	Wiring to/from CPC-2 Relay Fails Open, Pumps 1 & 2 Common	6.0E-7/h [7]	10						
PWR03AQB	Wiring to/from CPC-2 Relay Fails Open, Pump 1	6.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
PWR03BQB	Wiring to/from CPC-2 Relay Fails Open, Pump 2	6.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
PWR042PB	Exciter No. 1 Wiring to Generator Fails Open	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
PWR042PC	Exciter No. 1 Wiring to Generator Short Circuit	3.0E-7/h	10	1.0E-7/h [6]	10	3.0E-7/h [6]	10	3.0E-7/h [6]	10
PWR044PB	CPS Feeder to Exciter No. 1 PS Fails Open	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
PWR044PC	CPS Feeder to Exciter No. 1 PS Short Circuit	3.0E-7/h	10	1.0E-7/h [6]	10	3.0E-7/h [6]	10	3.0E-7/h [6]	10
PWR051TB	MG Set 1 Wiring to Pump 1 Fails Open	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
PWR051TC	MG Set 1 Wiring to Pump 1 Short Circuit	3.0E-7/h	10	1.0E-7/h [6]	10	3.0E-7/h [6]	10	3.0E-7/h [6]	10
PWR122YB	CPS Feeder to HP Power Supply No. 2 Fails Open	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3

Table 7.4 Primary Pump and Primary Pump M/G Set Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
PWR122YC	CPS Feeder to HP Power Supply No. 2 Short Circuit	3.0E-7/h	10	1.0E-7/h [6]	10	3.0E-7/h [6]	10	3.0E-7/h [6]	10
PWR142PB	Exciter No. 2 Wiring to Generator Fails Open	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
PWR142PC	Exciter No. 2 Wiring to Generator Short Circuit	3.0E-7/h	10	1.0E-7/h [6]	10	3.0E-7/h [6]	10	3.0E-7/h [6]	10
PWR144PB	CPS Feeder to Exciter No. 2 PS Fails Open	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
PWR144PC	CPS Feeder to Exciter No. 2 PS Short Circuit	3.0E-7/h	10	1.0E-7/h [6]	10	3.0E-7/h [6]	10	3.0E-7/h [6]	10
PWR151TB	MG Set 2 Wiring to Pump 2 Fails Open	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
PWR151TC	MG Set 2 Wiring to Pump 2 Short Circuit	3.0E-7/h	10	1.0E-7/h [6]	10	3.0E-7/h [6]	10	3.0E-7/h [6]	10
PWR201YB	2400 V Feeder to MG Set 1 Fails Open	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
PWR201YC	2400 V Feeder to MG Set 1 Short Circuit	3.0E-7/h	10	1.0E-7/h [6]	10	3.0E-7/h [6]	10	3.0E-7/h [6]	10
PWR202YB	2400 V Feeder to MG Set 2 Fails Open	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
PWR202YC	2400 V Feeder to MG Set 2 Short Circuit	3.0E-7/h	10	1.0E-7/h [6]	10	3.0E-7/h [6]	10	3.0E-7/h [6]	10

- Notes:
1. Taken from CRBRP-4 column
 2. High power solid state device value
 3. DC motor-generator failure value
 4. Low power solid state device value
 5. Value for 100 ft open circuit with beta factor of 0.1
 6. Short to ground value
 7. Value for 200 ft open circuit with beta factor of 0.1

Table 7.5. Shutdown Cooler Data

PAGE 1

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
DAM4AHAW	AV BULK NA TEMP RONA N X81MV/I FAILS HIGH ELEC P1-TMV/I-504A	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
DAM4ALAW	AV BULK NA TEMP RONA N X81MV/I FAILS LOW ELEC P1-TMV/I-504A	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
DAM4BHQW	EXPN RANGE TEMP RONA N X81MV/I FAILS HIGH ELEC P1-TMV/I-504B	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
DAM4BLQW	EXPN RANGE TEMP RONA N X81MV/I FAILS LOW ELEC P1-TMV/I-504B	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
DAN501AW	T/C REFERENCE JUNCT ION HIGH ANNUN FAIL S FOR R1-CRJ-50-1	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
DAN502AW	T/C REFERENCE JUNCT ION HIGH ANNUN FAIL S FOR R1-CRJ-50-2	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
DAN504AW	T/C REFERENCE JUNCT ION HIGH ANNUN FAIL S FOR R1-CRJ-50-4	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
DAN505AW	T/C REFERENCE JUNCT ION HIGH ANNUN FAIL S FOR R1-CRJ-50-5	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
DAN751AW	T/C REFERENCE JUNCT ION HIGH ANNUN FAIL S FOR R1-CRJ-75-1	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
DAN82LAW	BULK SODIUM LOW TEMPERATURE ANNUN FAILS	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
DARX82QB	RONAN X82 ALARM CIRCUIT FAILS HIGH P1-TA-504A	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
DARX82QO	RONAN X82 ALARM RELAY (COIL) FAILS OPEN P1-TA-504A	3.0E-7/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
DARX82QP	RONAN X82 ALARM RELAY FAILS TO OPEN P1-TA-504A	3.0E-4/d	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d	10
DARX82QW	RONAN X82 ALARM CIRCUIT FAILS LOW P1-TA-504A	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
DAT01LOO	COOLER 1 LOWER ACTUA TOR OR LINKAGE FAILS TO REMAIN CLOSED	3.0E-7/h	10	1.0E-7/h	3				
DAT01LOP	COOLER 1 LOWER ACTUA TOR OR LINKAGE FA ILS TO OPEN DAMPER	3.0E-3/d	5	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3
DAT01UOO	COOLER 1 UPPER ACTUA TOR OR LINKAGE FAILS TO REMAIN CLOSED	3.0E-7/h	10	1.0E-7/h	3				
DAT01UOP	COOLER 1 UPPER ACTUA TOR OR LINKAGE FA ILS TO OPEN DAMPER	3.0E-3/d	5	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3
DAT02LOO	COOLER 2 LOWER ACTUA TOR OR LINKAGE FAILS TO REMAIN CLOSED	3.0E-7/h	10	1.0E-7/h	3				
DAT02LOP	COOLER 2 LOWER ACTUA TOR OR LINKAGE FA ILS TO OPEN DAMPER	3.0E-3/d	5	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3

Table 7.5. Shutdown Cooler Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
DAT02UOO	COOLER 2 UPPER ACTUATOR OR LINKAGE FAILS TO REMAIN CLOSED	3.0E-7/h	10	1.0E-7/h	3				
DAT02UOP	COOLER 2 UPPER ACTUATOR OR LINKAGE FAILS TO OPEN DAMPER	3.0E-3/d	5	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3
DCR001OE	COOLER #1 NAK BLOCKAGE	1.0E-7/h	10	1.0E-9/h				8.0E-9/h	30
DCR001OF	COOLER #1 NAK LEAKAGE	3.0E-7/h	10	1.0E-9/h				8.0E-9/h	30
DCR002OE	COOLER #2 NAK BLOCKAGE	1.0E-7/h	10						
DCR002OF	COOLER #2 NAK LEAKAGE	3.0E-7/h	10	1.0E-9/h				8.0E-9/h	30
DHR501AW	T/C REFERENCE JUNCTION OVEN FAILS HIGH R1-CRJ-50-1	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
DHR502AW	T/C REFERENCE JUNCTION OVEN FAILS HIGH R1-CRJ-50-2	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
DHR504AW	T/C REFERENCE JUNCTION OVEN FAILS HIGH R1-CRJ-50-4	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
DHR505AW	T/C REFERENCE JUNCTION OVEN FAILS HIGH R1-CRJ-50-5	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
DHR751AW	T/C REFERENCE JUNCTION OVEN FAILS HIGH R1-CRJ-75-1	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
DHU000OL	HUMAN ERROR - FAILURE TO RECOVER COOLERS	4.0E-6/d	10						
DHU154PZ	HUMAN ERROR IMPROPER VALVE OPERATION PA-15	1.0E-4/d	10						
DHU500CZ	HUMAN ERROR ANY T/C REF JUNCT >152 F ALARM IGNORED	1.0E-2/d	10						
DHU82LCZ	HUMAN ERROR BULK SODIUM LOW TEMP ALARM IGNORED	1.0E-2/d	10						
DOO11Z00	FORCE IS ENOUGH TO LIFT SODIUM	1.0	1						
DOO22Z00	HOT LEG BREACH BELOW PUMP INLET	1.0	1						
DOO33Z00	BREACH OUTSIDE REACTOR AFTER HEAT EXCHANGER	1.0	1						
DPPCDLRF	COLD LEG BREACH	3.0E-7/h [1]	30						
DPPINNRF	INNER PIPE BREACH	1.0E-1/d [2]	3						
DPPOUTRF	OUTER PIPE BREACH	1.0E-6/h [3]	10						
DPV02AOQ	AIR REGULATOR COOLER 1 FAILS TO REMAIN OPEN R4-PVC-682A	3.3E-7/h	10	1.0E-7/h	3				

Table 7.5. Shutdown Cooler Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
DPV02BOQ	AIR REGULATOR COOLE R 2 FAILS TO REMAIN OPEN R4-PVC-682B	3.3E-7/h	10	1.0E-7/h	3				
DRE53ACO	AUX ANN RELAY PT 2 (COIL) FAILS OPEN ITEM 53A-PR2A	3.0E-7/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
DRE53ACP	AUX ANN RELAY PT 2 (CONTACT) FAILS TO OPEN ITEM 53A-PR2A	3.0E-4/d	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d	10
DSV01AON	SOLENOID COOLER #1 FAILS TO CLOSE R4-VS-681A	3.0E-3/d	5	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3
DSV01ACQ	SOLENOID FAILS DOES NOT REMAIN OPEN R4-VS-681A	5.0E-8/h	10	1.0E-7/h	3				
DSV01BON	SOLENOID COOLER #2 FAILS TO CLOSE R4-VS-681B	3.0E-3/d	5	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3
DSV01BOQ	SOLENOID FAILS DOES NOT REMAIN OPEN R4-VS-681B	5.0E-8/h	10	1.0E-7/h	3				
DTC40HRB	BULK SODIUM THERMOCOUPLE FAILS OPEN RI-TC-540H	1.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
DTC40HRD	BULK SODIUM THERMOCOUPLE SHORTED RI-TC-540H	1.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
DTC40KRB	BULK SODIUM THERMOCOUPLE FAILS OPEN RI-TC-540K	1.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
DTC40KRD	BULK SODIUM THERMOCOUPLE SHORTED RI-TC-540K	1.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
DTC40MRB	BULK SODIUM THERMOCOUPLE FAILS OPEN RI-TC-540M	1.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
DTC40MRD	BULK SODIUM THERMOCOUPLE SHORTED RI-TC-540M	1.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
DTC40NRB	BULK SODIUM THERMOCOUPLE FAILS OPEN RI-TC-540N	1.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
DTC40NRD	BULK SODIUM THERMOCOUPLE SHORTED RI-TC-540N	1.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
DTC40PRB	BULK SODIUM THERMOCOUPLE FAILS OPEN RI-TC-540P	1.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
DTC40PRD	BULK SODIUM THERMOCOUPLE SHORTED RI-TC-540P	1.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
DTW001OE	COOLER #1 CHIMNEY BLOCKED	1.1E-6/h	10						
DTW002OE	COOLER #2 CHIMNEY BLOCKED	1.1E-6/h	10						

- Notes: 1. Recommended Value Tables, EGG-SSRE-8875, Feb. 1990 - 100 ft
2. Based on Engineering Judgement
3. Recommended value tables, EGG-SSRE-8875, Feb. 1990

Table 7.6. Shield Cooling System Data

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
HAV346PQ	RX BLD FREON ISOLA TION FAILS TO REMAIN OPEN P17-VC-346B	1.0E-7/h	10	1.0E-7/h	3				
HAV347PQ	RX BLD FREON ISOLA TION FAILS TO REMAIN OPEN P17-VC-347B	1.0E-7/h	10	1.0E-7/h	3				
HAV348PQ	RX BLD FREON ISOLA TION FAILS TO REMAIN OPEN P17-VC-348B	1.0E-7/h	10	1.0E-7/h	3				
HAV349PQ	RX BLD FREON ISOLA TION FAILS TO REMAIN OPEN P17-VC-349B	1.0E-7/h	10	1.0E-7/h	3				
HAV702PQ	RX BLD FREON ISOLA TION FAILS TO REMAIN OPEN P17-VC-702B	1.0E-7/h	10	1.0E-7/h	3				
HAV703PQ	RX BLD FREON ISOLA TION FAILS TO REMAIN OPEN P17-VC-703B	1.0E-7/h	10	1.0E-7/h	3				
HAV727SO	AUTO DAMPER 727 FAILS TO REMAIN CLOSED R8-DM-727	1.0E-7/h	10	1.0E-7/h	3				
HCM040SR	40 TON COMPRESSOR FAILURE TO RUN	6.0E-6/h	10						
HCM201SR	20 TON COMPRESSOR #1 FAILURE TO RUN	6.0E-6/h	10						
HCM202SR	20 TON COMPRESSOR #2 FAILURE TO RUN	6.0E-6/h	10						
HDM705SW	VORTEX DAMPER FAN 705 INTERNAL FAILURE	3.6E-8/h	8						
HDM706SP	VORTEX DAMPER ON FAN 706 DOES NOT OPEN FN-706	3.0E-3/d	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
HDM706SQ	VORTEX DAMPER FAN 706 DOES NOT REMAIN OPEN FN-706	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HDM706SW	DAMPER 706 INTERNAL FAILURE R8-DM-706	3.6E-8/h	8						
HDM722SO	DAMPER 722 FAILS TO REMAIN CLOSED R8-DM-722	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HDM722SP	DAMPER 722 MECHANICAL FAILS TO OPEN R8-DM-722	3.0E-3/d	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
HDM722SQ	DAMPER 722 FAILS TO REMAIN OPEN R8-DM-722	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HDM722SW	DAMPER 722 INTERNAL FAILURE R8-DM-722	3.6E-8/h	8						
HDM723SN	DAMPER 723 MECHANICAL FAILS TO OPEN R8-DM-723	3.0E-3/d	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
HDM723SO	DAMPER 723 FAILS TO REMAIN CLOSED R8-DM-723	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HDM723SQ	DAMPER 723 FAILS TO REMAIN OPEN R8-DM-723	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3

Table 7.6. Shield Cooling System Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
HDM723SW	DAMPER 723 INTERNAL FAILURE R8-DM-723	3.6E-8/h	8						
HDM724SQ	DAMPER 724 FAILS TO REMAIN OPEN R8-DM-724	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HDM725SP	DAMPER 725 MECHANICAL FAILS TO OPEN R8-DM-725	3.0E-3/d	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
HDM725SQ	DAMPER 725 FAILS TO REMAIN OPEN R8-DM-725	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HDM725SW	DAMPER 725 INTERNAL FAILURE R8-DM-725	3.6E-8/h	8						
HDM726SN	DAMPER 726 MECHANICAL FAILS TO CLOSE R8-DM-726	3.0E-3/d	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
HDM726SO	DAMPER 726 FAILS TO REMAIN CLOSED R8-DM-726	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HDM726SQ	DAMPER 726 FAILS TO REMAIN OPEN R8-DM-726	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HDM726SW	DAMPER 726 INTERNAL FAILURE R8-DM-726	3.6E-8/h	8						
HDM727SP	DAMPER 727 MECHANICAL FAILS TO OPEN R8-DM-727	3.0E-3/d	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
HDM727SQ	DAMPER 727 FAILS TO REMAIN OPEN R8-DM-727	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HDM727SW	DAMPER 727 INTERNAL FAILURE R8-DM-727	3.6E-8/h	8						
HDM741SQ	ISOL DAMPER 741 DOES NOT REMAIN OPEN R8-DM-741	3.6E-8/h	8	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HDM742SL	ISOLATION DAMPER 742 LEFT CLOSED R8-DM-742	1.0E-4/d	10	1.0E-4/d	10	1.0E-4/d	3	1.0E-4/d	3
HDM742SQ	ISOL DAMPER 742 DOES NOT REMAIN OPEN R8-DM-742	3.6E-8/h	8	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HDM745SQ	ISOL DAMPER 745 DOES NOT REMAIN OPEN R8-DM-745	3.6E-8/h	8	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HDM746SL	ISOLATION DAMPER 746 LEFT CLOSED R8-DM-746	1.0E-4/d	10	1.0E-4/d	10	1.0E-4/d	3	1.0E-4/d	3
HDM746SQ	ISOL DAMPER 746 DOES NOT REMAIN OPEN R8-DM-746	3.6E-8/h	8	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HDM747SQ	FLTR 717 INLET DMPR 747 FAILS TO REMAIN OPEN FL717/R8-DM-747	3.6E-8/h	8	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HDM748SP	FLTR 718 INLET DAMPE R 748 FAILS TO O PEN FL718/R8-DM-748	2.0E-4/d	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
HDM748SQ	FLTR 718 INLET DMPR 748 FAILS TO REMAIN OPEN FL718/R8-DM-748	3.6E-8/h	8	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3

Table 7.6. Shield Cooling System Data (Cont'd)

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
HDM764SQ	ISOL DAMPER 764 DOES NOT REMAIN OPEN R8-DM-764	3.6E-8/h	8	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HDM765SL	ISOLATION DAMPER 765 LEFT CLOSED R8-DM-765	1.0E-4/d	10	1.0E-4/d	10	1.0E-4/d	3	1.0E-4/d	3
HDM765SQ	ISOL DAMPER 765 DOES NOT REMAIN OPEN R8-DM-765	3.6E-8/h	8	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HEN001R0	RX BUILDING AMBIENT TEMP TOO HIGH								
HFL717SE	ABSOLUTE FILTE RS 717 CLOGGED R8-FL-717	3.0E-5/h	10	1.0E-5/h		3.0E-5/h	10		
HFL718SE	ABSOLUTE FILTE RS 718 CLOGGED R8-FL-718	3.0E-5/h	10	1.0E-5/h		3.0E-5/h	10		
HFN706SS	EXHAUST FAN 706 FAILS TO START R8-F-706	1.0E-3/d	10	3.0E-4/d	3			3.0E-4/d	3
HFN708SR	RECIRCULATION FAN 708 FAILS TO RUN R8-F-708	3.0E-5/h	10	1.0E-5/h				1.0E-5/h	10
HFN708SS	RECIRCULATION FAN 708 FAILS TO START R8-F-708	1.0E-3/d	10	3.0E-4/d	3			3.0E-4/d	3
HGV346PQ	REGULATOR FOR VS-34 6A FAILS TO REMAIN OPEN P17-VS-346A	1.0E-7/h	10	1.0E-7/h	3				
HGV347PQ	REGULATOR FOR VS-34 7A FAILS TO REMAIN OPEN P17-VS-347A	1.0E-7/h	10	1.0E-7/h	3				
HGV348PQ	REGULATOR FOR VS-34 8A FAILS TO REMAIN OPEN P17-VS-348A	1.0E-7/h	10	1.0E-7/h	3				
HGV349PQ	REGULATOR FOR VS-34 9A FAILS TO REMAIN OPEN P17-VS-349A	1.0E-7/h	10	1.0E-7/h	3				
HGV702PQ	REGULATOR FOR VS-70 2A FAILS TO REMAIN OPEN P17-VS-702A	1.0E-7/h	10	1.0E-7/h	3				
HGV703PQ	REGULATOR FOR VS-70 3A FAILS TO REMAIN OPEN P17-VS-703A	1.0E-7/h	10	1.0E-7/h	3				
HGV735SL	COIL 715 #2 EXPANSION VALVE LEFT CLOSED R8-VC-735	1.0E-4/d	10	1.0E-4/d	10	1.0E-4/d	3	1.0E-4/d	
HGV735SQ	COIL 715 #2 EXP VALV E DOES NOT REMAIN OP EN R8-VC-735	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HGV736SL	COIL 715 #1 EXPANSION VALVE LEFT CLOSED R8-VC-736	1.0E-4/d	10	1.0E-4/d	10	1.0E-4/d	3	1.0E-4/d	
HGV736SQ	COIL 715 #1 EXP VALV E DOES NOT REMAIN OP EN R8-VC-736	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HGV737SL	COIL 718 #1 EXPANSION VALVE LEFT CLOSED R8-VC-737	1.0E-4/d	10	1.0E-4/d	10	1.0E-4/d	3	1.0E-4/d	

Table 7.6. Shield Cooling System Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
HGV737SQ	COIL 718 #1 EXP VALV E DOES NOT REMAIN OP EN R8-VC-737	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HGV738SL	COIL 718 #2 EXPANSION VALVE LEFT CLOSED R8-VC-738	1.0E-4/d	10	1.0E-4/d	3	1.0E-4/d	10	1.0E-4/d	3
HGV738SQ	COIL 718 #2 EXP VALV E DOES NOT REMAIN OP EN R8-VC-738	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HGV754SQ	COIL 713 #1 EXP VALV E DOES NOT REMAIN OP EN R8-VC-754	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HHU002YL	NO RECOVERY ACTION TO RESET CIRCUIT BREAKER	2.0E-2/d	10						
HHU003SL	NO RECOVERY ACTION TO USE DAMPER MANUAL HANDWHEEL	2.0E-2/d	10						
HHU004SL	NO RECOVERY ACTION TO USE MOVE DAMPER MANUALLY	2.0E-2/d	10						
HHU01BSZ	NO RECOVERY ACTION FOR CLOGGED FILTER R8-FL-717	2.0E-2/d	10						
HLC641SW	PRESSURE CONTROLLER FAILS R13-1564-1	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
HLC642SW	PRESSURE CONTROLLER FAILS R13-1564-2	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HPR641SW	PRESSURE TRANSDUCER FAILS R13-1564-1	3.0E-6/h	10	1.0E-6/h				3.0E-6/h	10
HPR642SW	PRESSURE TRANSDUCER FAILS R13-1564-2	3.0E-6/h	10	1.0E-6/h				3.0E-6/h	10
HRE00CSO	CONTROL RELAY CR-10 0 COIL FAILURE DM-722/CR-100	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
HRE00CSW	CONTROL RELAY CR-10 0 CONTACT FAILURE DM-722/CR-100	3.0E-4/d	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d	10
HRE00MSW	MOTOR OPERATOR RM-10 0 FAILS TO OPERATE DM-722/RM-100	3.0E-3/d	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3
HRE01CSW	CONTROL RELAY CR-10 1 CONTACT FAILURE DM-724/CR-101	3.0E-4/d	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d	10
HRE02CSO	CONTROL RELAY CR-10 2 COIL FAILURE DM-725/CR-102	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
HRE02CSW	CONTROL RELAY CR-10 2 CONTACT FAILURE DM-725/CR-102	3.0E-4/d	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d	10
HRE02MSW	MOTOR OPERATOR RM-10 2 FAILS TO OPERATE DM-725/RM-102	3.0E-3/d	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3
HRE051SO	MOTOR RELAY RM-51 COIL FAILU RE FN-705/RM-51	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10

Table 7.6. Shield Cooling System Data (Cont'd)

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
HRE097SW	MOTOR OPERATOR VORTEX DAMPER FAILS FN-706/RM-9	3.0E-3/d	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3
HRE99CSN	CONTROL RELAY CR-99 FAILS TO CLOSE DM-723/CR-99	3.0E-4/d	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d	10
HRE99CSO	CONTROL RELAY CR-99 COIL FAILURE DM-723/CR-99	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
HRE99MSW	MOTOR OPERATOR RM-99 FAILS TO OPERATE DM-723/RM-99	3.0E-3/d	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3
HSV095SO	SOLENOID VALVE RSV-9 5 FAILS TO REMAIN CLOSED DM-727/RSV-95	1.0E-7/h	3	1.0E-7/h	3				
HSV095SP	SOLENOID VALVE RSV-9 5 FAILS TO OPEN DM-727/RSV-95	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	30	1.0E-3/d	3
HSV095SQ	SOLENOID VALVE RSV-9 5 FAILS TO REMAIN OPEN DM-727/RSV-95	1.0E-7/h	3	1.0E-7/h	3				
HSV096SN	SOLENOID VALVE RSV-96 FAILS TO CLOSE DM-726/RSV-96	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	30	1.0E-3/d	3
HSV096SO	SOLENOID VALVE RSV-9 6 FAILS TO REMAIN CLOSED DM-726/RSV-96	1.0E-7/h	3	1.0E-7/h	3				
HSV096SQ	SOLENOID VALVE RSV-9 6 FAILS TO REMAIN OPEN DM-726/RSV-96	1.0E-7/h	3	1.0E-7/h	3				
HSV346PQ	SOLENOID VALVE VS-34 6A FAILS TO REMAIN OPEN P17-VS-346A	1.0E-7/h	3	1.0E-7/h	3				
HSV347PQ	SOLENOID VALVE VS-34 7A FAILS TO REMAIN OPEN P17-VS-347A	1.0E-7/h	3	1.0E-7/h	3				
HSV348PQ	SOLENOID VALVE VS-34 8A FAILS TO REMAIN OPEN P17-VS-348A	1.0E-7/h	3	1.0E-7/h	3				
HSV349PQ	SOLENOID VALVE VS-34 9A FAILS TO REMAIN OPEN P17-VS-349A	1.0E-7/h	3	1.0E-7/h	3				
HSV356SN	SPACE COOL 309 SOLENOID FAILS TO CLOSE R8-VS-356	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3
HSV356SO	SPACE COOL 309 SOLENOID FAILS TO REMAIN CLOSED R8-VS-356	1.0E-7/h	3	1.0E-7/h	3				
HSV357SP	STBY COILS 715 SOLENOID XFER FAILS TO OPEN R8-VS-357	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3
HSV357SQ	STBY COILS 715 SOLENOID FAILS TO REMAIN OPEN R8-VS-35	1.0E-7/h	3	1.0E-7/h	3				
HSV358LN	SPACE COOL 308 SOLENOID FAILS TO CLOSE R8-VS-358	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3
HSV358LO	SPACE COOL 308 SOLENOID FAILS TO REMAIN CLOSED R8-VS-358	1.0E-7/h	3	1.0E-7/h	3				

Table 7.6. Shield Cooling System Data (Cont'd)

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
HSV359SP	STBY COILS 718 SOLENOID XFER FAILS TO OPEN R8-VS-359	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3
HSV359SQ	STBY COILS 718 SOLENOID FAILS TO REMAIN OPEN R8-VS-35	1.0E-7/h	3	1.0E-7/h	3				
HSV400LN	CLR 308 COIL 2 SOLENOID FAILS TO CLOSE R8-VS-400	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3
HSV400LO	CLR 308 COIL 2 SOLENOID FAILS TO REMAIN CLOSED R8-VS-400	1.0E-7/h	3	1.0E-7/h	3				
HSV702PQ	SOLENOID VALVE VS-70 2A FAILS TO REMAIN OPEN P17-VS-702A	1.0E-7/h	3	1.0E-7/h	3				
HSV703PQ	SOLENOID VALVE VS-70 3A FAILS TO REMAIN OPEN P17-VS-703A	1.0E-7/h	3	1.0E-7/h	3				
HSV755SQ	COIL 714 #1 SOLENOID FAILS TO REMAIN OPEN R8-VS-75	1.0E-7/h	3	1.0E-7/h	3				
HSV756SQ	COIL 713 #2 SOLENOID FAILS TO REMAIN OPEN R8-VS-75	1.0E-7/h	3	1.0E-7/h	3				
HSV757SP	COIL 718 #1 SOLENOID FAILS TO OPEN R8-VS-757	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3
HSV757SQ	COIL 718 #1 SOLENOID FAILS TO REMAIN OPEN R8-VS-75	1.0E-7/h	3	1.0E-7/h	3				
HSV758SP	COIL 715 #2 SOLENOID FAILS TO OPEN R8-VS-758	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3
HSV758SQ	COIL 715 #2 SOLENOID FAILS TO REMAIN OPEN R8-VS-75	1.0E-7/h	3	1.0E-7/h	3				
HSV759SN	CLR 309 COIL 1 SOLENOID FAILS TO CLOSE R8-VS-759	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3
HSV759SO	CLR 309 COIL 1 SOLENOID FAILS TO REMAIN CLOSED R8-VS-759	1.0E-7/h	3	1.0E-7/h	3				
HSV766SN	CLR 309 COIL 2 SOLENOID FAILS TO CLOSE R8-VS-766	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3
HSV766SO	CLR 309 COIL 2 SOLENOID FAILS TO REMAIN CLOSED R8-VS-766	1.0E-7/h	3	1.0E-7/h	3				
HSV767SP	COIL 715 #1 SOLENOID FAILS TO OPEN R8-VS-767	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3
HSV767SQ	COIL 715 #1 SOLENOID FAILS TO REMAIN OPEN R8-VS-76	1.0E-7/h	3	1.0E-7/h	3				
HSV768SQ	COIL 714 #2 SOLENOID FAILS TO REMAIN OPEN R8-VS-76	1.0E-7/h	3	1.0E-7/h	3				
HSV769SQ	COIL 713 #1 SOLENOID FAILS TO REMAIN OPEN R8-VS-76	1.0E-7/h	3	1.0E-7/h	3				

Table 7.6. Shield Cooling System Data (Cont'd)

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
HSV770SP	COIL 718 #2 SOLENOID FAILS TO OPEN R8-VS-770	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3
HSV770SQ	COIL 718 #2 SOLENOID FAILS TO REMAIN OPEN R8-VS-77	1.0E-7/h	3	1.0E-7/h	3				
HSV771LN	CLR 308 COIL 1 SOLENOID FAILS TO CLOSE R8-VS-771	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3
HSV771LO	CLR 308 COIL 1 SOLENOID FAILS TO REMAIN CLOSED R8-VS-771	1.0E-7/h	3	1.0E-7/h	3				
HSW201ST	20 TON COMP #1 SWITCH IN WRONG POSITION	1.0E-4/d	10						
HSW202ST	20 TON COMP #2 SWITCH IN WRONG POSITION	1.0E-4/d	10						
HTZ096SW	DM726 POSITION INDICATOR ERROR DM-726/CR-96-7	3.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
HTZ099SW	DM723 POSITION INDICATOR ERROR DM-723/ES-99-1	3.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
HTZ741SW	DM741 POSITION INDICATOR ERROR	3.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
HTZ742SW	DM742 POSITION INDICATOR ERROR	3.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
HTZ745SW	DM745 POSITION INDICATOR ERROR DM745/MD-51-1	3.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
HTZ746SW	DM746 POSITION INDICATOR ERROR DM-746/MD-50-1	3.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
HWR705TB	FAN 705 POWER CABLE FAILS: OPEN CIRCUIT	3.0E-6/h	10						
HWR705TC	FAN 705 POWER CABLE FAILS: SHORT TO GROUND	3.0E-7/h	10						
HWR706TB	FAN 706 POWER CABLE FAILS: OPEN CIRCUIT	3.0E-6/h	10						
HWR706TC	FAN 706 POWER CABLE FAILS: SHORT TO GROUND	3.0E-7/h	10						
HWR707TB	FAN 707 POWER CABLE FAILS: OPEN CIRCUIT	3.0E-6/h	10						
HWR707TC	FAN 707 POWER CABLE FAILS: SHORT TO GROUND	3.0E-7/h	10						
HWR708TB	FAN 708 POWER CABLE FAILS: OPEN CIRCUIT	3.0E-6/h	10						
HWR708TC	FAN 708 POWER CABLE FAILS: SHORT TO GROUND	3.0E-7/h	10						
HXT048SW	RM48 480V/120V 250VA STARTER TRANSFORMER FAILS	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
HXT050SW	RM50 480V/120V 250VA STARTER TRANSFORMER FAILS	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3

Table 7.6. Shield Cooling System Data (Cont'd)

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
HXT051SW	RM51 480V/120V 250VA STARTER TRANSFORMER FAILS	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
HXT054SW	RM54 480V/120V 500VA STARTER TRANSFORMER FAILS	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
HXT098SW	RM98 120V/24V TRANSFORMER FAILS	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
HXT099SW	CR99 120V/24V TRANSFORMER FAILS	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
HXT100SW	CR100 120V/24V TRANSFORMER FAILS	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
HXV953SQ	COILS 713/714 ISOLATION FAILS TO REMAIN OPEN R8-V-953	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HXV954SQ	COIL 713 #2 ISOLATION FAILS TO REMAIN OPEN R8-V-954	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HXV955SQ	COIL 713 #1 ISOLATION FAILS TO REMAIN OPEN R8-V-955	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HXV956SQ	COIL 714 #2 ISOLATION FAILS TO REMAIN OPEN R8-V-956	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HXV957SQ	COIL 714 #1 ISOLATION FAILS TO REMAIN OPEN R8-V-957	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HXV958SL	COIL 715 #2 ISOLATION LEFT CLOSED R8-V-958	1.0E-4/d	10	1.0E-4/d	3	1.0E-4/d	10	1.0E-4/d	3
HXV958SQ	COIL 715 #2 ISOLATION FAILS TO REMAIN OPEN R8-V-958	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HXV959SL	STBY COILS 718 ISOLATION LEFT CLOSED R8-V-959	1.0E-4/d	10	1.0E-4/d	3	1.0E-4/d	10	1.0E-4/d	3
HXV959SQ	STBY COILS 718 ISOLATION FAILS TO REMAIN OPEN R8-V-959	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HXV960SL	COIL 715 #1 ISOLATION LEFT CLOSED R8-V-960	1.0E-4/d	10	1.0E-4/d	3	1.0E-4/d	10	1.0E-4/d	3
HXV960SQ	COIL 715 #1 ISOLATION FAILS TO REMAIN OPEN R8-V-960	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HXV961SL	COIL 718 #1 ISOLATION LEFT CLOSED R8-V-961	1.0E-4/d	10	1.0E-4/d	3	1.0E-4/d	10	1.0E-4/d	3
HXV961SQ	COIL 718 #1 ISOLATION FAILS TO REMAIN OPEN R8-V-961	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HXV962SL	COIL 718 #2 ISOLATION LEFT CLOSED R8-V-962	1.0E-4/d	10	1.0E-4/d	3	1.0E-4/d	10	1.0E-4/d	3
HXV962SQ	COIL 718 #2 ISOLATION FAILS TO REMAIN OPEN R8-V-962	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
HXV963SL	STBY COILS 715 ISOLATION LEFT CLOSED R8-V-963	1.0E-4/d	10	1.0E-4/d	3	1.0E-4/d	10	1.0E-4/d	3

Table 7.6. Shield Cooling System Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
HXV963SQ	STBY COILS 715 ISOLATION FAILS TO REMAIN OPEN R8-V-963	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3

Table 7.7. Thimble Cooling System Data

PAGE 1

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
TDM728SN	DISCHARGE DAMPER 728 FAILS TO CLOSE R8-DM-728	3.0E-3/d	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3
TDM728SO	DISC DAMPER 728 FAILS TO REMAIN CLOSED R8-DM-728	1.0E-7/h	10	1.0E-7/h	3				
TDM728SQ	DISC DAMPER 728 FAILS TO REMAIN OPEN R8-DM-728	1.0E-7/h	10	1.0E-7/h	3				
TDM728SW	DAMPER 728 INTERNAL FAILURE R8-DM-728	3.6E-8/h	8						
TDM729SO	DISC DAMPER 729 FAILS TO REMAIN CLOSED R8-DM-729	1.0E-7/h	10	1.0E-7/h	3				
TDM729SP	DISCHARGE DAMPER 729 FAILS TO OPEN R8-DM-729	3.0E-3/d	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3
TDM729SQ	DISC DAMPER 729 FAILS TO REMAIN OPEN R8-DM-729	1.0E-7/h	10	1.0E-7/h	3				
TDM729SW	DAMPER 729 INTERNAL FAILURE R8-DM-729	3.6E-8/h	8						
TDM749SQ	ISOL DAMPER 749 FAILS TO REMAIN OPEN R8-DM-749	1.0E-7/h	10	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3
TDM751SL	ISOLATION DAMPER 751 LEFT CLOSED R8-DM-751	1.0E-4/d	10	1.0E-4/d	3	1.0E-4/d	10	1.0E-4/d	3
TDM751SQ	ISOL DAMPER 751 FAILS TO REMAIN OPEN R8-DM-751	1.0E-7/h	10	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3
TDM761SQ	THROTTLE DMP 761 FAILS TO REMAIN OPEN R8-DM-761	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	
TDM762SQ	THROTTLE DMP 762 FAILS TO REMAIN OPEN R8-DM-762	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	
TFL001SE	HEPA FILTER #1 PLUGGED	3.0E-5/h	10	1.0E-5/h		1.0E-5/h	10	3.0E-5/h	
TFL002SE	HEPA FILTER #2 PLUGGED	3.0E-5/h	10	1.0E-5/h		3.0E-5/h	10		
TFL003SE	HEPA FILTER #3 PLUGGED	3.0E-5/h	10	1.0E-5/h		3.0E-5/h	10		
TFL0J1FE	J-1 INLET FILTER PLUGGED	3.0E-5/h	10	1.0E-5/h		3.0E-5/h	10		
TFL0J3FE	J-3 INLET FILTER PLUGGED	3.0E-5/h	10	1.0E-5/h		3.0E-5/h	10		
TFL0J4FE	J-4 INLET FILTER PLUGGED	3.0E-5/h	10	1.0E-5/h		3.0E-5/h	10		
TFLHEPSU	HEPA FILTERS PLUGGED DUE TO COMMON CAUSE	3.0E-6/h	10						
TFLJINFU	THIMBLE INLET FILTERS PLUGGED DUE TO COMMON CAUSE	3.0E-6/h	10						
TFN709SR	TURBO FAN 709 FAILS TO RUN R8-FN-709	3.0E-5/h	10	1.0E-5/h				1.0E-5/h	10

Table 7.7. Thimble Cooling System Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
TFN710SR	TURBO FAN 710 FAILS TO RUN R8-FN-710	3.0E-5/h	10	1.0E-5/h				1.0E-5/h	10
TFN710SS	TURBO FAN 710 FAILS TO START R8-FN-710	1.0E-3/d	10	3.0E-4/d	3			3.0E-4/d	3
THU002YL	NO RECOVERY ACTION TO RESET BREAKER	2.0E-2/d	10						
THU01BSZ	NO RECOVERY ACTION TO CROSS-CONNECT SHIELD COOLING	2.0E-2/d	10						
TPR01ASW	PRESSURE TRANSDUCER FAILS DP-PS-1A	3.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
TRE091SO	CONTROL RELAY RM-91 COIL FAILURE FN-709/CR-91	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
TRE092SO	CONTROL RELAY RM-92 COIL FAILURE FN-710/CR-92	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
TRE52XSO	CONTROL RELAY RM-52 X COIL FAILURE FN-709/CR-52X	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
TRE53TSN	TIME DELAY RELAY TD-53A CONTACT FAILS TO CLOSE	3.0E-4/d	10	1.0E-4/d	10	3.0E-4/d	10	3.0E-4/d	10
TRE53XSO	CONTROL RELAY RM-53 X COIL FAILURE FN-710/CR-53X	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
TSV091SQ	SOLENOID VALVE RSV-91 FAILS TO REMAIN OPEN RSV-91	1.0E-7/h	3	1.0E-7/h	3				
TSV092SP	SOLENOID VALVE RSV-92 FAILS TO OPEN RSV-92	1.0E-7/h	3	1.0E-7/h	3				
TSV092SQ	SOLENOID VALVE RSV-92 FAILS TO REMAIN OPEN RSV-92	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3	1.0E-3/d	3
TSW761ST	THROTTLE DMR 761 SWIT CH IN WRONG POSITIO N DM-761/RM93	1.0E-4/d	10	1.0E-4/d	3	1.0E-4/d	3	1.0E-4/d	3
TTZ728SW	DM728 POSITION INDICATOR ERROR DM-728	3.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
TTZ729SW	DM729 POSITION INDICATOR ERROR DM-729	3.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
TTZ749SW	DM749 POSITION INDIC ATOR ERROR DM-749/RSV-91	3.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
TTZ751SW	DM751 POSITION INDIC ATOR ERROR DM-751/RSV-92	3.0E-6/h	10	1.0E-6/h	10			3.0E-6/h	10
TWR709TB	FAN 709 POWER CABLE FAILS: OPEN CIRCUIT	3.0E-6/h	10						
TWR709TC	FAN 709 POWER CABLE FAILS: SHORT TO GROUND	3.0E-7/h	10						
TWR710TB	FAN 710 POWER CABLE FAILS: OPEN CIRCUIT	3.0E-6/h	10						

Table 7.7. Thimble Cooling System Data (Cont'd)

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
TWR710TC	FAN 710 POWER CABLE FAILS: SHORT TO GROUND	3.0E-7/h	10						
TXT052SW	RM52 480V/120V 250VA STARTER TRANSFORMER FAILS	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
TXT053SW	RM53 480V/120V 250VA STARTER TRANSFORMER FAILS	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
TXV005FQ	J-1 ISOLATION VALVE FAILS TO REMAIN OPEN RSA-5	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
TXV007FQ	J-3 ISOLATION VALVE FAILS TO REMAIN OPEN RSA-7	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
TXV008FQ	J-4 ISOLATION VALVE FAILS TO REMAIN OPEN RSA-8	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
TXV01ISQ	HEPA FILTER #1 INLET FAILS TO REMAIN OPEN	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
TXV01OSQ	HEPA FILTER #1 OUTLET FAILS TO REMAIN OPEN	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
TXV02ISQ	HEPA FILTER #2 INLET FAILS TO REMAIN OPEN	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
TXV02OSQ	HEPA FILTER #2 OUTLET FAILS TO REMAIN OPEN	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
TXV03ISQ	HEPA FILTER #3 INLET FAILS TO REMAIN OPEN	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
TXV03OSQ	HEPA FILTER #3 OUTLET FAILS TO REMAIN OPEN	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
TXV05AFO	J-1 BYPASS VALVE FAILS TO REMAIN CLOSED RSA-5A	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
TXV07AFO	J-3 BYPASS VALVE FAILS TO REMAIN CLOSED RSA-7A	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
TXV08AFO	J-4 BYPASS VALVE FAILS TO REMAIN CLOSED RSA-8A	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
TXV752SQ	BALANCE DAMPER LEG B FAILS TO REMAIN OPEN R8-DM-752	1.0E-7/h	10	1.0E-7/h	3				
TXV753SQ	BALANCE DAMPER LEG A FAILS TO REMAIN OPEN R8-DM-753	1.0E-7/h	10	1.0E-7/h	3				
TXV770SP	SHIELD COOLING X-CONN FAILS TO OPEN R8-DM-77	2.0E-4/d	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
TXV770SQ	SHIELD COOLING X-CONN FAILS TO REMAIN OPEN R8-DM-770	1.0E-7/h	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	3
TXVBYPFU	J AND O BYPASS VALVES FAIL TO REMAIN CLOSED - COMMON CAUSE	1.0E-8/h	10						
TXVISOFU	INLET ISOLATION VALVES FAIL TO REMAIN OPEN - COMMON CAUSE	1.0E-8/h	10						

Table 7.8. Stack Exhaust System Data

PAGE 1

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
YDMD1AOQ	DAMPER DA-1A FAILS TO REMAIN OPEN EXTERNALLY	6.4E-8/h [1]	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	
YDMD1AOW	DAMPER DA-1A FAILS TO REMAIN OPEN INTERNALLY	3.6E-8/h	8						
YDMD1BOQ	FAN VANES D-1B FAIL REMAIN OPEN EXTERNALLY	6.4E-8/h [1]	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	
YDMD1BOW	FAN VANES D-1B FAIL TO REMAIN OPEN INTERNALLY	3.6E-8/h	8						
YDMD1COQ	DAMPER DA-1C FAILS TO REMAIN OPEN EXTERNALLY	6.4E-8/h [1]	10	1.0E-7/h	3	1.0E-7/h	3	1.0E-7/h	
YDMD1COW	DAMPER DA-1C FAILS TO REMAIN OPEN INTERNALLY	3.6E-8/h	8						
YDMD2AOP	DAMPER DA-2A FAILS TO OPEN EXTERNALLY	3.0E-3/d	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3
YDMD2AOQ	DAMPER DA-2A FAILS TO REMAIN OPEN EXTERNALLY	6.4E-8/h [1]	10						
YDMD2AOW	DAMPER DA-2A FAILS TO REMAIN OPEN INTERNALLY	3.6E-8/h [2]	10						
YDMD2BOP	FAN VANES D-2B FAIL TO OPEN EXTERNALLY	3.0E-3/d	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3
YDMD2BOQ	FAN VANES D-2B FAIL TO REMAIN OPEN EXTERNALLY	6.4E-8/h [1]	10						
YDMD2BOW	FAN VANES D-2B FAIL TO REMAIN OPEN INTERNALLY	3.6E-8/h [2]	10						
YDMD2COP	DAMPER DA-2C FAILS TO OPEN EXTERNALLY	3.0E-3/d	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3
YDMD2COQ	DAMPER DA-2C FAILS TO REMAIN OPEN EXTERNALLY	6.4E-8/h [1]	10						
YFN056OR	FAN FM-56 FAILS TO RUN FM-56	2.0E-5/h	3	1.0E-5/h				1.0E-5/h	10
YFN058OR	FAN FM-58 FAILS TO RUN FM-58	2.0E-5/h	3	1.0E-5/h				1.0E-5/h	10
YFN058OS	FAN FM-58 FAILS TO START FM-58	3.0E-4/d	3	3.0E-4/d	3			3.0E-4/d	3
YHU001OL	NO RECOVERY ACTION TO MANUALLY OPEN DAMPER	1.0E-1/d [3]	10						

Notes: 1. Motor Operated Valve Minus Internal HVAC Valve
2. Manual HVAC Damper (IAEA)
3. Lower Recovery; non-EBR-II personnel

Table 7.9. Secondary Sodium System Data

PAGE 1

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
N00000UE	SECONDARY SODIUM BLOCKAGE	2.0 E-7/h	10						
N00000UF	SECONDARY SODIUM LEAK AND/OR FIRE	6.0 E-7/h	10						
NPMOVCWN	Probability of Secondary Pump Failing to Trip	1.0 E-3/d	10						
NSW000CZ	INADVERTANT OPERATOR ACTION TO DUMP SECONDARY Na	1.0 E-4/d	10						

Table 7.10. Balance of Plant Data - Steam System Data

PAGE 1

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
VDL000UW	DIESEL DRIVEN FIRE PUMPS NOT AVAILABLE	1.0E-2/d	5						
VOOOOOUW	OTHER BOP FAILURES	5.1E-5/h	10						
VRV161YP	STEAM HEADER RELIEF DOES NOT OPEN B3-VC-1610	3.0E-3/d	5						
VRV161YQ	STEAM HEADER RELIEF DOES NOT REMAIN OPEN B3-VC-1610	5.0E-6/h	10						

Table 7.11. Balance of Plant Data - General Data

PAGE 1

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
BOO01Z00	BOP Not Available; Sodium Drained	1.0	1						
BOO02Z00	BALANCE OF PLANT NOT AVAILABLE FOR LONG TERM HEAT REMOVAL	1.0	1						
BOO99ZUW	Operating on Hotel Mode	2.0E-3/d	10						

Table 7.12. Electrical Normal Power System Data

PAGE 1

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
JBS001OB	Loss of 138 kV Bus	1.0E-4/h [1]	10	1.0E-8/h	3	1.0E-8/h	3	1.0E-8/h	3
JBS001ZD	Overload Fault on 13.8 kV Bus 1	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
JBS002ZD	Overload Fault on 13.8 kV Bus 2	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
JBS200YD	2400 V Bus Overload Fault	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
JBS480ZU	480 Vac Normal Bus No. 1 and Bus No. 2 Common Mode Failure	1.0E-8/h [4]	10						
JBSN1LVC	Uncleared Overload on Normal Bus No. 1 Load Side	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
JBSN2LVC	Uncleared Overload on Normal Bus No. 2 Load Side	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
JBSP1AYC	Uncleared Overload on Normal Bus P-1A	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
JBSP2AYC	Uncleared Overload on Normal Bus P-2A	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
JBSP2BYC	Uncleared Overload on Normal Bus P-2B	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
JBSP3XYC	Uncleared Overload on Normal Bus P-3	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
JBSR1AYC	Uncleared Overload on Normal Bus R-1A	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
JBSR3XYC	Uncleared Overload on Normal Bus R-3	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
JBSS1AYC	Uncleared Overload on Normal Bus. S-1A	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
JCA131ZB	Supply Wiring Fails Open, Transf. No. 5	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
JCA132ZB	Supply Wiring Fails Open, Transf. No. 6	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
JCAN1SVC	Overload Fault Between Breaker 1CD and Normal Bus No. 1	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
JCAN2SVC	Overload Fault Between Breaker 7CD and Normal Bus No. 2	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
JCAT5OVC	Overload Fault Occurs Before Breaker 1CD	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
JCAT6OVC	Overload Fault Occurs Before Breaker 7CD	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
JCB002ZO	13.8 KV Breaker No. 2 Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JCB013ZO	13.8 KV Breaker No. 13 Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JCB02CYO	MCC-P2A Circuit Breaker 2C Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JCB03BVO	Normal Bus No. 1 Breaker 3B Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3

Table 7.12. Electrical Normal Power System Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
JCB05CVO	Normal Bus No. 2 Breaker 5C Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JCB11AYO	Breaker 11A of MCC P2B Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JCB11CYO	MCC-P2B Circuit Breaker 11C Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JCB12AYO	Normal Bus P-2B Breaker 12A Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	10
JCB1A1YO	Normal Bus R-1A Breaker 1A Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	10
JCB1A2YO	Normal Bus P-2A Breaker 1A Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	10
JCB1A3YO	Normal Bus P-1A Breaker 1A Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	10
JCB1A4YO	Normal Bus R-3 Breaker 1A Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	10
JCB1CDVO	480 Vac Breaker 1CD Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JCB1CDVP	Breaker 1CD Fails to Open on Demand	5.0E-4/d [2]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
JCB1D1YO	Normal Bus S-1A Breaker 1D Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	10
JCB2B1VO	Normal Bus No. 1 Breaker 2B Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JCB2C1VO	Normal Bus No. 1 Breaker 2C Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JCB2C1YO	Normal Bus P-3 Breaker 2C Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	10
JCB2D1VO	Normal Bus No. 1 Breaker 2D Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JCB3C1VO	Normal Bus No. 1 Breaker 3C Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JCB4BCVO	Tie Breaker 4BC Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JCB4BCVP	Breaker 4BC Fails to Open on Demand	5.0E-4/d [2]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
JCB5B1VO	Normal Bus No. 2 Breaker 5B Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JCB6C1VO	Normal Bus No. 2 Breaker 6C Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JCB7CDVO	480 Vac Breaker 7CD Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JCB7CDVP	Breaker 7CD Fails to Open on Demand	5.0E-4/d [2]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10

Table 7.12. Electrical Normal Power System Data (Cont'd)

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
JCBN3AVN	Normal Bus No. 1 Breaker 3A Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JCBN3AVZ	Operator Fails to Close Norm Bus No. 1 Brkr 3A	1.0E-4/d [5]	10						
JCBN5CVP	Normal Bus No. 2 Breaker 5C Fails to Open	5.0E-4/d [2]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
JCBSA1VO	Normal Bus No. 2 Breaker 5A Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JFU417VB	4BC, 1CD, 7CD Fault Detection Ckts Fuse Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
JFUN5CVB	Normal Bus No. 2 Breaker 5C Trip Fuse Opens	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
JGE001ZW	Generator Electrical or Mechanical Malfunction	3.0E-4/h [2]	3	1.0E-6/h [6]	10	3.0E-6/h [6]	3		
JIN12OVW	4BC-12 Fault Detection Ckt Fails	3.0E-6/h [2]	10	1.0E-6/h [7]	10	3.0E-6/h [7]	10	2.0E-6/h [7]	10
JIN1CDVW	ICD Fault Detection CKT Fails	3.0E-6/h [2]	10	1.0E-6/h [7]	10	3.0E-6/h [7]	10	2.0E-6/h [7]	10
JIN21OVW	4BC-21 Fault Detection Ckt Fails	3.0E-6/h [2]	10	1.0E-6/h [7]	10	3.0E-6/h [7]	10	2.0E-6/h [7]	10
JIN7CDVW	7CD Fault Detection Ckt Fails	3.0E-6/h [2]	10	1.0E-6/h [7]	10	3.0E-6/h [7]	10	2.0E-6/h [7]	10
JINN5CVW	Normal Bus No. 2 Breaker 5C Trip Signal Ckt Fails	3.0E-6/h [2]	10	1.0E-6/h [7]	10	3.0E-6/h [7]	10	2.0E-6/h [7]	10
JOO01Z00	Generator On-Line	1.0 [8]	1						
JOO10Z00	NO RECOVERY OF LONP	1.0	1						
JOO55Z00	NO RECOVERY OF 480 V NORMAL BUS P-2A	1.0	1						
JSG100ZU	13.8 kV Switchgear Common Mode Failure	3.0E-8/h [4]	10						
JSG103ZO	13.8 kV Switchgear Unit 3 Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JSG105ZO	13.8 kV Switchgear Unit 5 Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JSG107ZO	13.8 kV Switchgear Unit 7 Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JSG109ZO	13.8 kV Switchgear Bus Tie (Unit 9) Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JSG109ZP	13.8 kV Switchgear Bus Tie (Unit 9) Fails to Open	5.0E-4/d [2]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3
JSG110ZO	13.8 kV Switchgear Unit 10 Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JSG112ZO	13.8 kV Switchgear Unit 12 Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JSG200VU	2400 V Switchgear Common Mode Failure	3.0E-8/h [4]	10						

Table 7.12. Electrical Normal Power System Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
JSG201VO	MG Set 1 Input Breaker Fails Open	3.0E-7/h [9]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JSG202VO	MG Set 2 Input Breaker Fails Open	3.0E-7/h [9]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JSG203VO	2400 V Switchgear Unit 3 Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JSG209VO	2400 V Switchgear Unit 9 Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
JSG212VU	MG Set 1 and 2 Input Breaker Fails Open, Common Cause	3.0E-8/h [4]	10						
JSG221VU	MG Set 2 Input Breaker Fails Open, Common Cause	3.0E-8/h [4]	10						
JTR001OW	138/13.8 kV Transformer 1 Fails	1.0E-6/h [2]	10	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
JTR002OW	138/13.8 kV Transformer 2 Fails	1.0E-6/h [2]	10	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
JTR003OW	13.8/2.4 kV Transformer 3 Fails	1.0E-6/h [2]	10	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
JTR004OW	13.8/2.4 kV Transformer 4 Fails	1.0E-6/h [2]	10	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
JTR005OB	Transf. No. 5 Fails Open	1.0E-6/h [2]	10	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
JTR006OB	Transf. No. 6 Fails Open	1.0E-6/h [2]	10	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
JTR012OU	138/13.8 kV Transformers Common Mode Failure	1.0E-7/h [4]	10						
JTR034OU	13.8/2.4 kV Transformers Common Mode Failure	1.0E-7/h [4]	10						
JTRT56OU	13.8 KV/480 Vac Transf 5 and 6 Common Mode Failure	1.0E-7/h [4]	10						
JWR000OU	138 kV Feed Wires Common Mode Failure	1.0E-8/h [4]	10						
JWR001OB	138 kV Feed Wire to Transformer 1 Fails	1.0E-7/h [2]	10	1.0E-6/h [10]	10	3.0E-6/h [10]	10	3.0E-6/h [10]	3
JWR002OB	138 kV Feed Wire to Transformer 2 Fails	1.0E-7/h [2]	10	1.0E-6/h [10]	10	3.0E-6/h [10]	10	3.0E-6/h [10]	3
JWR100OU	13.8 kV Feed Wires to Switchgear Common Mode Failure	1.0E-8/h [4]	10						
JWR103ZB	13.8 kV Feed Wire to Transformer 3 Fails	1.0E-7/h [2]	10	1.0E-6/h [10]	10	3.0E-6/h [10]	10	3.0E-6/h [10]	3
JWR105OB	13.8 kV Feed Wire to Switchgear Unit 5 Fails	1.0E-7/h [2]	10	1.0E-6/h [10]	10	3.0E-6/h [10]	10	3.0E-6/h [10]	3
JWR107OB	Feed Wire to 13.8 kV Switchgear Unit 7 Fails	1.0E-7/h [2]	10	1.0E-6/h [10]	10	3.0E-6/h [10]	10	3.0E-6/h [10]	3
JWR110OB	13.8 kV Feed Wire to Switchgear Unit 10 Fails	1.0E-7/h [2]	10	1.0E-6/h [10]	10	3.0E-6/h [10]	10	3.0E-6/h [10]	3
JWR112ZB	13.8 kV Feed Wire to Transformer 4 Fails	1.0E-7/h [2]	10	1.0E-6/h [10]	10	3.0E-6/h [10]	10	3.0E-6/h [10]	3

Table 7.12. Electrical Normal Power System Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
JWR120OU	13.8 kV Feed Wires Common Mode Failure	1.0E-8/h [4]	10						
JWR200VU	2400 V Feed Wires Common Mode Failure	1.0E-8/h [4]	10						
JWR203VB	Feed Wire to 2400 V Switchgear Unit 3 Fails	1.0E-7/h [2]	10	1.0E-6/h [10]	10	3.0E-6/h [10]	10	3.0E-6/h [10]	3
JWR209VB	Feed Wire to 2400 V Switchgear Unit 9 Fails	1.0E-7/h [2]	10	1.0E-6/h [10]	10	3.0E-6/h [10]	10	3.0E-6/h [10]	3
JWRN3AVB	Normal Bus No. 1 Breaker 3A Wiring Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
JWRNE2YB	Normal Power Line to E-2 Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
JWRP1AYB	Wiring to Normal Bus P-1A Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
JWRP2AYB	Wiring to Normal Bus P-2A Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
JWRP2BYB	Wiring to Normal Bus P-2B Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
JWRP3XYB	Wiring to Normal Bus P-3 Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
JWRR1AYB	Wiring to Normal Bus R-1A Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
JWRR3XYB	Wiring to Normal Bus R-3 Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
JWRS1AYB	Wiring to Normal Bus S-1A Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3

- Notes:
1. Estimate based on loss of off-site power, ~ once per year
 2. Recommended value tables, EGG-SSRE-8875, Feb. 1990
 3. Short to ground value
 4. 0.1 x individual component value
 5. Est. value; time to perform action, lack of action obvious
 6. DC motor-generator failure value
 7. Low power solid state device value
 8. Assumed on-line when malfunction occurs
 9. Recommended value tables, EEG-SSRE-8875, Feb. 1990
 10. Open circuit value

Table 7.13. Electrical Emergency Power System Data

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
EBS001UW	NO POWER AT OUTPUT OF BREAKER MCC E1 1B	1.6E-6/h [1]	3						
EBS002UW	NO POWER AT OUTPUT OF BREAKER MCC E2 1B	1.6E-6/h [1]	3						
EBSE11YC	Uncleared Overload on Emergency Bus E-1	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
EBSEE2YC	Uncleared Overload on Emergency Bus E-2	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
EBSP1BYC	Uncleared Overload on Emergency Bus P-1B	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
EBSP4EYC	Uncleared Overload Fault on Bus P-4E	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
EBSR1BYC	Uncleared Overload on Emergency Bus R-1B	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
EBSR2AYC	Uncleared Overload on Emergency Bus R-2A	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
EBSR2BYC	Uncleared Overload on Emergency Bus R-2B	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
EBSS1BYC	Uncleared Overload on Emergency Bus S-1B	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
ECB02AYB	CIRCUIT BREAKER SPURIOUS TRIP MCC R2-2A	1.0E-5/h	10						
ECB02CYB	CIRCUIT BREAKER SPURIOUS TRIP MCC R2-4C	1.0E-5/h	10						
ECB02EYO	Breaker 2E of MCC P4E Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECB03AYB	CIRCUIT BREAKER SPURIOUS TRIP MCC R2-3A	1.0E-5/h	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECB03CYO	Breaker 3C of MCC P4E Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECB04AYB	CIRCUIT BREAKER SPURIOUS TRIP	1.0E-5/h	10						
ECB04CYB	CIRCUIT BREAKER SPURIOUS TRIP MCC R2B-4C	1.0E-5/h	10						
ECB10CYO	MCC-R1B Circuit Breaker 10C Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECB10DYO	MCC-R1B Circuit Breaker 10D Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECB11BYZ	Operator Error in Connecting E-1 to R-1B	1.0E-4/d [4]	10						
ECB13DYB	CIRCUIT BREAKER SPURIOUS TRIP MCC R1B-13D	1.0E-5/h	10						
ECB13EYB	CIRCUIT BREAKER SPURIOUS TRIP MCC R1B-13E	1.0E-5/h	10						

Table 7.13. Electrical Emergency Power System Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
ECB14AYO	Emergency Bus P-1B Breaker 14AA Fails Open	1.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECB15CYO	Bus R-1B Breaker 15C Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECB15CYP	Emergency Bus R-1B Brkr 15C Fails to Open on Demand	5.0E-4/d [2]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
ECB16AYO	Emergency Bus R-1B Breaker 16A Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECB16AYP	Bus R-1 Breaker 16A Fails to Open on Demand	5.0E-4/d [2]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
ECB18AYN	Bus R-1 Breaker 8A Fails to Close on Demand	5.0E-4/d [2]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	
ECB19AYN	Bus P-1 Breaker 9A Fails to Close on Demand	5.0E-4/d [2]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	
ECB10BGO	100 KW Generator Output Breaker Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECB1PBYZ	Operator Error in Connecting E-1 to P-1B	1.0E-4/d [4]	10						
ECB21AYN	R-2A Breaker 1A Fails To Close	5.0E-4/d [2]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
ECB23BYO	Emergency Bus R-2 Tie Breaker 3B Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECB23PYP	Bus R-2 Tie Breaker 3B Fails to Open on Overload	5.0E-4/d [2]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
ECB2BAYP	Bus E-1 Breaker 2B Fails to Open on Demand	5.0E-4/d [2]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
ECB2BBYO	Emergency Bus E-1 Breaker 2B Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECB2CAYP	Bus E-1 Breaker 2C Fails to Open on Demand	5.0E-4/d [2]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
ECB2CBYO	Emergency Bus E-1 Breaker 2C Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECB2DHYZ	Operator Error Fails to Close E-2 Breaker 2D	1.0E-4/d [4]	10						
ECB2DMYN	E-2 Breaker 2D Fails to Remain Closed	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECB2FFYO	Bus P-4E Breaker 2F Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECB3AAYO	Breaker 3AA of MCC-R2A Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECB3D2YN	E-1 Breaker 3D Failure to Close	5.0E-4/d [2]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
ECB3EEYO	Bus P-4E Breaker 3EE Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3

Table 7.13. Electrical Emergency Power System Data (Cont'd)

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
ECB42EYO	Emergency Bus S-1B Contractor 42E Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECB4OBHO	400 Kw Generator Output Breaker Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECB6AAYO	Emergency Bus S-1 Breaker 6A Fails Open	1.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECB8EAYO	Emergency Bus S-1B Breaker 8E Fails Open	1.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECBA16YO	Emergency Bus P-1B Breaker 16A Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECBA16YP	Bus P-1 Breaker 16A Fails to Open on Demand	5.0E-4/d [2]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
ECBB9BYB	CIRCUIT BREAKER SPURIOUS TRIP MCC R1B-9B	1.0E-5/h	10						
ECBB9CYB	CIRCUIT BREAKER SPURIOUS TRIP R1B-9C	1.0E-5/h	10						
ECBB9DYB	CIRCUIT BREAKER SPURIOUS TRIP MCC R1B-9D	1.0E-5/h	10						
ECBD15YO	Bus P-1B Breaker 15D Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECBE1AYO	Emergency Bus E-1 Breaker 1A Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECBE1BYO	Bus E-1 Breaker 1-B Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECBE1XYP	Selected Emergency Power Loads are not Shed	3.0E-6/h [2]	10	1.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10
ECBER2YP	Selected 100 KW Emergency Power Loads Are Not Shed	3.0E-6/h [2]	10	1.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10
ECBH1AYO	E-2 Bus Breaker 1A Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	10
ECBP42YO	P-4E Supply Breaker 2C Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECBT8AYO	Bus R-1 Breaker 8A Trips Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECBT9AYO	Bus P-1 Breaker 9A Trips Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
ECBU15YO	Bus R-1B Breaker 15 A Fails Open	3.0E-7/h [2]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
EDL1SFGS	100 KW Diesel Fails to Start on Command	1.0E-2/d [6]	5						
EDL41OHU	400/100 KW Diesel Fuel Supply Common Mode Failure	5.0E-4/h [7]	10						
EDL4SFHS	400 KW Diesel Fails to Start on Command	1.0E-2/d [2]	5	3.0E-2/d	3	3.0E-2/d	3	3.0E-2/d	3
EFU12BBB	Bus E-1 Breaker 2B Trip Fuse Opens	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10

Table 7.13. Electrical Emergency Power System Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
EFU12CBB	Bus E-1 Breaker 2C Trip Fuse Opens	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
EFU15CYB	Emergency Bus R-1B Brkr 15C Trip Fuse Opens	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
EFU16ABB	Bus R-1 Breaker 16A Trip Fuse Opens	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
EFU16PBB	Bus P-1 Breaker 16A Trip Fuse Opens	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
EFUD8ABB	Bus R-1 Breaker 8A Close Fuse Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
EFUD9ABB	Bus P-1 Breaker 9A Close Fuse Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
EFUES2BO	480 Vac Emerg Switch gear 125 Vdc Bus 1 to Bus 2 Fuse Opens	1.0E-7/h [6]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
EGE1EFGW	100 KW Diesel Fails to Run	5.0E-3/h [2]	10	1.0E-3/h	10	3.0E-3/h	10	3.0E-3/h	10
EGE4EFHW	400 Kw Diesel Fails to Run	5.0E-3/h [2]	10	1.0E-3/h	10	3.0E-3/h	10	3.0E-3/h	10
EIN12BYW	Bus E-1 Breaker Trip Signal Ckt Fails	3.0E-6/h [2]	10	1.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10
EIN12CYW	Bus E-1 Breaker 2C Trip Signal Ckt Fails	3.0E-6/h [2]	10	1.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10
EIN151YW	Brkr 15C - Brkr 1A Interlock Fails and 1A Closes	3.0E-6/h [2]	10	1.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10
EIN15CYW	Emergency Bus R-1B Brkr 15C Trip Signal Ckt Fails	3.0E-6/h [2]	10	3.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10
EIN16AYW	Bus R-1 Breaker 16A Trip Signal Ckt Fails	3.0E-6/h [2]	10	1.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10
EIN16PYW	Bus P-1 Breaker 16A Trip Signal Ckt Fails	3.0E-6/h [2]	10	1.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10
EIN1SSGW	100 KW Diesel Start Signal Ckt Fails	3.0E-6/h [2]	10	1.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10
EIN21AYW	R-2A Breaker 1A Close Signal Ckt Fails	3.0E-6/h [2]	10	1.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10
EIN3D3YW	E-1 Breaker 3D Close Signal Ckt Fails	3.0E-6/h [2]	10	1.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10
EIN42NYW	Emergency Bus S-1B Cntr 42N Close Signal Ckt Fails	3.0E-6/h [2]	10	1.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10
EIN4SSHW	400 Diesel Start Signal Ckt Fails	3.0E-6/h [2]	10	1.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10
EIND8AYW	Bus R-1 Breaker 8A Close Signal Ckt Fails	3.0E-6/h [2]	10	1.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10
EIND9AYW	Bus P-1 Breaker 9A Close Signal Ckt Fails	3.0E-6/h [2]	10	1.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10

Table 7.13. Electrical Emergency Power System Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
EINE1XYW	Brkr 5C - Brkr 3D Interlock Fails and 3D Closes	3.0E-6/h [2]	10	1.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10
EOO12Z00	NO RECOVERY OF 480 V EMERGENCY BUS P-4E	1.0	1						
EOO20Z00	NO RECOVERY OF 480 V EMERGENCY BUS P-1B	1.0	1						
ERE42NYO	Emergency Bus S-1B Cntr 42N Fails to Close on Demand	5.0E-4/d [2]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	10
ESGE2EYW	Emergency Bus E-2 Interlock Fails Gen. On, 3B, 2D closed	3.0E-6/h [2]	10	1.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10
ESW004YO	E-1 Bus Disconn. Switch 4 Fails Open	1.0E-6/h [2]	10	1.0E-7/h	10				
ESWE1RYO	Bus R-1B Manual Trans. Switch E-1 Side Fails Open	1.0E-5/d [2]	10	1.0E-7/h	10				
ESWE1RYZ	Operator Fails to Effect E-1 to R-2B Line on Demand	1.0E-4/d [4]	10						
ESWP4EYB	E-1 Side of P-4E Man. Trans. Switch Fails Open	1.0E-6/h [2]	10	1.0E-7/h	10				
ESWP4EYW	Total Failure of P-4E Man Transf Switch	1.0E-7/h [8]	10						
ESWP4NYB	Normal Pwr Side of P-4E Man. Transf. Switch Fails Open	1.0E-6/h [2]	10	1.0E-7/h	10				
ESWP4NYZ	Operator Fails to Switch P-4E Man. Transf. Sw to Norm Pwr	1.0E-4/d [4]	10						
ESWR1BYO	Bus R-1B Manual Trans. Switch R-1B Side Fails Open	1.0E-6/h [2]	10	1.0E-7/h	10				
EWR11BYB	E-1 to R-1B Line Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
EWR11BYC	Overload Fault on E-1 to R-1B Line	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
EWR12BYB	Bus E-1 Breaker 2B Trip Ckt Sup. Wiring Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
EWR12CYB	Bus E-1 Breaker 2C Trip Ckt Sup. Wiring Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
EWR15CYB	Emerg Bus R-1B Brkr 15C Trip Ckt Sup. Wiring Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
EWR16AYB	Bus R-1 Breaker 16A Trip Ckt Sup. Wiring Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
EWR16PYB	Bus P-1 Breaker 16A Trip Ckt Sup. Wiring Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
EWR1SWGB	100 KW Generator Supply Wiring Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3

Table 7.13. Electrical Emergency Power System Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
EWR2E1YB	480 Vac Normal Bus No. 2 to E-1 Wiring Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
EWR2E1YC	Overload on Normal Bus No. 2 to E-1 Line	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
EWR4SWHB	400 Kw Generator Supply Wiring Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
EWRD8AYB	Bus R-1 Breaker 8A Close Ckt Sup. Wiring Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
EWRD9AYB	Bus P-1 Breaker 9A Close Ckt Sup. Wiring Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
EWRE12YB	E-1 to R-2B Line Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
EWRE1AYB	Emergency Bus E-1 Breaker 1A Wiring Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
EWREE2YB	Emergency Pwr Line to E-2 Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
EWREE2YC	Uncleared Overload on Emergency Pwr Line to E-2	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
EWRER1YC	Overload Fault on E-1 to R-2B Line	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
EWRP12YB	Line From Bus P-1B to Ups 2 Fails	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
EWRP1BYC	Overload Fault on E-1 to P-1B Line	1.0E-7/h [2]	10						
EWRP4EYB	Line From P-4E to 10 KW Rectifier Fails	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
EWRP4UYB	Line From Bus P-4E to UPS 1 Fails	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
EWRPS1YB	P-1B to S-1B Line Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
EWRPS1YC	Overload Fault on P-1B to S-1B Line	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
EWRR12YB	R-1B to R-2B Line Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
EWRR12YC	Overload Fault on R-1B to R-2B Line	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
EWRSS1YB	S-1A to S-1B Line Fails Open	1.0E-7/h [2]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
EWRSS1YC	Overload Fault on S-1A to S-1B Line	1.0E-7/h [2]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10

- Notes:
1. Based on highest bus value for 100 days divided by 2400 h
 2. Recommended value tables, EGG-SSRE-8875, Feb. 1990
 3. Short to ground value
 4. Est. value; time to perform action, lack of action obvious
 5. Low power solid state device value
 6. Recommended value tables, EEG-SSRE-8875, Feb. 1990
 7. 0.1 x failure to run value
 8. 0.1 x switch failure value

Table 7.14. Electrical Continuous Power System Data

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
UBAU2IBW	135 VDC Ups 2 Battery Loses Capacity	1.0E-5/h [1]	5	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
UBAUP1BW	Charged 135 VDC UPS 1 Battery Fails Suddenly	1.0E-5/h [1]	5	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
UBAUP2BW	Charged 135 VDC Ups No. 2 Battery Fails Suddenly	1.0E-5/h [1]	5	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
UBAUPIBW	135 VDC UPS 1 Battery Loses Capacity	1.0E-5/h [1]	5	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
UBSCP1QC	Uncleared Overload on Cont. Power Panel No. 1	1.0E-7/h [1]	10	1.0E-7/h [2]	10	3.0E-7/h [2]	10	3.0E-7/h [2]	10
UBSCP2QC	Uncleared Overload on Cont. Power Panel No. 2	1.0E-7/h [1]	10	1.0E-7/h [2]	10	3.0E-7/h [2]	10	3.0E-7/h [2]	10
UBYUP1BW	135 VDC UPS 1 Battery Has Low Charge Rate	1.0E-5/h [1]	5	1.0E-6/h		1.0E-6/h	3		
UBYUP1YW	UPS 1 Charger Circuit Fails	1.0E-5/h [1]	5	1.0E-6/h		1.0E-6/h	3		
UBYUP2BW	135 VDC Ups 2 Battery Has Low Charge Rate	1.0E-5/h [1]	5	1.0E-6/h		1.0E-6/h	3		
UBYUP2YW	Ups 2 Charger Circuit Fails	1.0E-5/h [1]	5	1.0E-6/h		1.0E-6/h	3		
UCB013QW	CIRCUIT BREAKER FAULT SPURIOUS TRIP CPP 1 BKR 3	3.0E-7/h	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
UCB022QW	CIRCUIT BREAKER FAULT SPURIOUS TRIP CPP 2 BKR 2	3.0E-7/h	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
UCB023QW	CIRCUIT BREAKER FAULT SPURIOUS TRIP CPP 2 BKR 3	3.0E-7/h	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
UCB103FW	CIRCUIT BREAKER FAULT SPURIOUS TRIP CPP R10 BKR 3	3.0E-7/h	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
UCB125QO	CPS Panel 1 Breaker 25 Fails Open	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
UCB217QO	CPS Panel 2 Breaker 17 Fails Open	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
UCBB11YO	Ups No. 1 Bypass Ckt Brkr No. 1 Fails	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
UCBB12YO	Bypass Power Input Ckt Breaker to Sbs No.1 Fails	3.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
UCBB21YO	Bypass Power Input Ckt Breaker to SBS No. 2 Fails	3.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
UCBB22YO	Ups No. 2 Bypass Ckt Brkr No. 2 Fails	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
UCBIB1YO	Ups No. 1 Isolation Breaker Fails Open	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
UCBIB2YO	Ups No. 2 Isolation Breaker Fails Open	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3

Table 7.14. Electrical Continuous Power System Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
UCBS12YO	SBS No. 1 to SBS No. 2 Circuit Breaker Fails Open	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
UCBS21YO	SBS No. 2 to SBS No. 1 Circuit Breaker Fails Open	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
UCBU11YO	135 VDC UPS 1 Battery Output Ckt Brkr Fails Open	1.0E-7/h [3]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
UCBU22YO	135 VDC Ups 2 Battery Output Ckt Brkr Fails Open	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
UCBUP1YO	Ups No. 1 Inverter Output Breaker Fails Open	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
UCBUP2YO	Ups No. 2 Inverter Output Breaker Fails Open	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
UINU12YU	Ups No. 1 and Ups No. 2 Common Mode Failure	3.0E-6/h [4]	10	1.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10
UINU2BYW	135 VDC Ups 2 Battery Low Charge Rate Undetected	3.0E-6/h [1]	10	1.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10
UINV18YL	135 VDC UPS 1 Battery Capacity Loss Not Detected	1.0E-3/d [6]	30						
UINV18YW	135 VDC UPS 1 Battery Low Charge Rate Not Detected	3.0E-6/h [1]	10	1.0E-6/h [5]	10	3.0E-6/h [5]	10	2.0E-6/h [5]	10
UINV2BYL	135 VDC Ups 2 Battery Capacity Loss Not Detected	1.0E-3/d [6]	30						
UIVUP1YW	Ups No. 1 Inverter Fails	5.0E-6/h [1]	5			1.0E-4/h	3		
UIVUP2YW	Ups No. 2 Inverter Fails	5.0E-6/h [1]	5			1.0E-4/h	3		
USWSS1YW	Ups No. 1 Static Bypass Switch or SBS Control Fails	3.0E-6/h [1]	10	1.0E-6/h [7]	10	3.0E-6/h [7]	10	3.0E-6/h [7]	10
USWSS2YW	Ups No. 2 Static Bypass Switch or SBS Control Fails	3.0E-6/h [1]	10	1.0E-6/h [7]	10	3.0E-6/h [7]	10	3.0E-6/h [7]	10
USWUP1YO	Ups No. 1 Manual Bypass Switch Fails Open	1.0E-6/h [1]	10	1.0E-7/h	10				
USWUP2YO	Ups No. 2 Manual Bypass Switch Fails Open	1.0E-6/h [1]	10	1.0E-7/h	10				
UTRUBTYW	Ups Bypass 480 V / 120 V Transformer Fails	1.0E-6/h [1]	10	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
UVRBSSYW	Ups Bypass Surge Suppressor Fails	3.0E-6/h [1]	10	1.0E-6/h [7]	10	3.0E-6/h [7]	10	3.0E-6/h [7]	10
UWRS12YB	SBS No. 1 to SBS No. 2 Wiring Fails	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
UWRS21YB	SBS No. 2 to SBS No.1 Wiring Fails	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3

Table 7.14. Electrical Continuous Power System Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
UWRUBWYB	Ups Bypass Supply Wiring Fails	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
UWRUC1YB	Ups No. 1 to CPP No. 1 Wiring Fails	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
UWRUC2YB	Ups No. 2 to CPP No. 2 Wiring Fails	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3

- Notes:
1. Recommended value tables, EGG-SSRE-8875, Feb. 1990
 2. Short to ground value
 3. Recommended value tables, EEG-SSRE-8875, Feb. 1990
 4. EEG-SSRE-8875, sync logic module, low power solid state
 5. Low power solid state device value
 6. Est. value; operator does not detect deficiency upon testing
 7. High power solid state device value

Table 7.15. Electrical 115V DC System Data

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
LBA12CBW	125 VDC Station Battery Loses Capacity	1.0E-5/h [1]	5	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
LBA12MBW	Charged 125 VDC Station Battery Fails Suddenly	1.0E-5/h [1]	5	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
LBS001BU	125 VDC Distr. Center Bus Pwr Sources Common Mode Failure	1.0E-7/h [2]	10						
LBS002BU	125 VDC/480 Vac Norm Bus Common Mode Loss of Power Sources	1.0E-7/h [2]	10						
LBS003BU	125 VDC/480 Vac Emer Bus Common Mode Loss of Power Sources	1.0E-7/h [2]	10						
LBS004BU	125 VDC/2.4 KVAC Bus Common Mode Loss of Power Sources	1.0E-7/h [2]	10						
LBS005BU	125 VDC/13.8 KVAC Bus Common Mode Loss of Power Sources	1.0E-7/h [2]	10						
LBS121BC	First 125 VDC Ground Fault Occurs	1.0E-7/h [1]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
LBS122BC	Second 125 VDC Ground Fault Occurs	1.0E-7/h [1]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
LBS124BC	Uncleared Overload on 125 VDC/480 Vac Bus	1.0E-7/h [1]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
LBS12EBC	Uncleared Overload on 125 VDC Emergency Bus	1.0E-7/h [1]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
LBS12MBA	Line-to-Line Short on 125 VDC Distr Center Bus	1.0E-7/h [1]	10	1.0E-8/h	10	3.0E-8/h	10	3.0E-8/h	10
LBS133BC	Uncleared Overload on 125 VDC/13.8 KV Bus	1.0E-7/h [1]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
LBS244BC	Uncleared Overload on 125 VDC/2.4 KV Bus	1.0E-7/h [1]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
LBSEB1BC	Uncleared Overload on 125 VDC/480 Vac Emerg Bus No. 1	1.0E-7/h [1]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
LBSEB2BC	Uncleared Overload on 125 VDC/480 Vac Emerg Bus No. 2	1.0E-7/h [1]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
LBY12CBW	125 VDC Station Battery Has Failed Charger	1.0E-5/h [1]	5	1.0E-6/h		1.0E-6/h	3		
LCBD06BO	125 VDC Distr Center Bus Breaker 6 Fails Open	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
LCBD07BO	125 VDC Distr Center Bus Breaker 7 Fails Open	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
LCBD10BO	125 VDC Distr Center Bus Breaker 10 Fails Open	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
LCBD11BO	125 VDC Distr Center Bus Breaker 11 Fails Open	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3

Table 7.15. Electrical 115V DC System Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
LCBE11BO	125 VDC Emergency Bus Breaker 1 Fails Open	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
LCBE12BO	125 VDC Emergency Bus Breaker 2 Fails Open	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
LCBE13BO	125 VDC Emergency Bus Breaker 3 Fails Open	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
LCBE14BO	125 VDC Emergency Bus Breaker 4 Fails Open	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
LCBMCCBC	Uncleared Overload on 125 VDC MCC Bus	1.0E-7/h [1]	10	1.0E-7/h [3]	10	3.0E-7/h [3]	10	3.0E-7/h [3]	10
LCBMCCBO	125 VDC Distr Center Bus Breaker 5 Fails Open	3.0E-7/h [1]	10	1.0E-5/h	10	3.0E-5/d	10	3.0E-5/d	3
LDC12EBW	10 KW Rectifier Fails	1.0E-5/h [1]	5	1.0E-6/h [4]	10	3.0E-6/h [4]	10	3.0E-6/h [4]	10
LFU100BB	100 KW Diesel Start Sup. Fuse Open	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
LFU1CDVB	Breaker 1CD Trip Ckt Fuse Fails Open	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
LFU21ABB	R-2A Breaker 1A 125 VDC Sup. Wire/Fuse Opens	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
LFU3D4BB	E-1 Breaker 3D 125 Vdc Sup. Wire/Fuse Opens	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
LFU400BB	400 Kw Diesel Start Sup. Fuse Open	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
LFU4BCVB	Breaker 4BC Trip Ckt Fuse Fails Open	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
LFU7CDVB	Breaker 7CD Trip Ckt Fuse Fails Open	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	10
LIN12CBL	125 VDC Station Battery Capacity Loss Not Detected	1.0E-3/d [5]	30						
LIN12CBW	125 VDC Station Battery Low Charge Rate Not Detected	3.0E-6/h [1]	10	1.0E-6/h [6]	10	3.0E-6/h [6]	10	2.0E-6/h [6]	10
LIN12MBW	First 125 VDC Ground Fault Alarm CKT Fails	3.0E-6/h [1]	10	1.0E-6/h [6]	10	3.0E-6/h [6]	10	2.0E-6/h [6]	10
LIN12MBZ	Operator Error Ignores First 125 VDC Gnd Fault Alarm	1.0E-4/d [7]	10						
LSW12TBB	125 VDC Throwover Sw. Sta. Battery Side Fails	1.0E-6/h [1]	10	1.0E-7/h	10				
LSWMD1BO	125 VDC MTS 1 Distr Cent Bus Side Fails Open	1.0E-6/h [1]	10	1.0E-7/h	10				
LSWMD2BO	125 VDC MTS 2 Distr Cent Bus Side Fails Open	1.0E-6/h [1]	10	1.0E-7/h	10				
LSWMD3BO	125 VDC MTS 3 Distr Cent Bus Side Fails Open	1.0E-6/h [1]	10	1.0E-7/h	10				

Table 7.15. Electrical 115V DC System Data (Cont'd)

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
LSWMD4BO	125 VDC MTS 4 Distr Center Bus Side Fails Open	1.0E-6/h [1]	10	1.0E-7/h	10				
LSWME1BO	125 VDC MTS 1 Emergency Bus Side Fails Open	1.0E-6/h [1]	10	1.0E-7/h	10				
LSWME1BZ	Operator Fails to Close 125 VDC MTS 1 Emerg Bus Side	1.0E-4/d [8]	10						
LSWME2BO	125 VDC MTS 2 Emergency Bus Side Fails Open	1.0E-6/h [1]	10	1.0E-7/h	10				
LSWME2BZ	Operator Fails to Close 125 VDC MTS 2 Emerg Bus Side	1.0E-4/d [8]	10						
LSWME3BO	125 VDC MTS 3 Emergency Bus Side Fails Open	1.0E-6/h [1]	10	1.0E-7/h	10				
LSWME3BZ	Operator Fails to Close 125 VDC MTS 3 Emerg Bus Side	1.0E-4/d [8]	10						
LSWME4BO	125 VDC MTS 4 Emergency Bus Side Fails Open	1.0E-6/h [1]	10	1.0E-7/h	10				
LSWME4BZ	Operator Fails to Close 125 VDC MTS 4 Emerg Bus Side	1.0E-4/d [8]	10						
LSWUPTBB	125 VDC Throwover SW UPS 1 Bat Side Fails	1.0E-6/h [1]	10	1.0E-7/h	10				
LSWUPTBZ	Operator Error Fails to Switch to UPS 1 Bat	1.0E-4/d [8]	10						
LWR12MBB	Line From 125V Throw over SW to Distr Cen Bus Fails Open	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
LWR12OVb	4BC-12 Fault Detection Ckt Supply Wiring Fails Open	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
LWR12TBB	Line From Sta. Battery to 125 VDC Throwover Switch Fails	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
LWR133BB	Wiring From 125 VDC MTS 3 to Bus Fails Open	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
LWR1CDVB	1CD Fault Detection Ckt Supply Wiring Fails Open	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
LWR21ABB	R-2A Breaker 1A 125 VDC Sup. Wiring Opens	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
LWR21OVb	4BC-21 Fault Detection Ckt Supply Wiring Fails Open	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	
LWR244BB	Wiring From 125 VDC MTS 4 to Bus Fails Open	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3
LWR3D4BB	E-1 Breaker 3D 125 Vdc Sup. Wiring Opens	1.0E-7/h [1]	10	1.0E-6/h	10	3.0E-6/h	10	3.0E-6/h	3

Table 7.16. Air System Data

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
IAV695LQ	Isolation Valve R1-VC-695 Fails Closed	2.0E-6/h	3						
ICM001YW	Primary Inst. Air Compressor, Aftercooler or Motor Failure	3.0E-5/h [1]	10					6.0E-6/h [2]	
ICM002YW	Standby Inst. Air Compressor, Aftercooler or Motor Failure	3.0E-5/h [1]	10					6.0E-6/h [2]	
ICM002YX	Standby Inst. Air Compressor Down for Maintenance	6.0E-3/d [3]	10						
ICM003YW	Plant Air Compressor or Motor Failure	6.0E-6/h [1]	10					6.0E-6/h [2]	
ICM003YX	Plant Air Compressor Down for Maintenance	6.0E-3/d [3]	10						
ICM004LW	Emergency Air Compressor or Motor Failure	3.0E-5/h [1]	10					6.0E-6/h [2]	
ICM004LX	Emergency Air Compressor Down for Maintenance	6.0E-3/d [3]	10						
ICZ011YO	Primary Cont. Main Switch MAN Contacts Fail Open	1.0E-7/h [4]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
ICZ012YO	Primary Cont. Overload Contacts (1/3) Fail Open	3.0E-7/h [5]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
ICZ021YO	Standby Cont. Main Switch AUTO Contacts Fail Open	1.0E-7/h [4]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
ICZ022YO	Standby Cont. Overload Contacts (1/3) Fail Open	3.0E-7/h [5]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
ICZ031YO	Controller Main Switch Contacts Fail Open	1.0E-7/h [4]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
ICZ032YO	Controller Overload Contacts(1/2) Fail Open	2.0E-7/h [6]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
ICZ041LO	Controller Main Switch STANDBY Contacts Fail Open	1.0E-7/h [4]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
ICZ042LO	Controller Overload Contacts (1/2) Fail Open	2.0E-7/h [6]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
ICZ092YO	Controller Compressor Motor Contacts Fail Open	1.0E-7/h [4]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
ICZ094LO	Controller Compressor Motor Contacts Fail Open	1.0E-7/h [4]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
ICZ190YO	Primary Cont. Comp. Motor Contacts Fail Open	1.0E-7/h [4]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	
ICZ290YO	Standby Cont. Comp. Motor Contacts Fail Open	1.0E-7/h [4]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
IMS002YS	Standby Compressor Motor Fails to Start	3.0E-5/d [1]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3

Table 7.16. Air System Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
IMS003YS	Plant Air Compressor Motor Fails to Start	3.0E-5/d [1]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3
IMS004LS	Emergency Air Compressor Motor Fails to Start	3.0E-5/d [1]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3
IOO01A00	Plant/Instrument Air Emergency Crossover Open	1.0	1						
IOO01B00	Plant/Instrument Air Emergency Crossover Closed	0.0	1						
IOO02Z00	Instrument Air System Partially Isolated	1.0	1						
IPP001YF	Main Instrument Air System Distribution Line Failure	8.0E-9/h [7]	30	1.0E-9/h [8]				8.0E-9/h [8]	30
IPP004YF	Plant Air Aftercooler to Receiver Line Failure	8.0E-9/h [7]	30	1.0E-9/h [8]				8.0E-9/h [8]	30
IPP005YF	Plant Air Receiver to Dryer Line Failure	8.0E-9/h [7]	30	1.0E-9/h [8]				8.0E-9/h [8]	30
IPP012YF	Plant Air Filter to Distribution Line Failure	8.0E-9/h [7]	30	1.0E-9/h [8]				8.0E-9/h [8]	30
IPP110YF	Standby Aftercooler to Receiver Line Failure	8.0E-9/h [7]	30	1.0E-9/h [8]				8.0E-9/h [8]	30
IPP111YF	Primary Aftercooler to Receiver Line Failure	8.0E-9/h [7]	30	1.0E-9/h [8]				8.0E-9/h [8]	30
IPP113YF	Instrument Air Receiver to Main Filter Line Failure	8.0E-9/h [7]	30	1.0E-9/h [8]				8.0E-9/h [8]	30
IPP117YF	Instrument Air Main Filter to Dryer Line Failure	8.0E-9/h [7]	30	1.0E-9/h [8]				8.0E-9/h [8]	30
IPP122YF	Instrument Air Dryer to Filter Line Failure	8.0E-9/h [7]	30	1.0E-9/h [8]				8.0E-9/h [8]	30
IPP125LF	Emergency Air Filter/Dryer Bypass Line Failure	8.0E-9/h [7]	30	1.0E-9/h [8]				8.0E-9/h [8]	30
IPP132YF	Sec. Filter to Instrument Air Distribution Line Failure	8.0E-9/h [7]	30	1.0E-9/h [8]				8.0E-9/h [8]	30
IPP201YF	Secondary Instrument Air Distribution Line Leakage	8.0E-9/h [7]	30	1.0E-9/h [8]				8.0E-9/h [8]	30
IPP301YF	Main Plant Air Distribution Line Failure	8.0E-9/h [7]	30	1.0E-9/h [8]				8.0E-9/h [8]	30
IPP501LF	Emergency Air System Distribution Line Failure	8.0E-9/h [7]	30	1.0E-9/h [8]				8.0E-9/h [8]	30
IPP634LF	Emergency Air Compressor to Receiver Line Failure	8.0E-9/h [7]	30	1.0E-9/h [8]				8.0E-9/h [8]	30

Table 7.16. Air System Data (Cont'd)

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
IPP716YF	Plant Air Compressor to Aftercooler Line Failure	8.0E-9/h [7]	30	1.0E-9/h [8]				8.0E-9/h [8]	30
IPP718YF	Primary Compressor to Aftercooler Line Failure	8.0E-9/h [7]	30	1.0E-9/h [8]				8.0E-9/h [8]	30
IPP720YF	Standby Compressor to Aftercooler Line Failure	8.0E-9/h [7]	30	1.0E-9/h [8]				8.0E-9/h [8]	30
IPP731YF	Plant Air Dryer to Filter Line Failure	8.0E-9/h [7]	30	1.0E-9/h [8]				8.0E-9/h [8]	30
IPR004LB	Emergency Air Controller Pressure Regulator Fails	1.0E-6/h [1]	10	1.0E-3/d [9]	3	1.0E-3/d [9]	3	1.0E-3/d [9]	3
IPR011YB	Primary Compressor Oil Pressure Sensor Fails Low	1.0E-6/h [1]	10	1.0E-7/h	10				
IPR012YB	Primary Comp. Controller Pressure Regulator Fails	1.0E-6/h [1]	10	1.0E-3/d [9]	3	1.0E-3/d [9]	3	1.0E-3/d [9]	3
IPR021YB	Standby Compressor Oil Pressure Sensor Fails Low	1.0E-6/h [1]	10	1.0E-7/h	10				
IPR022YB	Standby Comp. Controller Pressure Regulator Fails	1.0E-6/h [1]	10	1.0E-3/d [9]	3	1.0E-3/d [9]	3	1.0E-3/d [9]	3
IPR031YB	Compressor Oil Pressure Sensor Fails Low	1.0E-6/h [1]	10	1.0E-7/h	10				
IPR032YB	Plant Air Controller Pressure Regulator Fails	1.0E-6/h [1]	10	1.0E-3/d [9]	3	1.0E-3/d [9]	3	1.0E-3/d [9]	3
IPR042LB	Emergency Air Oil Pressure Sensor Fails Low	1.0E-6/h [1]	10	1.0E-7/h	10				
IRR635LF	Emergency Air Receiver R1-T-1635 Tank Failure	1.0E-10/h [10]	30						
IRR717YF	Plant Air Receiver P3-T-717 Tank Failure	1.0E-10/h [10]	30						
IRR719YF	Instrument Air Receiver P3-T-719 Tank Failure	1.0E-10/h [10]	30						
IRV001YN	Instrument Air Dryer Relief Valve Fails to Close	3.0E-3/d [1]	5	2.0E-2/d	3	2.0E-2/d	3	1.0E-2/d	3
IRV003YN	Instrument Air Receiver Relief Valve Fails to Close	3.0E-3/d [1]	5	2.0E-2/d	3	2.0E-2/d	3	1.0E-2/d	3
IRV004YN	Primary Separator Relief Valve Fails to Close	3.0E-3/d [1]	5	2.0E-2/d	3	2.0E-2/d	3	1.0E-2/d	3
IRV005YN	Primary Compressor Relief Valve Fails to Close	3.0E-3/d [1]	5	2.0E-2/d	3	2.0E-2/d	3	1.0E-2/d	3
IRV006YN	Plant Air Receiver Relief Valve Fails to Close	3.0E-3/d [1]	5	2.0E-2/d	3	2.0E-2/d	3	1.0E-2/d	3

Table 7.16. Air System Data (Cont'd)

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
IRV007YN	Standby Compressor Relief Valve Fails to Close	3.0E-3/d [1]	5	2.0E-2/d	3	2.0E-2/d	3	1.0E-2/d	3
IRV009YN	Plant Air Moisture Separator Relief Valve Fails to Close	3.0E-3/d [1]	5	2.0E-2/d	3	2.0E-2/d	3	1.0E-2/d	3
IRV013LN	Emergency Air Receiver Relief Valve Fails to Close	3.0E-3/d [1]	5	2.0E-2/d	3	2.0E-2/d	3	1.0E-2/d	3
ITE011YB	Primary Comp. Water Temp. Sensor Fails High	3.0E-6/h [1]	10					6.0E-5/h [2]	
ITE012YB	Primary Compressor Air Temp. Sensor Fails High	3.0E-6/h [1]	10					6.0E-5/h [2]	
ITE021YB	Standby Comp. Water Temp. Sensor Fails High	3.0E-6/h [1]	10					6.0E-5/h [2]	
ITE022YB	Standby Compressor Air Temp. Sensor Fails High	3.0E-6/h [1]	10					6.0E-5/h [2]	
ITE031YB	Compressor Water Temp. Sensor Fails High	3.0E-6/h [1]	10					6.0E-5/h [2]	
IXT001YW	Primary Comp. 480/120 V Transformer Fails	1.0E-6/h [1]	10	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
IXT002YW	Standby Comp. 480/120 V Transformer Fails	1.0E-6/h [1]	10	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
IXT003YW	Controller 480/120 V Transformer Fails	1.0E-6/h [1]	3	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
IXT004LW	Controller 480/120 V Transformer Fails	1.0E-6/h [1]	10	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
IXV120LQ	Instrument Air Valve RA-120 Fails to Remain Open	1.0E-7/h	3						
IXV120LX	Instrument Air Valve RA-120 Closed	0.0	1						
IXV121LQ	Instrument Air Valve RA-121 Fails to Remain Open	1.0E-7/h	3						
IXV121LX	Instrument Air Valve RA-121 Closed	0.0	1						
IXV134YQ	Instrument Air Valve PA-134 Fails to Remain Open	1.0E-7/h	3						
IXV134YX	Instrument Air Valve PA-134 Closed	0.0	1						
IXV149YQ	Instrument Air Valve PA-149 Fails to Remain Open	1.0E-7/h	3						
IXV149YX	Instrument Air Valve PA-149 Closed	0.0	1						
IXV309YQ	Instrument Air Valve PA-309 Fails to Remain Open	1.0E-7/h	3						
IXV309YX	Instrument Air Valve PA-309 Closed	0.0	1						

Table 7.16. Air System Data (Cont'd)

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- Notes:
1. Recommended value tables, EGG-SSRE-8875, Feb. 1990
 2. Taken from Table 7-3, SAND-87-7138
 3. Probability based on 1 hr/week maintenance
 4. From CRBRP
 5. From CRBRP; 3 x individual component value
 6. From CRBRP; 2 x individual component value
 7. Generic
 8. Value for pipe < 3 in. diameter
 9. Solenoid valve failure to operate
 10. Estimated value
 11. Recommended value tables, EGG-SSRE-8875, Feb. 1990

Table 7.17. Reactor Water Systems Data

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
WCV001YQ	Reactor Aux. Primary Pump Check Valve Fails Closed	5.0E-9/h [1]	10	1.0E-7/h	10	3.0E-7/h	10	1.0E-4/d	3
WCV002YQ	Reactor Aux. Standby Pump Check Valve Fails Closed	5.0E-9/h [1]	10	1.0E-7/h	10	3.0E-7/h	10	1.0E-4/d	3
WCZ101YO	Primary Controller RUN Contacts Fail Open	1.0E-7/h [2]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
WCZ102YO	Primary Controller OL Contacts (1/2) Fail Open	2.0E-7/h [3]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
WCZ115YO	Primary Controller Pump Motor Contacts Fail Open	1.0E-7/h [2]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
WCZ201YO	Standby Cont. STANDBY Contacts Fail Open	1.0E-7/h [2]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
WCZ202YO	Standby Controller OL Contacts (1/2) Fail Open	2.0E-7/h [3]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
WCZ215YO	Standby Controller Pump Motor Contacts Fail Open	1.0E-7/h [2]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
WFL095YE	Reactor Aux. Water CUNO Filter Blocked	3.0E-5/h [4]	10	1.0E-5/h	10	3.0E-5/h	10		
WGV342PQ	RX AUX WATER FLOW CONTROL FAILURE P17-VC-342	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
WGV343PQ	RX AUX WATER FLOW CONTROL FAILURE P17-VC-343	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
WGV701PQ	RX AUX WATER FLOW CONTROL FAILURE P8-VC-701	3.0E-6/h	10	1.0E-6/h	10	3.0E-6/h	10	2.0E-6/h	10
WMS002YW	Standby Pump Motor Starter Failure	3.0E-5/d [1]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3
WPM001YW	Reactor Aux. Primary Pump Fails	3.0E-5/h [4]	10	1.0E-5/h	10	3.0E-5/h	10	3.0E-5/h	10
WPM002YW	Reactor Aux. Standby Pump Fails	3.0E-5/h [4]	10	1.0E-5/h	10	3.0E-5/h	10	3.0E-5/h	10
WPM002YX	Reactor Aux. Standby Pump Down for Maintenance	6.0E-3 [5]	10						
WPP006OF	Break in Supply Line from Cooling Tower	8.0E-9/h [6]	30	1.0E-9/h [7]				8.0E-9/h [7]	30
WPP07AOF	Raw Water Supply Line Failure	8.0E-9/h [6]	30	1.0E-9/h [7]				8.0E-9/h [7]	30
WPP07BOF	Insufficient Water Return to Cooling Tower	8.0E-9/h [8]	30	1.0E-9/h [7]				8.0E-9/h [7]	30
WPP211YF	Primary Air Compressor Cooling Water Line Failure	8.0E-9/h [6]	30	1.0E-9/h [7]				8.0E-9/h [7]	30
WPP238YF	Standby Air Compressor Cooling Water Line Failure	8.0E-9/h [6]	30	1.0E-9/h [7]				8.0E-9/h [7]	30
WPP410YF	Plant Air Compressor Cooling Water Line Failure	8.0E-9/h [6]	30	1.0E-9/h [7]				8.0E-9/h [7]	30

Table 7.17. Reactor Water Systems Data (Cont'd)

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EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
WPR001YO	Water Pressure Low Switch Fails to Remain Closed	1.0E-6/h [1]	10	1.0E-7/h	10				
WPR002YN	Water Pressure Low Switch Fails to Close	3.0E-7/h [1]	10	1.0E-4/d	3	1.0E-4/d	3	1.0E-4/d	3
WPR002YO	Water Pressure Low Switch Fails to Remain Closed	1.0E-6/h [1]	10	1.0E-7/h	10				
WXT101YW	Primary Cont. 480/120 V Transformer Fails	1.0E-6/h [1]	3	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
WXT201YW	Standby Cont. 480/120 V Transformer Fails	1.0E-6/h [1]	10	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3

- Notes:
1. Recommended value tables, EGG-SSRE-8875, Feb. 1990
 2. From CRBRP
 3. From CRBRP; 2 x individual component value
 4. From IREP
 5. Probability based on 1 hr/week maintenance
 6. Generic
 7. Value for pipe < 3 in. diameter
 8. Estimated value, generic, for pipes < 3 in. diameter

Table 7.18. Plant Water Systems Data

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
QCV001YQ	Plant Water Primary Pump Check Valve Plugs	5.0E-9/h [1]	10	1.0E-7/h	10	3.0E-7/h	10	1.0E-4/d	3
QCV002YP	Plant Water Standby Pump Check Valve Fails to Open	5.0E-5/d [1]	10	1.0E-4/d	3	1.0E-4/d	3	1.0E-4/d	3
QCV002YQ	Plant Water Standby Pump Check Valve Plugs	5.0E-9/h [1]	10	1.0E-7/h	10	3.0E-7/h	10	1.0E-4/d	3
QCZ101YO	Primary Controller RUN Contacts Fail Open	1.0E-7/h [2]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
QCZ102YO	Primary Controller OL Contacts (1/2) Fail Open	2.0E-7/h [3]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
QCZ117YO	Primary Controller Pump Motor Contacts Fail Open	1.0E-7/h [2]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
QCZ201YO	Standby Cont. STANDBY Contacts Fail Open	1.0E-7/h [2]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
QCZ202YO	Standby Controller OL Contacts (1/2) Fail Open	2.0E-7/h [3]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
QCZ217YO	Standby Controller Pump Motor Contacts Fail Open	1.0E-7/h [2]	10	1.0E-7/h	10	3.0E-4/d	10	3.0E-4/d	3
QFL092YE	Plant Water CUNO Filter Blocked	3.0E-5/h [4]	10	1.0E-5/h	10	3.0E-5/h	10		
QMS002YW	Standby Pump Motor Starter Failure	3.0E-5/d [1]	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3
QPM001YW	Plant Water Primary Pump Fails	3.0E-5/h [4]	10	1.0E-5/h	10	3.0E-5/h	10	3.0E-5/h	10
QPM002YW	Plant Water Standby Pump Fails	3.0E-5/h [4]	10	1.0E-5/h	10	3.0E-5/h	10	3.0E-5/h	10
QPM002YX	Plant Water Standby Pump Down for Maintenance	6.0E-3 [5]	10						
QPP006OF	Break in Supply Line from Cooling Tower	8.0E-9/h [6]	30	1.0E-9/h [7]				8.0E-9/h [7]	30
QPR001YO	Water Pressure Low Switch Fails to Remain Closed	1.0E-6/h [1]	10	1.0E-7/h	10				
QPR002YN	Water Pressure Low Switch Fails to Close	3.0E-7/d [1]	10	1.0E-4/d	3	1.0E-4/d	3	1.0E-4/d	3
QPR002YO	Water Pressure Low Switch Fails to Remain Closed	1.0E-6/h [1]	10	1.0E-7/h	10				
QXT101YW	Primary Cont. 480/120 V Transformer Fails	1.0E-6/h [1]	10	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3
QXT201YW	Standby Cot. 480/120 V Transformer Fails	1.0E-6/h [1]	10	1.0E-6/h	3	1.0E-6/h	3	1.0E-6/h	3

Notes: 1. Recommended value tables, EGG-SSRE-8875, Feb. 1990

2. From CRBRP

3. From CRBRP; 2 x individual component value

4. From IREP

5. Probability based on 1 hr/week maintenance

6. Generic

7. Value for pipe < 3 in. diameter

Table 7.19. Argon Systems Data

PAGE 1

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
OTG001OE	VENT LINE TO IHX OUTLET CLOSED	2.5E-7/h	10						
OTG002OE	VENT LINE TO IHX INLET CLOSED	1.0E0/d	1						

Table 7.20. Reactor Building Isolation Data

PAGE 1

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
KAU000PH	REACTOR BLD ISOLATION VALVE ACCUMULATORS FAIL	5.0E-6/h	10						
KAV23ASQ	ISOLATION VALVE 323A FAILS TO REMAIN OPEN R13-VR-323A	6.4E-8/h [1]	10	1.0E-7/h	3				
KAV23BSP	ISOLATION VALVE 323B FAILS TO OPEN R13-VR-323B	2.0E-3/d	10	1.0E-3/d	10	3.0E-3/d	10	3.0E-3/d	3
KAV23BSQ	ISOLATION VALVE 323B FAILS TO REMAIN OPEN R13-VR-323B	1.0E-7/h	10	1.0E-7/h	3				
KAV320SQ	ISOLATION VALVE 320 FAILS TO REMAIN OPEN R13-VR-320	6.4E-8/h [1]	10	1.0E-7/h	3				
KDM000UW	REACTOR BUILDING ISOLATION SIGNAL	1.0E-4/h [2]	30						
KIN003UW	REACTOR BUILDING TOTAL ISOLATION	2.3E-6/h [3]	10						

- Notes:
1. Air Operated Valve Minus Internal HVAC Valve
 2. Assumes One Partial Isolation in Four 100 Day Runs
 3. Based on no inadvertent actuations in 25 yr

Table 7.21. Safety Rod Data

PAGE 1

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
SCD100RG	Forces due to Bowing and Binding	6.0E-5/d [1]	20						
SCD200RW	Failure of Rise Mechanism, Connecting Shafts, Bearings	6.0E-5/d [1]	20						
SCD300RI	Forces due to Foreign Materials in Sodium	6.0E-5/d [1]	20						
SCL000RP	Magnetic Clutch Fails to Release SRs	2.0E-4/d [2]	20					3.0E-4/d [3]	5
SSR100RG	Forces due to Bowing and Binding	6.0E-5/d [1]	20						
SSR300RI	Forces due to Foreign Materials in Sodium	6.0E-5/d [1]	20						
SSW300CP	3MPB Mech Parts Fail	2.0E-6/d [4]	10						
SSW30ACP	3MPB Switch of String A Fails to Open	1.0E-5/d [3]	5	1.0E-4/d	10	3.0E-5/d	10	1.0E-5/d [3]	5
SSW30BCP	3MPB Switch of String B Fails to Open	1.0E-5/d [3]	5	1.0E-4/d	10	3.0E-5/d	10	1.0E-5/d [3]	5

- Notes:
1. Based on combined CCF Prob of 3.0E-4/d for the 2 SRs
 2. Clutch failure to release SRs less probable than to engage
 3. Recommended value tables, EGG-SSRE-8875, Feb. 1990
 4. Based on Engineering Judgement

Table 7.22. Test and Maintenance Unavailabilities

		Value Used 3/90 [1]	
Event Name	Description	Failure Rate	Error Rate
ICM002YX	Standby Inst. Air Compressor Down for Maintenance	$6.0 \cdot 10^{-3}$	10
ICM004LX	Emergency Air Compressor Down for Maintenance	$6.0 \cdot 10^{-3}$	10
QPM002YX	Plant Water Standby	$6.0 \cdot 10^{-3}$	10
WPM002YX	Reactor Aux. Standby Pump Maintenance	$6.0 \cdot 10^{-3}$	10

Notes: Probability based on 1 hr/week maintenance.

Table 7.23. Plant-Specific Component Failure Data For Reactor Shutdown System

Event Name ⁽²⁾	Failure Rate ⁽¹⁾	Event Description	Operation Experience
RRE1A1QP, etc. RRE2A1QP, etc. RRE3C1QP, etc. RRECC1QP, etc. RRECB1QP, etc. RRESROQP, etc.	$1.0 \cdot 10^{-7} \text{ hr}^{-1}$	Relay Armature jams and fails to move to open position	There are about 250 clark relays, model 5UU, in the plant. Five unsafe relay failures have been noticed.
RND2AORW, RND2BORW, RND2CORW	$3.0 \cdot 10^{-6} \text{ hr}^{-1}$	Guarded fission chamber of a wide-range channel fails low	There are 3 similar chambers in the plant. One unsafe failure was recorded, CREDO #EBR850006.
RCA2AORW, RCA2BORW, RCA2CORW	$9.0 \cdot 10^{-7} \text{ hr}^{-1}$	Cables and connectors of a wide-range channel reduce or eliminate signal from detector	There are 3 similar cables in the plant. No failure.
RBI2A1QW, etc. RBI2A2QW, etc.	$2.0 \cdot 10^{-7} \text{ hr}^{-1}$	High-flux and low-volt bistables of a wide-range channel fail	There are 12 similar components in the plant circuits. No failure
RAM2AOQW, RAM2BOQW, RAM2COQW	$1.0 \cdot 10^{-6} \text{ hr}^{-1}$	Current amplifier of a wide-range channel fails	There are 3 similar components in the plant. One failure was detected and overcome during operation.
RDC2A1QW, RDC2B1QW, RDC2C1QW	$3.0 \cdot 10^{-6} \text{ hr}^{-3}$	$\pm 15 \text{ Vdc}$ power supply in a wide-range channel fails	There is a large number of these components in the wide-range channels. Failures have occurred but not in the scram portion of the channels. Estimate.
RDC2A2QW, RDC2B2QW, RDC2C2QW,	$9.0 \cdot 10^{-7} \text{ hr}^{-1}$	High Volt DC power supply in a wide-range channel fails	There are 3 similar units in the plant. No failure.

Table 7.23. Plant-Specific Component Failure Data For Reactor Shutdown System (Cont'd)

RVIIA1QW, etc. RVI3COQW, etc.	$5.0 \cdot 10^{-7} \text{ hr}^{-1}$	MV/I converter in a low-flow channel or SOT channel fails	There is a large number of this component, Ronan X-81, in the plant circuits. Estimated accum. hrs of operation up to 1987 = $9.75\text{E}+5 \text{ H}$. No failure.
RBI1AOQW, etc.	$5.0 \cdot 10^{-7} \text{ hr}^{-1}$	Trip bistable in a low-flow channel or SOT channel fails	There is a large number of this component, Ronan X-82, in the plant circuits. Estimated accum. hrs. of operation up to 1987 = $9.32\text{E}+5 \text{ H}$. No failure.

NOTES:

- (1) Mean failure rate = $(2N+1)/(2T)$
N is number of failures,
T is estimated hours of power operation, with $T = 1.80\text{E}+5 \text{ H}$ for one component
- (2) An example for a basic event from each channel or string is provided.

Table 7.24. Special Data

PAGE 1

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
XHE001RF	IXH TUBE FAILURE	1.0E-6/h [1]	10						
XOO00Z00	DRAIN VALVES OPENED	1.0	1						
XOO11Z00	IHX OUTLET DRAIN VALVE OPEN	1.0	1						
XOO22Z00	SURGE TANK DRAIN VALVE OPEN	1.0	1						
XOO33Z00	EM PUMP DRAIN VALVE OPEN	1.0	1						
XOOPMZ00	PUMP REMAINS ON	1.0	1						
XPP090LF	PIPE BREACH IN PURIFICATION SYSTEM	3.0E-7/h [1]	10						
XPP100RF	OUTER BOUNDARY BREACH	1.0E-6/h [2]	10						
XPP101RF	INNER BOUNDARY BREACH	1.0E-1/d [1]	3						
XRV001NK	PRESSURE RELIEF ALLOWS SODIUM OVERFLOW; OPENS AND REMAINS SO	1.0E-3/d [1]	10						
XTK001NF	SODIUM STORAGE TANK RUPTURES	1.0E-6/h [1]	10						
XVV001LP	FAILURE OF VALVE 1 TO OPEN	1.0E-2/d [1]	10						
XVV002LP	FAILURE OF VALVE 2 TO OPEN	1.0E-2/d [1]	10						
XVV00RLP	FAILURE OF REDUNDANT SYSTEM; VALVE FAILS TO OPEN	1.0E-2/d [1]	10						

Notes: 1. Recommended value tables, EGG-SSRE-8875, Feb. 1990
 2. Engineering Judgement for CCF

Table 7.24. Special Data (Cont'd)

PAGE 1

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
VHEDM1WF	WATER AND SODIUM DUMP AFTER LEAK ESCALATES	1.0E-1/d	5						
VHEDM2WF	WATER AND SODIUM DUMP AFTER LEAK ESCALATES	9.0E-1/d	1						
VHEFSOWF	PROBABILITY OF 2 SIMULTANEOUS DEG FAILURES	3.2E-3/d	10						
VHELKOWF	PROBABILITY OF LEAK DETECTION SYSTEM FAILURE	3.5E-4/d	10						
VHEMO1WF	LEAK ESCALATES WITHIN FIRST MINUTE	1.6E-1/d	3						
VHEMO2WF	LEAK ESCALATES WITHIN FIRST MINUTE	8.4E-1/d	1						
VHEPF2WF	PRESSURE RELIEF DOES NOT OCCUR	5.0E-1/d	10						
VHEPF3WF	PRESSURE RELIEF DOES NOT OCCUR	1.0E-1/d	10						
VHEPF4WF	PRESSURE RELIEF DOES NOT OCCUR	9.0E-1/d	1						
VHERIXWF	REACTOR SHUTDOWN AFTER IHX FAILURE	1.0E-2/d	10						
VHESLKWF	SMALL OR MEDIUM LEAK	3.2E-2/d	10						
VHESPRWW	SODIUM PRESSURE RELIEF SYSTEM FAILS	4.4E-4/d	10						

Table 7.24. Special Data (Cont'd)

PAGE 1

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
FPP001RF	PIN-TO-PIN BREACH PROPAGATION	1.0E-4/d	10						
FPP002RF	PIN-TO-PIN BREACH PROPAGATION	1.0E-2/d	10						
FPP003RF	PIN-TO-PIN BREACH PROPAGATION	1.0E-1/d	3						
FPP004RF	PIN-TO-PIN BREACH PROPAGATION	1.0E-3/d	10						
FPP111RF	PIN-TO-PIN BREACH PROPAGATION NOT SELF-LIMITING	1.0E-4/d	10						
FPP112RF	PIN-TO-PIN BREACH PROPAGATION NOT SELF-LIMITING	1.0E-3/d	10						
FPP901RF	SUB-TO-SUB DAMAGE PROPAGATION	1.0E-4/d	10						
FPP902RF	SUB-TO-SUB DAMAGE PROPAGATION	1.0E-3/d	10						
FPP903RF	SUB-TO-SUB DAMAGE PROPAGATION	1.0E-2/d	10						
FPP991RF	PIN BREACH IN FAULTED SUBASSEMBLY	9.0E-1/d	3						

Table 7.24. Special Data (Cont'd)

PAGE 1

EVENT NAME	DESCRIPTION	VALUE USED 3/91		CRBRP-4		IREP		GENERIC	
		Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor	Failure Rate	Error Factor
GND002RW	BREACH DETECTION FAILURE	1.0E-5/d	10						
GND003RW	BREACH DETECTION FAILURE	1.0E-4/d	10						

7.2.4 Reactivity Insertion Data

7.2.4.1 Introduction

One important class of events in the EBR-II PRA is the reactivity insertion events which arise from inadvertent motion of control or safety rods. The relative importance depends on two factors, 1) the frequency of a given event and 2) the amount of reactivity inserted. The purpose of this discussion is to present the assessment of reactivity available for insertion from accidental rod run-in in EBR-II. Estimates are generated for the magnitude of reactivity inserted, the insertion rate and the relative frequency for different classifications of rods.

These estimates were developed from administrative and physical limits, operating procedures and actual operating history. Technical Specifications [7.5] impose upper bounds on rod worths. Rod and core designs constitute additional limitations on rods worths and reactivity insertion rates. Operating Procedures [7.6] imply patterns for axial rod positioning and movement. Finally, data files from the Digital Acquisition System (DAS) provide an extensive history of rod positions and, implicitly, reactivity insertion potential. Probability density functions (pdfs) for reactivity insertion magnitudes, and estimates of insertion rates and relative frequencies were constructed from these sources.

7.2.4.2 Administrative and Physical Reactivity Limits

7.2.4.2.1 Reactivity Insertion Magnitude

The reactivity that a rod can insert in a TOP accident depends on the total worth of the rod and how far the rod was inserted before the event. Since rod worths vary roughly linearly with insertion over their 14 inch travel, the TOP reactivity is approximately

$$R_i = R_t*(1-Z/14), \quad (7.1)$$

where R_t is the total worth and Z is the pre-accident rod position, in inches, measured from full out. (Recall that the EBR-II rods are fueled, so reactivity increases as they are inserted.)

The administrative and physical limits on rod worths (R_t) and positions (Z) are discussed in the following paragraphs. These limits are then used with Eq. 7.1 to estimate reactivity insertion magnitudes.

1. Range of TOP Reactivity, R_t

There are three basic rod types, each with different characteristic values of R_t . The control rods are referred to as "normal" or "high-worth" types, depending on their design. In recent years, each loading has had only one normal worth rod and seven to nine high-worth rods. The third type is safety rods, and every loading has two of them. They are driven as a pair, so their worths are combined for this assessment.

The total worth of a rod is limited by the design of the rod and the surrounding core. It is unlikely that these parameters will change significantly and this analysis assumes that they will not. The grid locations of the rods, the fuel pin and boron follower components of the rods and even the basic core layout are all invariant. The rod worths vary due to such factors as the burnup of the fuel in the rod and the types of subassemblies in the vicinity of the rod.

The range of R_t values that occurred over the 29 subruns from Run 139A (May 1986) through Run 153A (March 1990) are shown in Table 7.23. The ratio of maximum to minimum total worth of a rod is about 1.5 for all three types. To confirm that the upper limits in Table 7.25 are accurate, the maximum rod worth for each rod type was checked for every fifth numbered run starting with Run 100A (1979). A slightly higher worth was found for the normal type, 44.95 cents, but not for the others. Thus, the upper bounds for rod worths are \$1.43 for safety rods, \$0.45 for normal-type control rods and \$0.99 for high worth-type control rods.

Control rod worths are also limited by Technical Specifications. No control rod is allowed to be worth more than \$1.30, but this limit is of no practical concern since that value

Table 7.25. Rod Worth^a By Type From Run 139A Through Run 153A (¢)

<u>Type</u>	<u>Safety Rods</u>	<u>Normal CR</u>	<u>High Worth CR</u>
Minimum Worth (cents)	96.2	31.2	64.0
Average Worth (cents)	116.2	38.2	79.0
Maximum Worth (cents)	143.1	44.0	99.0

^a For a single control rod or the pair of safety rods

cannot be approached with the current design. There is a limit of practical interest, \$0.90, when a control rod is operated under computer control (automatic control rod drive system, or ACRDS) at the fast drive speed. However, this mode of operation is only allowed as part of an approved plant transient experiment. Thus, these two Technical Specification limits are not really relevant to steady-power operation.

2. Range of Rod Position, Z

There is a Technical Specification that implies a minimum control rod bank position. It is that the control rods must be able to withdraw at least \$3.70 when scrammed during full power operation. Assuming that 1) the scram worth is proportional to the distance moved, 2) control rod worths are the averages in Table 7.22, and 7.3) the number of control rods is the maximum used in the last 10 years (one normal and nine high worth), this Specification implies a minimum control rod bank position (Z) of 6.9 inches.

The practical minimum control rod bank position is about an inch higher than this, 8.0 inches. It is the policy of the Chief Physicist to reject any proposed core loading that would require a control bank position less than 8 inches [7.7]. The reason for this policy is to keep the boron portion of high worth control rods out of the active core region. A survey of beginning-of-run bank positions (every fifth numbered run from Run 100 to Run 135 and every subrun starting from Run 139A) showed that the minimum bank position that has actually been used is 8.0 inches.

There are two specifications that restrict the safety rod position. One is an Operating Instruction prescribing an absolute minimum insertion of 7.5 inches. The other is a Technical Specification that the safety rods must be able to withdraw reactivity equal to at least 120% of the power reactivity decrement (PRD). This second limit is almost always less constraining than the 7.5 inch minimum; with an average safety rod worth (\$1.16) and the average PRD (32 cents) it implies a bank position of only 4.6 inches. In the most extreme case since 1978 (94 cent rod worth and 46 cent PRD in Run 103B), the PRD constraint was the limiting one, with a limit of 8.3 inches. In recent years, the safety rods have been positioned near the 7.5 inch minimum

on a number of occasions to allow more control rod insertion when there was a high fissile loading.

The Operating Instructions prescribe additional limits on the rod positions during steady power operation. Control rods are divided into two categories in these instructions. The Operations Physicist designates a particular control rod to be the "controlling" rod for the duration of a subrun. This rod is used to make the short-term reactivity adjustments necessary to compensate for burnup. The allowable range of positions for the controlling rod is 9 to 12 inches if it is a high worth rod and 5 to 9.5 inches if it is the normal worth rod. The remaining control rods are all banked at a common position.

Once every few days the rods must be rebanked. When the controlling rod approaches its upper position limit, either the control rod bank or the safety rod pair is moved up a short distance ($\sim 1/4$ inch) as the controlling rod is lowered to approximately its lower limit. Which set of rods is moved up depends on how much excess reactivity is present in the core at the time. Assuming a large initial excess, the current procedure calls for rebanking the control rods until they reach 10 inches, then rebanking the safety rods until they are fully inserted, and then rebanking the control rods again.

Putting these facts together yields the following position histories for each rod category. For the controlling rod, the position varies uniformly between 9 and 12 inches if it is a high worth rod or 5 and 9.5 inches if it is a normal worth rod. For the control rod bank, the position varies uniformly between a lower limit, determined by the initial excess, and an upper limit, determined by the final burnup. The lower limit is at least 8 inches and the upper limit can be as large as full insertion (14 inches). For the safety rods, the history in general is a bimodal curve with a low plateau between the peaks. The position of the first peak is determined by the initial excess and can be as little as 7.5 inches. Because the safety rods are worth less than $1/5$ as much as the control rod bank, the safety rods spend relatively little time between this lower position and the second peak, which is at the, fully inserted, 14 inch position. The safety rods stay at 14 inches for the remainder of the run.

Combining these position histories with the average rod worths from Table 7.23 implies an R_i distribution for each rod category. For the controlling rod, the R_i varies uniformly between 28 and 11 cents if it is a high worth rod or between 24 and 12 cents if it is a normal worth rod. For a rod in the control rod bank, R_i varies uniformly from as much as 34 cents to as little as zero. For the safety rods, there can be a peak where R_i is as much as 54 cents, and then a short time that is spread uniformly from there to $R_i = 0$, which is the value maintained for the remainder of the run.

The Operating Instructions prescribe one other kind of rod motion during steady power operation, a daily control rod motion test. In this test, each control rod must be moved at least three inches, either up or down, to verify that it is working properly. This implies that, for a small fraction of one hour per day, some control rods are down at least three inches from the bank position. This adds a low frequency, higher-than-normal worth contribution to the control rod insertion potential. This phenomenon is treated in Section 7.2.4.3.

7.2.4.2.2 Rate of Insertion

For a given value of R_i , the rate of reactivity insertion depends on the initial rod position Z and the drive speed. The drive speed is 2.11 inches/minute for safety rods and 4.76 inches/minute for control rods. (The ACRDS drives control rod #4 at 72 inches/minute in high speed mode but this is only done as part of a plant experiment, not during steady power operation.) Again assuming linear worth variation with rod position, it is easy to estimate reactivity insertion rates. For example, an average safety rod run in from the 7.5 inch position would introduce a total of ~ 54 cents at ~ 0.29 cents/s, an average high-worth control rod starting at 8 inches would introduce a total of ~ 34 cents at ~ 0.45 cents/s, and an average normal worth rod run in from 8 inches would introduce a total of ~ 16 cents at ~ 0.22 cents/s.

The linear worth variation approximation underestimates the insertion rate over part of the rod travel. The actual worth profiles are shown in Fig. 7.1. For high-worth control rods, the maximum insertion rate is 14% larger than the average and it is essentially constant at this

Fig. 7.1. Rod Worth Profiles

maximum value between 5 and 10 inches. For safety rods and normal-worth control rods, the maximum insertion rate is 32% larger than the average and it is essentially constant at this maximum value between 4 and 9 inches.

It should be noted that insertion rate bounds in the Technical Specifications are quite conservative. Actual bounds can be found using maximum rod worths from Table 7.23, physical drive speeds, and nonlinearity factors from the previous paragraph. The maximum insertion rate for the highest worth control rod is 0.64 cents/s and for the highest worth safety rods it is 0.47 cents/s. These are much smaller than the Technical Specification limit of 1.0 cents/s. Similarly, the maximum insertion rate for the highest-allowed-worth control rod (90 cents), driven by the ACRDS at high speed, is 8.8 cents/s, compared to the Technical Specification limit of 12 cents/s.

7.2.4.2.3 Relative Initiator Frequencies

The occurrence frequencies for rod run-in events depends, of course, on the postulated failure modes. No attempt is made here to make a complete evaluation of failure modes, but to point out the relative frequency for the various rod categories for what appears to be the most likely failure mode. Run-in of a single rod is considered first, followed by simultaneous run-in of multiple rods.

The most likely scenario for a single rod run-in is failure of a rod to stop moving in when the reactor operator releases the up motion control switch. The spring-loaded control rod motion switch could fail closed in the up position, for example.

For this mode, the failure frequency is proportional to the frequency of up-motion demands. The demand frequency depends on the rod category. The controlling control rod is the one moved most frequently. Both discussions with Operations [7.8] and observation of controlling rod motions recorded by the DAS indicate that this rod is moved up three times per hour on average. The other control rods are moved up at most once every few days to follow burnup, but are moved up once per day during the daily motion check (0.04 times per hour).

The safety rods are not involved in daily motion tests and are moved only at one stage of burnup, as described in Section II.A.2. From May 1986 through February 1990, the safety rods were moved up 86 times in the 22,486 hours of full-power operation. This implies an average up motion frequency of 0.004 times per hour.

Simultaneous run-in of more than one rod is much less likely than a single-rod run-in. Interlocks prevent more than one rod from being driven in at a time. A failure mode such as a short must occur before more than one rod could move in at a time. The number of rods that could move up at one time and the frequency at which this could occur has been determined by a failure analysis. Only the upward motion of one rod and two rods is considered in Section 7.2.4.3.

7.2.4.3 Analysis of DAS Data

7.2.4.3.1 Processing the Data

The DAS records of rod positions were used to construct more precise descriptions of rod run-in reactivity. The results in Section 7.2.2.2 provided guidance for development of a procedure to process the DAS data.

Data going back to May 1986 were analyzed, since that is the time at which the current format of DAS records was implemented. The most recent data included are from February 28, 1990. These data cover 15 numbered runs, 29 subruns and 22486 hours of full-power operation. This history is extensive enough to represent the range of conditions expected to occur in EBR-II and to provide good statistical precision for the probability density functions derived from it.

A DAS record contains time averages of the power level and position of each control and safety rod. One minute, 10 minute, and one hour time averages are available. One hour records were used here to make it feasible to process many years worth of data.

The disadvantage of using one-hour averages is that short-duration departures from an

average position are not apparent. However, movements at steady power occur much less frequently than once per hour with two exceptions. One is adjustment of the controlling rod. This occurs about three times per hour but the total movement over an hour is only about 0.03 inches, which is much less than the resolution of interest.

The other exception is the daily control rod motion check. This is an important departure from the average that must be accounted for. It was treated by modelling a typical daily motion test in a separate subroutine of the processing code written for this analysis. A typical rod motion pattern was established by interviewing Operations management and recording the actual motion test on March 20, 1990. Typically, each rod is moved 3.5 inches from its normal position and is immediately returned to the normal position. To keep the power level steady, pairs of rods are moved simultaneously in opposite directions. Thus only half the rods go lower than their normal position in a test. Only the motion of these rods was modeled, a slight conservatism that saved considerable effort. The motion-test subroutine was called to process the records corresponding to the first hour of each day, since that is then motion tests actually are done.

Only records corresponding to steady power operation were accepted when processing the data. Records with a power level outside the range 52 to 65 MWt were rejected, which eliminated all but 11 records of unacceptable data. This residual was identified by having the program flag records with extraordinary rod position values, and then hand-checking documentation to determine whether a plant transient experiment was run at that time.

A rod position, Z , was converted to a reactivity insertion potential, R_i , using the measured total worth of that rod and the appropriate axial worth distribution (profile). Accordingly, the measured total worth of each rod at each subrun was supplied to the processing code. Actual worth profiles depart modestly from the linear profile assumed in most of Section II. Four different rod worth profiles, in the form of quartic polynomials, were available. By testing these profiles against actual rod calibration data from Runs 144 and 149, one profile was found to be adequate for the safety and normal rods, and another profile was shown to be adequate for high worth rods. These are the two profiles shown in Fig. 7.1.

Discrete probability density functions were constructed from the R_i values. The range of possible R_i values was divided into five cent intervals or bins. For each R_i value encountered, the bin encompassing that value was identified and then a weight was added to that bin. The weight was one (hour) except when the hour represented by a record was subdivided to simulate the daily motion test. This process produced a frequency distribution. To convert this to a probability density function, the weight in each bin was divided by the total of all bin weights.

A separate pdf for single-rod run-in was constructed for each rod category. Three categories were defined, each with a characteristic rod drive speed and relative frequency for initiating a TOP accident, as identified in Section 7.2.4.2. The categories are safety rods, the controlling rod and the banked control rods.

Simultaneous run-in of two control rods also was analyzed. The R_i from each pair of control rods was formed, producing a two-control-rod pdf.

7.2.4.3.2 Results

The pdfs are shown in Table 7.26 and Figures 7.2 and 7.3. The pdfs show the conditional probability that the magnitude of the TOP initiator will be within a 5 cent interval given that there is an accidental run-in of a rod of a particular type. The results are also shown as cumulative distribution functions (cdfs) in Table 7.25.

The distributions are consistent with the estimates made in Section 7.2.4.2. For example, the safety rod pdf shows there is about a 20% probability that R_i will be around 50 cents, a 60% probability that R_i will be nearly zero cents, and a small, nearly uniform probability that R_i will have some intermediate value. The control rod pdfs have a low probability tail at the high worth end, which is the result of the daily motion tests. The controlling rod pdf is narrower than the pdf for the banked control rods, reflecting the more restricted range of Z positions prescribed by the Operating Instructions. The two-rod run-in pdf is the broadest and it bears some resemblance to a normal distribution function. The cdfs show that the 99.99th percentile is below 55 cents for safety rods, 40 cents for any control rod and 75 cents for two control rods.

Safety rods are treated as having two or three states, because of their uneven pattern of activity. About 58% of the time they are fully inserted and inactive. Therefore, the time when safety rods could run-in is 42% of the 6900 hours of steady power operation per year (~2900 hr/yr). When the number of hours exposed to an insertion potential is the dominant consideration for the failure frequency, then the safety rod columns in Tables 7.26 and 7.27 are used after adjustment. The adjustment consists of subtracting 0.58 from the first bin and renormalizing. When, however, the number of up-motion demands are the key, then it must be recognized that the safety rods are inactive for almost all the time they are at their lowest position. A flat pdf spanning the range from 0 to 55 cents is then the most appropriate one to use, since the safety rods are moved up uniformly from their startup position to 14 inches. (The variations in the middle portion of the safety rod pdf in Table 7.26 are largely just statistical fluctuations.)

The rate of reactivity insertion, as well as the total magnitude, is needed to do transient calculations. The rate can be determined by combining the reactivity magnitude, a corresponding Z value, a rod worth profile and a drive speed. The average Z value corresponding to each reactivity bin is given in Table 7.28. The drive speed, S, is 0.0352 inches/s for the safety rods and 0.0793 inches/s for control rods. The worth profile is expressed as a quartic polynomial,

$$P(z) = A0 + A1*Z + A2*Z**2 + A3*Z**3 + A4*Z**4, \quad (7.2)$$

and the coefficients for safety rods and control rods are given in Table 7.29.

Table 7.26. Discrete Probability Density Functions For Rod Run-in Reactivity

Rho (cents)		Single Rod Run-in			
		Safety	Cntl CR	Banked CRs	Two CRs
MIN	MAX				
0	5	0.606822023	0.000132345	0.061044466	0.010040046
5	10	0.040736458	0.075747506	0.150779352	0.020526435
10	15	0.007782620	0.448401672	0.183909662	0.038487127
15	20	0.032864894	0.381694672	0.231532430	0.077445187
20	25	0.010984613	0.080360714	0.231616327	0.107048768
25	30	0.014364493	0.011718298	0.101984695	0.145214671
30	35	0.040780930	0.001677569	0.033815610	0.132012425
35	40	0.027439296	0.000255383	0.005253354	0.154911625
40	45	0.016098906	0.000008715	0.000038014	0.143157532
45	50	0.180201014	0.000002369	0.000017457	0.073531546
50	55	0.021924753	0.000000747	0.000006374	0.051334064
55	60	0.000000000	0.000000010	0.000002124	0.031193269
60	65	0.000000000	0.000000000	0.000000133	0.008349918
65	70	0.000000000	0.000000000	0.000000000	0.003968931
70	75	0.000000000	0.000000000	0.000000000	0.002769935
75	80	0.000000000	0.000000000	0.000000000	0.000005152
80	85	0.000000000	0.000000000	0.000000000	0.000002127
85	90	0.000000000	0.000000000	0.000000000	0.000000880
90	95	0.000000000	0.000000000	0.000000000	0.000000320
95	100	0.000000000	0.000000000	0.000000000	0.000000042

Fig. 7.2. Probability Density Functions for Single-Rod Run-In Reactivity

Fig. 7.3. Probability Density Function for Double-Rod Run-In Reactivity

Table 7.27. Discrete Cumulative Distribution Functions For Rod Run-in Reactivity

Rho (cents)		Single Rod Run-in			
		<u>Safety</u>	<u>Cntl CR</u>	<u>Banked CRs</u>	<u>Two CRs</u>
<u>MIN</u>	<u>MAX</u>				
0	5	0.606822023	0.000132345	0.061044466	0.010040046
5	10	0.647558481	0.075879851	0.211823818	0.030566481
10	15	0.655341101	0.524281523	0.395733480	0.069053608
15	20	0.688205995	0.905976195	0.627265910	0.146498795
20	25	0.699190607	0.986336909	0.858882237	0.253547563
25	30	0.713555101	0.998055208	0.960866932	0.398762234
30	35	0.754336031	0.999732777	0.994682542	0.530774659
35	40	0.781775327	0.999988160	0.999935897	0.685686285
40	45	0.797874233	0.999996875	0.999973911	0.828843816
45	50	0.978075247	0.999999244	0.999991369	0.902375362
50	55	1.000000000	0.999999990	0.999997743	0.953709426
55	60	1.000000000	1.000000000	0.999999867	0.984902695
60	65	1.000000000	1.000000000	1.000000000	0.993252613
65	70	1.000000000	1.000000000	1.000000000	0.997221544
70	75	1.000000000	1.000000000	1.000000000	0.999991479
75	80	1.000000000	1.000000000	1.000000000	0.999996631
80	85	1.000000000	1.000000000	1.000000000	0.999998758
85	90	1.000000000	1.000000000	1.000000000	0.999999638
90	95	1.000000000	1.000000000	1.000000000	0.999999958
95	100	1.000000000	1.000000000	1.000000000	1.000000000

Table 7.28. Average Rod Position For Each Reactivity Bin and Rod Category

Rho (cents)		Single Rod Run-in			
		<u>Safety</u>	<u>Cntl CR</u>	<u>Banked CRs</u>	<u>Two CRs</u>
<u>MIN</u>	<u>MAX</u>				
0	5	14.006	13.023	12.450	13.555
5	10	11.996	11.658	11.402	12.860
10	15	11.227	10.914	11.058	12.241
15	20	10.690	9.896	10.338	11.678
20	25	9.857	9.349	9.721	11.200
25	30	9.597	8.954	9.091	10.688
30	35	9.044	8.857	8.743	10.382
35	40	7.868	7.837	7.706	10.007
40	45	7.832	6.950	6.794	9.817
45	50	7.541	6.575	6.333	9.465
50	55	7.517	5.991	5.916	9.077
55	60		5.561	5.507	8.879
60	65			5.180	8.607
65	70				7.997
70	75				7.566
75	80				7.393
80	85				7.075
85	90				6.664
90	95				6.291
95	100				6.136

Table 7.29. Polynomial Coefficients for the Worth Profiles, Eq. 2.

<u>Coefficient</u>	<u>Value</u>	
	<u>Safety or Normal</u>	<u>High Worth</u>
A0	1.001092	0.9997367
A1	-3.472015E-2	-5.249227E-2
A2	-9.896010E-3	-4.656591E-3
A3	5.692658E-4	2.734573E-4
A4	-3.577061E-6	-2.653448E-6

7.2.4.3.3 Scram Reactivity

While processing the DAS data for rod run-in, it was also possible to resolve the question of how many control rods must successfully scram to achieve neutronic shutdown. A pdf for reactivity removal by one or more high worth control rods was constructed to answer this question.

The pdfs are shown in Table 7.30 and Fig. 7.4. Corresponding cumulative distribution functions are displayed in Table 7.29. From Fig. 7.4 one notes that even one rod scrambling will usually be sufficient, but examination of the tables shows that each distribution has a long, low probability tail at the low worth end. This tail arises from the few minutes each day that some control rods are well below the rod bank position during the daily rod motion test. (There should be a corresponding high worth tail from rods above the bank position, but this was not modeled because it is of no consequence here.) Although this produces a significant probability, 3×10^{-5} , that one rod will not be enough, two rods are clearly sufficient to overcome the PRD.

The most likely source of external reactivity, which would also have to be overcome by scrambling rods, is accidental control rod run-in. Since the rod moving in is obviously not jammed, it is very likely to be one of the rods that scrams. Then its run-in reactivity will be withdrawn, making scram of two rods still sufficient. If for some reason it does not scram, as many as three rods might be necessary for shutdown, considering the probable run-in reactivity magnitudes.

In the case of scrambling more than four rods, a variation on the analysis was done. Instead of tallying all possible combinations of N rods, N times the average scram worth was used. This ignores the fluctuations at any given hour, but it has the property that the pdf for N rods scrambling can be found by scaling the abscissa of the single-rod pdf. (There is a modest discretization error from dividing the worths into five cent intervals.) If the pdfs based

Table 7.30. Probability Density Functions for Reactivity From Scram
of High-Worth Rods (Exact)

Reactivity (ρ)		1 Rod	2 Rods	3 Rods	4 Rods
min	max				
0	5	0.000000000	0.000000000	0.000000000	0.000000000
5	10	0.000000000	0.000000000	0.000000000	0.000000000
10	15	0.000000000	0.000000000	0.000000000	0.000000000
15	20	0.000000041	0.000000000	0.000000000	0.000000000
20	25	0.000000691	0.000000000	0.000000000	0.000000000
25	30	0.000004778	0.000000000	0.000000000	0.000000000
30	35	0.000020446	0.000000000	0.000000000	0.000000000
35	40	0.001042961	0.000000000	0.000000000	0.000000000
40	45	0.007465519	0.000000000	0.000000000	0.000000000
45	50	0.059038983	0.000000000	0.000000000	0.000000000
50	55	0.164624479	0.000000000	0.000000000	0.000000000
55	60	0.215690198	0.000000029	0.000000000	0.000000000
60	65	0.212298742	0.000000270	0.000000000	0.000000000
65	70	0.167526359	0.000000651	0.000000000	0.000000000
70	75	0.095187643	0.000001153	0.000000000	0.000000000
75	80	0.049038117	0.000580310	0.000000000	0.000000000
80	85	0.020095220	0.002321351	0.000000000	0.000000000
85	90	0.007965821	0.003709001	0.000000000	0.000000000
90	95	0.000000000	0.007157751	0.000000000	0.000000000
95	100	0.000000000	0.023017615	0.000000008	0.000000000
100	105	0.000000000	0.046587482	0.000000122	0.000000000
105	110	0.000000000	0.084354309	0.000000392	0.000000000
110	115	0.000000000	0.121839728	0.000000556	0.000000000
115	120	0.000000000	0.143700601	0.000146622	0.000000000
120	125	0.000000000	0.139866926	0.001471564	0.000000000
125	130	0.000000000	0.123281913	0.002364413	0.000000000
130	135	0.000000000	0.094745509	0.001359784	0.000000000
135	140	0.000000000	0.074649307	0.001383141	0.000000000
140	145	0.000000000	0.054035134	0.004439226	0.000000040
145	150	0.000000000	0.036830947	0.013364774	0.000000211
150	155	0.000000000	0.020455282	0.023957448	0.000000391
155	160	0.000000000	0.011812309	0.037666305	0.000000459
160	165	0.000000000	0.006561492	0.059927742	0.000593059
165	170	0.000000000	0.003248248	0.077954115	0.001907468
170	175	0.000000000	0.000894760	0.101293041	0.001453449
175	180	0.000000000	0.000347921	0.107759736	0.000455755
180	185	0.000000000	0.000000000	0.103900663	0.000858144
185	190	0.000000000	0.000000000	0.089316851	0.002123516
190	195	0.000000000	0.000000000	0.080055579	0.003770195
195	200	0.000000000	0.000000000	0.067839918	0.007496005
200	205	0.000000000	0.000000000	0.058436646	0.014297219
205	210	0.000000000	0.000000000	0.048479407	0.022988700
210	215	0.000000000	0.000000000	0.039029338	0.032051795
215	220	0.000000000	0.000000000	0.028527142	0.044503694
220	225	0.000000000	0.000000000	0.020548488	0.054582154
225	230	0.000000000	0.000000000	0.012138655	0.072882359
230	235	0.000000000	0.000000000	0.008334194	0.083074090
235	240	0.000000000	0.000000000	0.004546147	0.089352818
240	245	0.000000000	0.000000000	0.003133398	0.080794937
245	250	0.000000000	0.000000000	0.001281194	0.074391391
250	255	0.000000000	0.000000000	0.001064301	0.067148227
255	260	0.000000000	0.000000000	0.000188618	0.060964815

Table 7.30. Probability Density Functions for Reactivity From Scram
of High-Worth Rods (Exact) (Cont'd)

Reactivity (β)		1 Rod	2 Rods	3 Rods	4 Rods
min	max				
260	265	0.000000000	0.000000000	0.000090470	0.052880528
265	270	0.000000000	0.000000000	0.000000000	0.045804723
270	275	0.000000000	0.000000000	0.000000000	0.040354003
275	280	0.000000000	0.000000000	0.000000000	0.035891786
280	285	0.000000000	0.000000000	0.000000000	0.029981097
285	290	0.000000000	0.000000000	0.000000000	0.024004467
290	295	0.000000000	0.000000000	0.000000000	0.018484081
295	300	0.000000000	0.000000000	0.000000000	0.013285326
300	305	0.000000000	0.000000000	0.000000000	0.008144082
305	310	0.000000000	0.000000000	0.000000000	0.005785951
310	315	0.000000000	0.000000000	0.000000000	0.003874059
315	320	0.000000000	0.000000000	0.000000000	0.002235698
320	325	0.000000000	0.000000000	0.000000000	0.001476479
325	330	0.000000000	0.000000000	0.000000000	0.001144326
330	335	0.000000000	0.000000000	0.000000000	0.000588441
335	340	0.000000000	0.000000000	0.000000000	0.000228851
340	345	0.000000000	0.000000000	0.000000000	0.000145212
345	350	0.000000000	0.000000000	0.000000000	0.000000000

Fig. 7.4. Exact Probability Density Functions for Scram Worth

Table 7.31. Cumulative Distribution Functions for Reactivity From Scram
of High-Worth Rods (Exact)

Reactivity (ρ)		1 Rod	2 Rods	3 Rods	4 Rods
min	max				
0	5	0.000000000	0.000000000	0.000000000	0.000000000
5	10	0.000000000	0.000000000	0.000000000	0.000000000
10	15	0.000000000	0.000000000	0.000000000	0.000000000
15	20	0.000000041	0.000000000	0.000000000	0.000000000
20	25	0.000000732	0.000000000	0.000000000	0.000000000
25	30	0.000005510	0.000000000	0.000000000	0.000000000
30	35	0.000025957	0.000000000	0.000000000	0.000000000
35	40	0.001068917	0.000000000	0.000000000	0.000000000
40	45	0.008534436	0.000000000	0.000000000	0.000000000
45	50	0.067573420	0.000000000	0.000000000	0.000000000
50	55	0.232197899	0.000000000	0.000000000	0.000000000
55	60	0.447888097	0.000000029	0.000000000	0.000000000
60	65	0.660186840	0.000000299	0.000000000	0.000000000
65	70	0.827713199	0.000000950	0.000000000	0.000000000
70	75	0.922900842	0.000002103	0.000000000	0.000000000
75	80	0.971938959	0.000582413	0.000000000	0.000000000
80	85	0.992034179	0.002903764	0.000000000	0.000000000
85	90	1.000000000	0.006612765	0.000000000	0.000000000
90	95	1.000000000	0.013770516	0.000000000	0.000000000
95	100	1.000000000	0.036788131	0.000000008	0.000000000
100	105	1.000000000	0.083375613	0.000000130	0.000000000
105	110	1.000000000	0.167729922	0.000000522	0.000000000
110	115	1.000000000	0.289569650	0.000001078	0.000000000
115	120	1.000000000	0.433270252	0.000147700	0.000000000
120	125	1.000000000	0.573137178	0.001619264	0.000000000
125	130	1.000000000	0.696419091	0.003983677	0.000000000
130	135	1.000000000	0.791164600	0.005343461	0.000000000
135	140	1.000000000	0.865813906	0.006726602	0.000000000
140	145	1.000000000	0.919849040	0.011165828	0.000000040
145	150	1.000000000	0.956679988	0.024530602	0.000000251
150	155	1.000000000	0.977135270	0.048488050	0.000000642
155	160	1.000000000	0.988947579	0.086154355	0.000001101
160	165	1.000000000	0.995509071	0.146082098	0.000594161
165	170	1.000000000	0.998757319	0.224036213	0.002501629
170	175	1.000000000	0.999652079	0.325329254	0.003955078
175	180	1.000000000	1.000000000	0.433088989	0.004410833
180	185	1.000000000	1.000000000	0.536989653	0.005268977
185	190	1.000000000	1.000000000	0.626306504	0.007392493
190	195	1.000000000	1.000000000	0.706362082	0.011162688
195	200	1.000000000	1.000000000	0.774202001	0.018658693
200	205	1.000000000	1.000000000	0.832638647	0.032955912
205	210	1.000000000	1.000000000	0.881118054	0.055944611
210	215	1.000000000	1.000000000	0.920147392	0.087996406
215	220	1.000000000	1.000000000	0.948674534	0.132500100
220	225	1.000000000	1.000000000	0.969223022	0.187082255
225	230	1.000000000	1.000000000	0.981361677	0.259964613
230	235	1.000000000	1.000000000	0.989695871	0.343038703
235	240	1.000000000	1.000000000	0.994242018	0.432391521
240	245	1.000000000	1.000000000	0.997375417	0.513186458
245	250	1.000000000	1.000000000	0.998656610	0.587577849
250	255	1.000000000	1.000000000	0.999720911	0.654726075
255	260	1.000000000	1.000000000	0.999909530	0.715690890

Table 7.31. Cumulative Distribution Functions for Reactivity From Scram
of High-Worth Rods (Exact) (Cont'd)

<u>Reactivityv (¢)</u>		<u>1 Rod</u>	<u>2 Rods</u>	<u>3 Rods</u>	<u>4 Rods</u>
<u>min</u>	<u>max</u>				
260	265	1.000000000	1.000000000	1.000000000	0.768571418
265	270	1.000000000	1.000000000	1.000000000	0.814376142
270	275	1.000000000	1.000000000	1.000000000	0.854730144
275	280	1.000000000	1.000000000	1.000000000	0.890621930
280	285	1.000000000	1.000000000	1.000000000	0.920603027
285	290	1.000000000	1.000000000	1.000000000	0.944607494
290	295	1.000000000	1.000000000	1.000000000	0.963091575
295	300	1.000000000	1.000000000	1.000000000	0.976376901
300	305	1.000000000	1.000000000	1.000000000	0.984520983
305	310	1.000000000	1.000000000	1.000000000	0.990306934
310	315	1.000000000	1.000000000	1.000000000	0.994180993
315	320	1.000000000	1.000000000	1.000000000	0.996416691
320	325	1.000000000	1.000000000	1.000000000	0.997893169
325	330	1.000000000	1.000000000	1.000000000	0.999037496
330	335	1.000000000	1.000000000	1.000000000	0.999625937
335	340	1.000000000	1.000000000	1.000000000	0.999854788
340	345	1.000000000	1.000000000	1.000000000	1.000000000
345	350	1.000000000	1.000000000	1.000000000	1.000000000

on the average worth closely match the exact ones, then it should be reasonable to get the pdfs for more than four rods by scaling.

The exact and approximate pdfs are displayed graphically in Fig. 7.5 and the approximate pdf and cdf values are given in Tables 7.32 and 7.33. The agreement improves as N increases and is quite good by $N = 4$. The only significant deficiency there is the missing low worth tail, which was removed by the averaging. At most one rod is below the bank at any moment, so the missing tail is about the same for any N . Therefore, a reasonably accurate procedure is to strip off the tail from the exact pdf, scale the remainder and then add back the tail.

The worth distributions for scrambling between one and four high-worth control rods have been determined by analyzing control rod data from 22486 hours of operation at steady power (four years). With a probability precision of 10^{-9} , the minimum scram worth from one, two, three and four rods is \$0.15, \$0.55, \$0.95 and \$1.40, respectively. With a precision of 10^{-7} , the minimum scram worths are five cents higher.

Shutdown is achieved when the reactivity removed is greater than the sum of the PRD and any external reactivity. A survey of PRD measurements over many years showed that the average PRD is about 32 cents and the most extreme values are 26 and 46 cents [7.9]. Table 7.30 shows that there is only about a 1% probability that a single control rod would not overcome even the largest PRD ever observed. Combining the scram pdf and the PRD magnitude pdf would give a more general probability that a single rod scram would overcome the PRD. Successful scram of two rods will certainly overcome the PRD.

Fig. 7.5. Exact and Approximate Probability Density Functions for Scram Worth

Table 7.32. Probability Density Functions for Reactivity From Scram
of High-Worth Rods (Approximate)

Reactivity (β)		1 Rod	2 Rods	3 Rods	4 Rods
min	max				
0	5	0.000000000	0.000000000	0.000000000	0.000000000
5	10	0.000000000	0.000000000	0.000000000	0.000000000
10	15	0.000000000	0.000000000	0.000000000	0.000000000
15	20	0.000000000	0.000000000	0.000000000	0.000000000
20	25	0.000000000	0.000000000	0.000000000	0.000000000
25	30	0.000000000	0.000000000	0.000000000	0.000000000
30	35	0.000000000	0.000000000	0.000000000	0.000000000
35	40	0.000000000	0.000000000	0.000000000	0.000000000
40	45	0.002979632	0.000000000	0.000000000	0.000000000
45	50	0.006448457	0.000000000	0.000000000	0.000000000
50	55	0.091167838	0.000000000	0.000000000	0.000000000
55	60	0.319042960	0.000000000	0.000000000	0.000000000
60	65	0.284888375	0.000000000	0.000000000	0.000000000
65	70	0.183892200	0.000000000	0.000000000	0.000000000
70	75	0.096059771	0.000000000	0.000000000	0.000000000
75	80	0.015520768	0.000000000	0.000000000	0.000000000
80	85	0.000000000	0.000000000	0.000000000	0.000000000
85	90	0.000000000	0.002979632	0.000000000	0.000000000
90	95	0.000000000	0.004714044	0.000000000	0.000000000
95	100	0.000000000	0.001734413	0.000000000	0.000000000
100	105	0.000000000	0.030196567	0.000000000	0.000000000
105	110	0.000000000	0.060971271	0.000000000	0.000000000
110	115	0.000000000	0.133994485	0.000000000	0.000000000
115	120	0.000000000	0.185048475	0.000000000	0.000000000
120	125	0.000000000	0.179934181	0.000000000	0.000000000
125	130	0.000000000	0.104954194	0.000000000	0.000000000
130	135	0.000000000	0.106065997	0.002979632	0.000000000
135	140	0.000000000	0.077826203	0.004669572	0.000000000
140	145	0.000000000	0.058925554	0.000044472	0.000000000
145	150	0.000000000	0.037134217	0.001734413	0.000000000
150	155	0.000000000	0.010717780	0.013830828	0.000000000
155	160	0.000000000	0.004802989	0.032464645	0.000000000
160	165	0.000000000	0.000000000	0.044872365	0.000000000
165	170	0.000000000	0.000000000	0.102152450	0.000000000
170	175	0.000000000	0.000000000	0.093346971	0.000133416
175	180	0.000000000	0.000000000	0.123543538	0.002846215
180	185	0.000000000	0.000000000	0.136974117	0.003646714
185	190	0.000000000	0.000000000	0.080716891	0.001067331
190	195	0.000000000	0.000000000	0.067197367	0.000000000
195	200	0.000000000	0.000000000	0.074490794	0.001734413
200	205	0.000000000	0.000000000	0.064173263	0.008538646
205	210	0.000000000	0.000000000	0.045228142	0.021657920
210	215	0.000000000	0.000000000	0.042381927	0.026772214
215	220	0.000000000	0.000000000	0.031441786	0.034199057
220	225	0.000000000	0.000000000	0.022236058	0.081873166
225	230	0.000000000	0.000000000	0.007960509	0.052121320
230	235	0.000000000	0.000000000	0.005470070	0.095659521
235	240	0.000000000	0.000000000	0.002090189	0.089388953
240	245	0.000000000	0.000000000	0.000000000	0.101618785
245	250	0.000000000	0.000000000	0.000000000	0.078315396
250	255	0.000000000	0.000000000	0.000000000	0.056879836
255	260	0.000000000	0.000000000	0.000000000	0.048074357

Table 7.32. Probability Density Functions for Reactivity From Scram
of High-Worth Rods (Approximate) (Cont'd)

<u>Reactivity (β)</u>		<u>1 Rod</u>	<u>2 Rods</u>	<u>3 Rods</u>	<u>4 Rods</u>
<u>min</u>	<u>max</u>				
260	265	0.000000000	0.000000000	0.000000000	0.054745175
265	270	0.000000000	0.000000000	0.000000000	0.051320822
270	275	0.000000000	0.000000000	0.000000000	0.041848261
275	280	0.000000000	0.000000000	0.000000000	0.035977942
280	285	0.000000000	0.000000000	0.000000000	0.029796318
285	290	0.000000000	0.000000000	0.000000000	0.029129236
290	295	0.000000000	0.000000000	0.000000000	0.021480032
295	300	0.000000000	0.000000000	0.000000000	0.015654185
300	305	0.000000000	0.000000000	0.000000000	0.006537401
305	310	0.000000000	0.000000000	0.000000000	0.004180379
310	315	0.000000000	0.000000000	0.000000000	0.004313795
315	320	0.000000000	0.000000000	0.000000000	0.000489193
320	325	0.000000000	0.000000000	0.000000000	0.000000000
325	330	0.000000000	0.000000000	0.000000000	0.000000000
330	335	0.000000000	0.000000000	0.000000000	0.000000000
335	340	0.000000000	0.000000000	0.000000000	0.000000000
340	345	0.000000000	0.000000000	0.000000000	0.000000000
345	350	0.000000000	0.000000000	0.000000000	0.000000000

Table 7.33. Cumulative Distribution Functions for Reactivity From Scram
of High-Worth Rods (Approximate)

Reactivity (β)		1 Rod	2 Rods	3 Rods	4 Rods
min	max				
0	5	0.000000000	0.000000000	0.000000000	0.000000000
5	10	0.000000000	0.000000000	0.000000000	0.000000000
10	15	0.000000000	0.000000000	0.000000000	0.000000000
15	20	0.000000000	0.000000000	0.000000000	0.000000000
20	25	0.000000000	0.000000000	0.000000000	0.000000000
25	30	0.000000000	0.000000000	0.000000000	0.000000000
30	35	0.000000000	0.000000000	0.000000000	0.000000000
35	40	0.000000000	0.000000000	0.000000000	0.000000000
40	45	0.002979632	0.000000000	0.000000000	0.000000000
45	50	0.009428089	0.000000000	0.000000000	0.000000000
50	55	0.100595926	0.000000000	0.000000000	0.000000000
55	60	0.419638886	0.000000000	0.000000000	0.000000000
60	65	0.704527261	0.000000000	0.000000000	0.000000000
65	70	0.888419461	0.000000000	0.000000000	0.000000000
70	75	0.984479232	0.000000000	0.000000000	0.000000000
75	80	1.000000000	0.000000000	0.000000000	0.000000000
80	85	1.000000000	0.000000000	0.000000000	0.000000000
85	90	1.000000000	0.002979632	0.000000000	0.000000000
90	95	1.000000000	0.007693676	0.000000000	0.000000000
95	100	1.000000000	0.009428089	0.000000000	0.000000000
100	105	1.000000000	0.039624655	0.000000000	0.000000000
105	110	1.000000000	0.100595926	0.000000000	0.000000000
110	115	1.000000000	0.234590412	0.000000000	0.000000000
115	120	1.000000000	0.419638886	0.000000000	0.000000000
120	125	1.000000000	0.599573068	0.000000000	0.000000000
125	130	1.000000000	0.704527261	0.000000000	0.000000000
130	135	1.000000000	0.810593258	0.002979632	0.000000000
135	140	1.000000000	0.888419461	0.007649204	0.000000000
140	145	1.000000000	0.947345015	0.007693676	0.000000000
145	150	1.000000000	0.984479232	0.009428089	0.000000000
150	155	1.000000000	0.995197011	0.023258917	0.000000000
155	160	1.000000000	1.000000000	0.055723561	0.000000000
160	165	1.000000000	1.000000000	0.100595926	0.000000000
165	170	1.000000000	1.000000000	0.202748377	0.000000000
170	175	1.000000000	1.000000000	0.296095348	0.000133416
175	180	1.000000000	1.000000000	0.419638886	0.002979632
180	185	1.000000000	1.000000000	0.556613004	0.006626345
185	190	1.000000000	1.000000000	0.637329894	0.007693676
190	195	1.000000000	1.000000000	0.704527261	0.007693676
195	200	1.000000000	1.000000000	0.779018056	0.009428089
200	205	1.000000000	1.000000000	0.843191319	0.017966735
205	210	1.000000000	1.000000000	0.888419461	0.039624655
210	215	1.000000000	1.000000000	0.930801388	0.066396869
215	220	1.000000000	1.000000000	0.962243174	0.100595926
220	225	1.000000000	1.000000000	0.984479232	0.182469092
225	230	1.000000000	1.000000000	0.992439740	0.234590412
230	235	1.000000000	1.000000000	0.997909811	0.330249933
235	240	1.000000000	1.000000000	1.000000000	0.419638886
240	245	1.000000000	1.000000000	1.000000000	0.521257671
245	250	1.000000000	1.000000000	1.000000000	0.599573068
250	255	1.000000000	1.000000000	1.000000000	0.656452904
255	260	1.000000000	1.000000000	1.000000000	0.704527261

Table 7.33. Cumulative Distribution Functions for Reactivity From Scram
of High-Worth Rods (Approximate) (Cont'd)

Reactivity (β)		1 Rod	2 Rods	3 Rods	4 Rods
min	max				
260	265	1.000000000	1.000000000	1.000000000	0.759272436
265	270	1.000000000	1.000000000	1.000000000	0.810593258
270	275	1.000000000	1.000000000	1.000000000	0.852441519
275	280	1.000000000	1.000000000	1.000000000	0.888419461
280	285	1.000000000	1.000000000	1.000000000	0.918215779
285	290	1.000000000	1.000000000	1.000000000	0.947345015
290	295	1.000000000	1.000000000	1.000000000	0.968825047
295	300	1.000000000	1.000000000	1.000000000	0.984479232
300	305	1.000000000	1.000000000	1.000000000	0.991016633
305	310	1.000000000	1.000000000	1.000000000	0.995197011
310	315	1.000000000	1.000000000	1.000000000	0.999510807
315	320	1.000000000	1.000000000	1.000000000	1.000000000
320	325	1.000000000	1.000000000	1.000000000	1.000000000
325	330	1.000000000	1.000000000	1.000000000	1.000000000
330	335	1.000000000	1.000000000	1.000000000	1.000000000
335	340	1.000000000	1.000000000	1.000000000	1.000000000
340	345	1.000000000	1.000000000	1.000000000	1.000000000
345	350	1.000000000	1.000000000	1.000000000	1.000000000

7.2.4.4 Other Operating Modes

7.2.4.4.1 Startup and Shutdown

Rod positions and/or frequency of rod motion during startup and shutdown can be different than when at steady power. This makes the reactivity insertion potential different. An understanding of the rod characteristics during these modes of operation was obtained from the Operating Instructions and interviews with Operations personnel [7.10, 7.11, 7.12]. Operation reports for Runs 139 through 152 were reviewed to estimate the frequency of startups, restarts and shutdowns. The five month wait for DOE restart approval in 1989 was excluded for this purpose.

The goal was to construct a very simple, but not overly conservative, representation of reactivity insertion potential. Accordingly, the focus was strictly on lumping events into categories that have similar rod position and motion frequency characteristics. Two categories were identified. For each category, the parameters of interest are 1) magnitude of the potential reactivity insertion, 2) frequency of rod activation, 3) duration of the operation, and 4) frequency of the operation.

The most important category is termed the 14-in. bank configuration. In this configuration, the safety rods and several control rods are at 14 inches (fully inserted), several control rods are at 0 inches and one control rod is partially inserted. Rods at 14 inches obviously cannot insert any reactivity and rods at 0 inches can insert their full worth. (Table 7.25 shows the average and range of full rod worths.)

The number of control rods at 14 and 0 inches depends on the reactivity excess of the core loading. For the startup of Run 152A, which had a large initial excess, there were five control rods at 14 inches and four control rods at zero inches. This configuration may be approximated by four high worth control rods with their full worths available for a TOP accident, and the rest of the rods with no insertion potential.

The main operation where the rods are in this configuration is the startup period from first attainment of criticality to the time when rods are banked in a steady power configuration for the ascent to full power. This period typically lasts about four hours, but it can be as short as two hours and as long as 24 hours. This period includes rod calibration, which is not done in every startup but sufficiently often that it will be assumed always to occur. During this period the number of up-motion demands made for rods at 0 inches is approximately 15.

There are two other operations that can be represented by the 14-in. bank configuration. One is a period near the end of a normal shutdown when rods are rebanked from the steady power configuration to the 14-in. configuration. About 7 normal shutdowns are performed per year. Because of the burnup, few if any rods are at 0 inches and the reactor is kept critical in this configuration for only minutes. Thus this condition can be accounted for by an appropriate increase in the time estimate for the startup. The other operation to include here is the measurement of the prompt power coefficient. In recent runs, this has been done at every startup. The relevant event in this measurement is that Rod #1 is at 0 inches and latched for several minutes. The other rods are generally at their steady power positions. Rod #1 is a normal worth rod, so it is worth much less than the typical rod. Thus again an increase in the startup time estimate would account for this operation.

In summary, the 14-in. bank configuration has a relatively large reactivity magnitude potential but constitutes a very small fraction of the reactor's time at critical. On the order of 80 cents could be introduced and about four rods have this potential. About 15 up-motion demands are placed on these rods over a period of about 4 hours. This condition occurs about 10 times per year, the frequency of reactor startups. The roughly 40 hours per year in this state are to be compared with approximately 6900 hours per year in the steady-power mode.

The other category is characterized by a low controlling rod position, a steady-power configuration for the other rods and a high frequency of up-motion demands on the controlling rod. Included in this category are the ascent to authorized power during startups and restarts. The power ascent takes about 6 hours during a startup and 2 to 3 hours during a restart. During each of these operations there may be as many as 600 up-motion demands made on the

controlling rod. It takes about 200 hours of steady-power operation to equal this demand. The controlling rod is below its steady power position range much of this time. A typical 80 cent rod is capable of inserting as much as 60 cents at the beginning of this period if the final position is 9 inches. Its average insertion potential during these operations is about 36 cents, however. Startups occur about 10 times per year and restarts occur about 3 times per year.

The importance of this category depends on the principal failure mode leading to rod run-in. When the key parameter is hours of exposure, this category is unimportant since it accounts for just 1% of the time at critical. This feature is simply added to the steady-power pdfs in a manner similar to that used for the daily motion tests. When the number of up motion demands is the key, then it is quite important; the number of demands per year is about 40% as large as at steady power and the magnitude of the reactivity insertion is almost twice as large as at steady power.

Other operating modes present no significant rod run-in threat. Only down motion demands are made during normal power descent, anticipatory shutdown and scram. The reactor is made subcritical in about 2 hours during normal shutdown and much more quickly in the other shutdown operations. In restarts, the previous steady power bank position is established before the reactor is critical, so that this phase of restarts is not relevant.

7.2.4.4.2 Transient Experiments

This mode of operation is less constrained by fixed specifications than any other mode. Still, a reasonably clear picture of the rod run-in potential can be developed. The description presented here was developed from the Operations Reports for Runs 139 through 152, five Approved Procedures for experiments run in that period, and interviews with Operations personnel [7.10, 7.13, 7.14, 7.15].

In Runs 139 through 152 there was an average of five transient experiments per year. The ACRDS was used in more than half of these, with high speed mode used in one case and slow speed mode in the others. High-worth rods were always used in controlling the transients.

The transients always lasted less than two hours, usually much less.

The ACRDS was used in slow speed mode to produce either reactivity oscillations or a nearly step increase in power. There is only one ACRDS-controlled up-motion demand for the step power increase but there could be hundreds for reactivity oscillations. In either case Rod #4, which is the one the ACRDS controls, was positioned between 6 and 8 inches because the worth slope is constant in that vicinity. It is assumed that the rod would always be at least at 5.5 inches because of the desirability of having a constant worth slope. From 5.5 inches it could insert up to 55 cents at a maximum rate of 0.58 cents/s.

The ACRDS can be programmed to respond to reactor power or to rod position. When programmed to maintain a steady power, the ACRDS could, in principle, defeat passive feedbacks if a LOF or LOHS accident were to occur during a transient experiment [7.16]. Two things protect against this, a bulk sodium high temperature interlock and an Emergency Procedures requirement for manual scram if either of these accidents occurs. Thus, for there to be an accidental rod run-in where the ACRDS defeats passive safety, the probabilities must be combined for 1) LOF or LOHS occurrence during a transient experiment, 2) accidental run-in, 3) interlock failure, and 4) scram failure. Experiments are excluded from the EBR-II PRA, but it is expected the combined probabilities would be below the threshold of interest.

The one use of the high speed mode is described because there is a possibility that the procedure will become a routine way of measuring prompt reactivity feedback, replacing rod drop measurements. In this experiment, EX-234, a control rod worth 87 cents was moved in a ± 0.24 inch range about the 7 inch position. In a run-in accident from this position, it would have had the potential to insert about 44 cents at a maximum rate of 8.6 cents/s. The power oscillated between 57 and 59 MWt. The ACRDS was programmed to respond to rod position. The experiment lasted about one hour, during which time the rod was moved up and down roughly 500 times. If this becomes a routine measurement, it would occur up to 10 times per year.

As in the slow speed case, it is reasonable to assume that the high-speed rod would be positioned above 5.5 inches in order to keep it in the range of the constant worth slope. From there, a 90 cent rod could introduce 55 cents. Hard-wired high and low position limit switches exist that trip the rod to manual control; a requirement to keep the rod above 5.5 inches would therefore be easy to ensure.

In the experiments where all rods were under manual control, there were no extraordinary rod positions or up motion demand rates. The lowest position seems to have been 8 inches.

With the exception of high-speed ACRDS use, it is reasonable to argue that plant transient experiments are covered by the treatments of steady-power and startup operations. The reactivity insertion rates are the same as in those operations. Compared to the 14-in. bank configuration in startups, the rod positions are higher and the frequency is less for transients. The failure probability for the ACRDS control system is not compared to that for manual activation of control rods, since plant experiments are excluded.

High speed operation of the ACRDS has to be treated separately because of the uniquely high reactivity insertion rate. Currently this mode is used very infrequently but it might be desirable to anticipate that it could be used ~ 10 times/year in the future. A maximum insertion of 55 cents should be assumed, as discussed earlier. It would have to be stated clearly in the PRA that the analysis is valid only for Rod #4 positioned higher than 5.5 inches.

In estimating a rod run-in frequency for plant experiments, it should be recognized that each such experiment is reviewed and approved before it is run. It must be shown that no damage would occur in protected accidents to gain approval. Fast-speed ACRDS experiments require a safety analysis demonstrating a damage-free outcome if a 12 cent/s rod run-in (with scram) were to occur (Sections 3.18 and 7 of the Technical Specifications [7.5]).

7.2.4.5 Summary

Most of the important results are summarized in Table 7.32. The parameters include the initial rod position (Z), the insertion magnitude (R_i) and the insertion rate. Both median and 99.99 percentile values of the insertion magnitude and insertion rate are given. (When these were taken from pdfs, they are the upper limit of the appropriate bin.) Up-demands and hours per year should be useful in a failure analysis to determine run-in frequencies. The situations are listed across the top.

In steady-power operation, there are three categories of rods, which are distinguished by the rod design and/or the way they are used. Table 7.34 shows just two characteristic insertion magnitudes from the pdfs developed in Section 7.2.4.3. Safety rods have two entries for some parameters; the first applies if exposure time determines the failure frequency and the second applies if up-demands do. In either case, the safety rods have a lower run-in frequency than the other two types but they have a larger insertion magnitude. The controlling rod is by far the most likely one to run-in if up-demand frequency is key.

Run-in of multiple rods is anticipated to be a very rare event. Consequently, the only situation considered in this analysis is run-in of two control rods during steady power operation. (The pair of safety rods is treated as a single rod because the two rods are driven as a unit.) This situation has a higher insertion magnitude and rate than any other at steady power.

Although not analyzed in the EBR-II PRA, two situations were identified for startup and shutdown operations. The 14 in. bank configuration occurs most often in the initial critical phase of startups. It has a low occurrence frequency but the largest insertion magnitude of any operation. It is reasonably conservative to treat this configuration as having four rods at 0 inches and the rest at 14 inches. The normal bank configuration occurs in the next phase of startups and during restarts. It is distinguished from the true steady-power configuration by the controlling rod characteristics, which are what is shown in the table; the controlling rod has a lower position, so a larger R_i , and a large number of up-demands per hour.

Table 7.34. Summary of Rod Run-in Characteristics

	<u>Steady-Power Operation</u>				<u>Startup & Shutdown</u>		<u>Transients</u>
	<u>Single Rod Run-in</u>				14-in. Bank	Normal Bank	
	<u>Safety</u>	<u>Cntl CR</u>	<u>Banked CRs</u>	<u>Two CRs</u>	<u>O-in. Rod</u>	<u>Cntl CR</u>	<u>Fast ACRDS</u>
Median Z (in.)	7.5, 10.8	10.9	10.3	10.4	0.0	7.5	-
Median R _i (¢)	50, 27	15	20	30	79	35	-
99.99% R _i (¢)	55	35	40	70	99	81	55
Median Rate (¢/s)	0.29	0.42	0.42	0.84	0.44	0.42	7.7
Max. Rate (¢/s)	0.47	0.64	0.64	1.22	0.64	0.64	8.8
Up-Demands/yr	26	20700	288	-	150	7800	5000
Oper. Hours/yr	2900	6900	6900	6900	40	68	10

7.3 Aging

Aging of vital pieces of equipment, as a concern at EBR-II, has been and continues to be addressed by a comprehensive program of analysis, tracking, and plant modification. This is exemplified by the report, "Extending the Operating Lifetime of EBR-II to 30 Years and Beyond" [7.17], which addressed all the concerns associated with aging of major pieces of equipment as well as other issues such as ensuring continuity of staff knowledge as knowledgeable engineers retire. Currently, (1990), in progress is an extension of this study to a 40 year life [7.18]. Results presented in this section will draw upon work completed for this project. In general we one asks whether there are aging effects to be accounted for in the initiating event quantification, and in the safety system response (systems analyses). These will be discussed separately below.

7.3.1 Aging Effects in Initiating Events

In Section 5, the initiating event frequencies were derived using a number of methods, historical data based occurrences, fault tree analysis, and expert judgement. The aging factors for the initiating events have been considered differently depending on the method of deriving the initiating event frequency. Section 7.3.1.1 describes the method used for historical data. Section 7.3.2 evaluates aging at a system level for those initiating events derived by system fault trees and for the response of mitigations system. Section 7.3.1.2 is focused on the aging component of structural failures which have never happened and are not tractable using fault tree analysis and therefore rely on expert judgement supported by other means of analysis. Some initiating events, especially local facts, since these involve fuel and or experimental subassemblies which are in essence consumables and replaced at regular intervals, have no aging effects.

7.3.1.1 Initiating Events Quantification

The initiating event frequencies based upon direct observation of EBR-II experience were grouped into 5 year bins in order to detect any trends. In all cases, except one,

the trend was downwards (less events in recent history), or no trend was discernable. The apparent exception being in the secondary sodium systems (TSEC). On closer inspection the cause of the apparent increase was two software errors in loading a controller for a test program which drove the secondary flow to zero. Therefore no specific aging concerns were identified by data-based initiating event frequencies.

7.3.1.2 Structural Failures

As part of the EBR-II PRA, the likelihood and consequences of structural failures must be considered. As part of the analysis required to support the extension of the operating life of EBR-II to 40 years, structural analysis has been performed on most of the major components [7.19]. The methodology applied is described in an earlier work [7.20], upon which many of the conclusions are based. In this section, the methodology of the above analyses is presented briefly in conjunction with a simple interpretation in a probabilistic sense. The results are then given as an annual frequency of failure for all major components. Finally the results and their implications are discussed.

7.3.1.2.1 Materials Aging Assessment

A forty-year lifetime for the EBR-II plant requires that structural and irreplaceable components perform in service for more time than has been experienced elsewhere. This requires a careful study of the materials aging factors of these components. The structures within the primary system are largely constructed of austenitic stainless steel, primarily Type-304. The austenitic stainless steels are particularly resistant to mechanical property degradation and environmental factors. The particular conditions to which the materials are subjected are a long-term thermal environment, irradiation-induced modification of materials properties, and long-term compatibility with sodium. Therefore, the property of primary interest is mechanical integrity. Of additional interest is the geometrical stability of the grid plate in that it fixes the position of all fueled assemblies within the primary system.

Embrittlement and Impact Properties

A potential safety concern for nuclear reactors is the inservice embrittlement of reactor vessel materials. While there are problems of this kind of both water-cooled and liquid-metal-cooled systems, the origin and degree of these concerns are quite different for each system type. Liquid-metal-cooled reactors (such as EBR-II) are constructed from stainless austenitic steels while water-cooled reactor vessels are constructed using ferritic steels. The chemical composition of the two types of steel, and resultant crystal structure are very different, which results in differences in the embrittlement phenomenon observed for each system.

Reactor vessel embrittlement, manifest as a reduction in ductility when subjected to long-term loading, sudden loading (impact), or cyclic loading (fatigue), is caused by several factors. Neutron irradiation creates atomic displacements in vessel materials that result in increases in dislocation densities similar to work hardening. These dislocation densities tend to saturate with neutron dose and, therefore, the associated embrittlement phenomenon also tends to saturate at some given loss of ductility. The degree of ductility loss is often dependent upon temperature because dislocations are removed by application of high temperatures (annealing), the higher the temperature, the faster is the removal, and thus the saturation in ductility loss is less severe.

Austenitic steels are more susceptible to ductility loss caused by work-hardening, so liquid-metal-cooled reactors are more subject to this type of embrittlement. Austenitic steels, however, often have inherently high initial ductilities and therefore observed ductility losses result in a material that retains adequate ductility as not to present a safety concern. Moreover, neutron doses required to cause the embrittlement saturation will not occur at reactor vessel locations over the usual lifetime of a nuclear reactor. Reactor vessel embrittlement is therefore not likely to be severe or to pose safety implications for liquid-metal-cooled systems.

A similar embrittlement phenomenon occurs in the ferritic materials used in water-cooled reactors, where certain tramp or chemical-alloying elements (copper, nickel, phosphorus) tend to cluster and create crystal defects which impede plastic flow. This effect tends to occur at relatively low neutron doses when these chemical elements are present. Of paramount

importance to safety concerns is that the embrittlement phenomenon strongly affects the ability of the material to withstand impact loading. Ferritic steels are subject to a phenomenon where the material suddenly loses all ductility and its ability to absorb impact energy without breaking. The temperature where this occurs is normally quite low, and is called the nil ductility temperature (NDT). Irradiation causes the NDT to rise to values within the nuclear reactor operational ranges. Austenitic steels used in liquid-metal-cooled reactors do not experience NDT.

Other embrittling phenomena are related to absorbed gases (helium and hydrogen) in the metal. Austenitic steels have a high nickel content, which can produce helium in a neutron irradiation environment. Because the irradiation causes both helium embrittlement and dislocation embrittlement concurrently, the effects are not easily separated and are always measured together. Therefore, the previous discussion concerning the lack of safety issues regarding the embrittlement of austenitic stainless steel vessel components in liquid-metal-cooled reactors remains valid for irradiation hardening and He-embrittlement effects combined. Moreover, recent evidence has shown that helium embrittlement is of no concern for irradiation temperatures of less than 932-1022°F; EBR-II inlet and outlet temperatures are 700°F and 883°F, respectively. Therefore, no major core structural components will be affected. This finding is supported by the data from the surveillance capsules and control-rod-thimble replacement.

Another embrittling phenomenon that may become important to a 40-year lifetime is thermal embrittlement of Type-304 stainless steel, and particularly of welds mating Type-304 to Type-304 using Type-308 stainless steel as a filler material. The primary tank is constructed of 304 plate sections welded together using 308 welding electrodes.

The embrittlement phenomenon is characterized by degradation of the free ferrite in cast 304 or in the 304/308 welds after long periods of time due to phase transformations and formation of precipitates. EBR-II did not use a pure case form of 304 (CR-8 or CF-8A) in the construction of the primary tank so this should not be a problem. In addition, the potential embrittlement problem was anticipated by the designers of the reactor and weld criteria were

specified to limit the ferrite content in both weld rods used and in the welds themselves by adhering to ASTM Specification A240-57T with an additions requirement the maximum carbon content be 0.08 percent, with 0.04 to 0.06 percent preferred. By working in close cooperation with the primary tank fabricator, all steel used met the desired requirements.

Void Swelling

Neutron-irradiation damage of stainless steel components eventually creates a density-change in the stainless steel. Voids that form in the material lowers the density, and the result is that the component swells. The behavior has been studied extensively, and the amount of swelling can be fairly accurately predicted by knowing the fast ($E > 0.1$ MeV) neutron fluence exposure and the irradiation temperature.

The core contains many different types of stainless steel and the swelling behavior of each, and allowable dimensional changes of each component are used to provide exposure limitations. In the core, where the neutron flux is high, these exposure limits are used to assess when a component assembly needs to be replaced. Fuel handling force surveillance is eventually utilized to monitor whether an exposure limit has been properly assigned and prevent unexpected problems from occurring.

The areas surrounding the main core contain structures which are irreplaceable such as the grid plate assembly, the reactor vessel, and the neutron shield. The only steel used within this region is Type-304. Because the components are irreplaceable, the swelling behavior is assessed to determine the useful lifetime of these components as a measure of the reactor lifetime. The neutron flux to which these components are exposed is much less than that of in-core components.

The most critical of these components, in terms of swelling-related problems is the grid plates. There is a higher neutron exposure to the upper than lower plate. It was proposed that differential swelling may cause crowning of the upper plate and/or misalignment of the holes in the plates. A program to assess the problem and monitor the hole alignment was instituted in

1974, and no problems have been found during periodic gauging of the grid. As a precaution, the diameter of the lower part of the pole piece of some subassemblies has been reduced to insure that no binding occurs.

The expected peak fast neutron exposure on the grid plate will be less than 2×10^{22} n/cm² in the year 2004. The predicted void swelling ($\Delta V/V_0$) at that point is predicted to be less than 0.1%. This indicates that grid plate crowning is not likely the problem, but the monitoring program will insure detection if it should occur.

Other irreplaceable components are not exposed to as high a neutron flux, and their dimensional changes are not as critical. Therefore, the void swelling assessment will be limited to the grid plate.

Sodium Compatibility

The exposure of steel to sodium has also been widely studied. The high-temperature corrosion effects have been observed characterized by changes in the near-surface chemistry of the steel (nickel loss, etc.) and also carbon transfer out of the steel with perhaps a carburization of other components in the reactor. The operating temperature within the EBR-II reactor systems and the materials used do not create a situation where major sodium compatibility effects would be seen.

7.3.1.2.2 Structural Assessment

The basic structural analysis is described in [7.20]. Briefly, standard codes are used for thermal hydraulic and structural analysis, THTB is used for thermal analysis and ANSYS and SAP6 used for structural and piping analyses. The components considered include the reactor vessel, the reactor vessel cover, the intermediate heat exchanger (IHX), the superheater, the primary piping system, and some of the secondary piping between the IHX and the superheater.

(1) Reactor Vessel Cover

The total fluence per MWd at the reactor cover bottom plate above the core rows 1 and 16 locations are given in Ref. 7.31 and they are 1.4×10^{16} and 5.3×10^{15} n/cm² for year reactor operation assuming a plant factor of 80%. The expected fatigue damage factor of the thermal baffle is 1.5×10^{-3} which is considerably less than 1, and the fatigue damage is within the limit specified by the ASME code [7.21].

(2) Reactor Vessel

The creep and fatigue damage of the reactor vessel was found to have negligible effect for reactor normal and transient operations [7.17]. The irradiation damage of the reactor vessel is small, and the stress and strain ranges of the reactor vessel are within the design limit specified by the ASME code for EBR-II 40 year operation [7.19].

(3) Intermediate Heat Exchanger (IHX)

The irradiation damage of components in the IHX is small and can be neglected. Analysis in [7.20] indicated that the creep damage of the IHX during reactor normal operation is negligible. A reactor scram from 123% overpower was identified as the most severe transient for the IHX components due to fatigue damage, and 1000 duty cycles are allowed for the transient through the life of the reactor. The structural damage for steady-state and other transient operations are small and can be neglected. If we conservatively use this transient to represent all protected overpower transient occurring during EBR-II operation, then the IHX can survive at least 1000 protected overpower transients. In the past, the EBR-II experienced 249 reactor scram transients [7.22]. If we conservatively assume that 250 additional overpower transients are expected in the future reactor operation, then the total number of the protected overpower transients throughout the life of the reactor is about 500, which is less than 1000 (corresponds to damage factor of 0.5), and the design criterion is satisfied.

(4) Piping

The analysis in [7.20] indicated that the fatigue and creep damage to each of the primary-inlet, the primary-outlet (Z-pipe), and the secondary piping systems are very small for transient and steady-state reactor operations. To extend the EBR-II life to 40 years will not affect the structural performance of piping systems.

(5) Primary Tank

The primary tank includes an inner vessel and an outer vessel. The primary tank inner vessel is suspended from the primary-tank cover, and the load of the primary tank is transmitted from the six radial beams in the primary-tank cover to the six radial beams in the top support structure through six stainless steel hangers. The average primary-tank sodium temperature is 700°F during normal operation, and a temperature gradient across the cover occurs. As a result, a downward bowing in the primary-tank cover is induced, which causes a moment and a shear force at the top of the inner-vessel tank.

During reactor normal and transient operations, a temperature difference between top and bottom of the shell occurs due to stratification. A study of the measured data recorded by the primary-tank probes for the SHRT tests and the plant inherent control test (PICT) series [7.23] indicated that the maximum axial temperature difference in the inner vessel will not exceed 200°F, and the maximum temperature across the primary cover is below 750°F. For an axial vessel temperature difference and the temperature across the cover are 200°F and 750°F, respectively, the stress intensities at the top and the bottom of the inner vessel are 15,000 and 14,800 psi, respectively. These stress intensities are considerably less than the $3 S_m$ (44,000 psi) allowed by the ASME code and the design criterion is satisfied.

7.3.1.2.3 Damage Criteria

Creep and fatigue damage were evaluated in the preceding analysis using the ASME code [7.19], a total creep fatigue damage, D , being

evaluated. A design life is defined as $D=1$, but this does not give a failure probability.

Consider first the case when the damage is being caused by transients, either protected overpower or loss-of-flow transients and assume that N of the limiting transients are required to give $D=1$.

Hence: $Nd_T = 1$ (7.3)

where d_T is the damage in each transient. Further, let us conservatively assume that the effects of every scram at EBR-II gives damage d_T and that there are n_s scrams per year. Therefore, using (7.3) the design life of a component, A , is given by

$$An_s d_T = 1 \quad \text{or } A = 1/(n_s d_T) \quad (7.4)$$

To transform this into an estimation of probability requires an interpretation of design life.

Here the assumption is made that design life can be interpreted as a "2- σ " probability of failure, i.e., if $D=1$ then, using a normal distribution

$$\text{Pr (component failure)} = 2.3 \cdot 10^{-2} \quad (7.5)$$

Consultation with structural engineering consultants indicated that this assumption is conservative [7.24].

Now from Table 5.6, the number of scrams is $\sim 4.4/\text{yr}$ and if a uniform failure rate model is assumed the annual frequency of failure can be estimated as

$$\begin{aligned} & (2.3 \cdot 10^{-1} d_T) \times (4.4 \text{ yr}^{-1}) \\ & \simeq 0.1 d_T \text{ yr}^{-1} \end{aligned} \tag{7.6}$$

Table 7.35. Structural Analyses of Major EBR-II Components

	d_T	Mean Failure Freq. yr^{-1}
Reactor Vessel Core	$6.67 \cdot 10^{-6}$	$6.7 \cdot 10^{-7}$
Reactor Vessel	$< 10^{-6}$	$< 10^{-7}$
IHX	$1 \cdot 10^{-3}$	$1 \cdot 10^{-4}$
Super Heater (Fig. 7.6)		
Region 1 Shield Transition	10^{-6}	$1 \cdot 10^{-7}$
2 Duplex Tube in air	$1.4 \cdot 10^{-5}$	$1.4 \cdot 10^{-6}$
3 Duplex Tube in sodium	$< 10^{-6}$	$< 1 \cdot 10^{-7}$
4 Steam Reducer	$< 10^{-6}$	$< 1 \cdot 10^{-7}$
Piping	$< 10^{-6}$	$< 10^{-7}$
Tank	$< 10^{-6}$	$< 10^{-7}$
Evaporator	$< 10^{-6}$	$< 10^{-7}$

Fig. 7.6. Schematic View of Superheater, Showing Regions for Which Thermal-analysis Models Were Developed

The table shows that the IHX is most vulnerable to failure due to repeated thermal cycling. IHX failure is treated in the LOCA analysis, where a higher failure rate than that derived above is used.

Additional studies have been carried out upon the grid plate and no embrittlement or swelling concerns have been identified as shown in the preceeding section. This conclusion is based upon the performance of surveillance materials irradiated to fluxes greater than experienced by any component in the reactor tank.

The concern for superheater problems lies in sodium-water reactions. This is considered to be an independent initiating event (SGTR). A generic estimate for a tube failure of 10^{-6} failure hr^{-1} , $7.2 \cdot 10^{-3} \text{ yr}^{-1}$, is available [7.25]. This generic estimate is based upon data from sodium steam generators of the 1960s and steam/water steam generators.

The same structural evaluation identified creep in steady operation as the life limiting damage mechanism affecting the superheaters. The analysis of [7.20] identifies a creep damage factor of 0.43 over 40 years with an 80% capacity factor, or a 93 year design life, for the limiting case of the transition region where the creep damage is greatest. If a design life is interpreted, as above, as a " 2σ " probability of failure then the failure rate can be estimated as $2.5 \cdot 10^{-4} \text{ yr}^{-1}$. The failure rates evaluated in this way for the other regions of the superheater give the following values $7.5 \cdot 10^{-6} \text{ yr}^{-1}$, $2.5 \cdot 10^{-6} \text{ yr}^{-1}$ and $7.5 \cdot 10^{-6} \text{ yr}^{-1}$.

Failure in the transition would not lead to immediate sodium-water contact, nevertheless in view of the simplicity of the arguments presented here, an estimated value of 10^{-4} yr^{-1} will be adopted for steam generator tube rupture with sodium water interaction.

7.3.2 Aging Effects at the Component Level

Aging effects at the component level can affect the reliability of the safety systems. In most cases the EBR-II systems are undergoing continual upgrading. Therefore,

where aging concerns might have existed, these have been addressed. The issue of aging is addressed in a system by system basis below.

7.3.2.1 Reactor Shutdown System (RSS)

One common cause failure in the RSS was identified which can be considered as due to an aging mechanism. This event is the failure of the relays to act upon demand. The specific mechanism is wear due to 60 Hz vibration for relays power by AC. The relays in the RSS are to be replaced with new relays with hardened shafts that guard against this failure.

Another symptom of aging in EBR-II is reflected in the steady loss of flow instrumentation. This is reflected in the PRA by the various possible trip configurations and specifically in that a flow transient on loop #1 will not be sensed directly because there is no remaining flow instrumentation there.

7.3.2.2 Primary Pumps

Primary pump #1 was removed for repair in December 1970 and again in May 1988. Primary pump #2 was removed for repair in May 1982. Primary pump #2 was removed for repair in May 1982. The impellers of both pumps were replaced during their last removal from the primary tank. Inspection of the pump shafts and associated revealed no degradation of the pumps. The pump motors were inspected and found to be in "like new" conditions. This is primarily the result of the argon atmosphere within the motor housings. Thus, there appears to be no aging effects other than impeller wear which has been accommodated by replacement of worn parts.

7.4 Dependency Analysis

Dependent failures are extremely important in PRA and must be carefully treated in order to avoid underestimation of risk [7.26, 7.27]. Dependent failures are those failures that defeat

the redundancy or diversity which is usually utilized to improve the availability of critical plant functions such as reactor abnormal-transient detection. In the absence of dependent failures, separate trains of a redundant system, or diverse methods of providing the same function are regarded as independent so that unavailability of the function is basically the product of the unavailabilities of the separate trains or diverse systems. However, a dependent failure arises from some cause that fails more than one system, or more than one train of a system, simultaneously. Accordingly, the effect of dependent failures is to largely increase (by several orders of magnitudes in some cases) the unavailability of the system with respect to the situation of true independence.

Generally, dependent failures could result from initiating events (external events), interassembly dependencies, and intercomponents dependencies [7.26]. Intersystem and intercomponent dependencies cover functional dependence (for example, when a system is designed to function only in conjunction with the successful operation of other systems), shared equipment dependence, physical interaction dependence (for example, extreme environmental stresses created by the failure of one system), and human interaction dependence. Most of these dependencies are usually accounted for explicitly in the PRA logic models (event trees and fault trees). For example, intersystem dependence arising from shared equipments, such as the undervoltage CP-relays in the shutdown strings shared by the low-flow channels and subassembly outlet temperature channels, are included in the logic model by introducing basic events, which represent component failure modes associated with the shared equipments. Failures resulting from the failure of another component (cascading or propagating failures) are also modeled explicitly. Similarly, errors made during maintenance and testing of component operability in redundant channels are modeled explicitly in fault trees.

However, many dependencies among component failures can not be accounted for explicitly in the logic models, and this means that some basic events are not truly independent. This is addressed by introducing the concept of common cause basic events, which represent the class of residual dependent failures whose root causes are not explicitly modeled. Thus, a common cause event is defined as the failure or unavailable state of more than one component at the same time or within a short interval and due to some shared cause.

There are three important concepts in understanding and modeling common cause events [7.27]. The first is the root cause of failure or unavailability of the component. A root cause is a mechanism of a transition of state from operating or functionally available to that of failed or unavailable. Given the existence of the root cause, the second concept of importance is that of linking or coupling mechanism, which is what leads to multiple equipment failure. Whereas, each component fails because of its susceptibility to the conditions created by the root cause, the coupling mechanism makes those conditions common to several components. The third factor that enters into determining the potential for dependent failures is the existence of engineered or operational defenses against unanticipated equipment failures. Some of the tactics adopted in defense against CCFs include design control, physical separation of equipments, well-designed test and inspection procedures, maintenance procedures, and training of personnel.

Common cause failures among components can result from various root causes and by different mechanisms. The following list broadly categorizes possible root causes of component failure. This list has been used in identifying and assessing potential CCFs in EBR-II systems.

1. Design Inadequacy - This category covers actions and decisions during design, manufacturing, or installation of components both before and after the reactor is operational.
2. Procedure Error - This category refers to the ambiguity, incompleteness, or errors in procedures for operation and maintenance of equipment.
3. Human Actions - This represents causes related to errors of omission and commission on the part of the plant staff, such as failure to follow a correct procedure.
4. Abnormal Environmental Stress - This includes all causes related to a harsh environment that is not within the specified design criteria of the components.
5. Internal Failure - This encompasses malfunctioning of something internal within the component boundary as a result of normal wear-out or other intrinsic failures.
6. Unknown - The cause of the failure can not be identified.

The key output based on the previous guidelines is the identification of the groups of components for which CCFs may be important. The objective of the next stage is to modify the logic model to incorporate common cause events, convert this logic model to a probability estimation model, and to analyze the data for quantifying the parameters of the selected probability model [7.27]. The number of parameters in the model, single or multiple, determines the number of the basic events required in the logic model and obviously the number of the parameters to be derived from the data analysis. The most widely used single parameter model is known as the beta-factor model. According to this model a function (beta) of the component failure rate can be associated with common cause events shared by the other components in that group. Whenever a common cause event occurs, all components within the common cause component group are assumed to fail. Although data collected from the operation of nuclear power plants (mainly LWR plants) indicate that common cause events do not always fail all redundant components, experience from using this model shows that, in many cases it gives reasonably accurate but slightly conservative results for redundancy levels up to about three or four items. Beyond such a redundancy level, this model generally yields results that are conservative. A more accurate analysis model for systems with higher level of redundancy is the Multiple Greek Letter (MGL) model. This multiple parameter model is an extension of the beta method. For n redundant components the model generally requires n different parameters which are used to quantify all possible ways of CCFs.

Ideally, the numerical values of the model parameters (only a single parameter is required with the beta-factor method) should be estimated in a manner that makes the maximum possible use of plant specific event data. This would include data screening and classification of event data and developing uncertainty distributions and point estimate of the parameters. However, due to virtually non-existent common cause events in the EBR-II operating history, such a detailed data analysis becomes futile and one might need to consider data from the LWR operating experience and a variety of sources to make statistical inferences about the frequencies of common cause events [7.28, 7.29, 7.30]. It is worth noting in this regard that the procedures for treating common cause failures described in Ref. 7.27 recognized the need in some analyses to bypass the event screening step and use instead published numerical values of common cause event parameters. Accordingly, the common cause failures in the RSS fault trees have been

addressed in terms of generic beta factors as follows:

- The total failure probability of each redundant component is expanded into dependent and independent failure contributions. The probability of dependent failure is assumed equal to the total failure probability of each component multiplied by the beta factor. The independent failure probability of each component would then normally be the difference between the total failure probability and dependent failure probability. But since independent component failures in the RSS do not contribute any significant amounts to the system unavailability, it was found convenient (data checking) in the analyses to have the independent failure probability the same as the total failure probability.
- The generic beta factors are applied to similar components in any subsystem or in redundant subsystems or strings. These similar components have to meet the following CCF requirements: (i) same initial conditions, (ii) same failure mode, (iii) same use or function, and (iv) same design or same manufacturer.
- To account for the effect of number of redundancy, a generic beta factor of 0.100 is used for redundant components numbered less than four, 0.020 for redundant components numbered four or five, and 0.008 for redundant components numbered six or more. Combined with this method of modeling CCFs is a substantial effort to explicitly model most of the unsafe dependent events. Any remaining dependent failures can be viewed as CCF with weak linkage between redundant components. An example of this weak linkage is wear-out of parts of components by extended usage. Weak linkage implies that although a linkage mechanism can be postulated, its probability of occurrence in all components at the same time gets smaller with increase in redundancy level.

The main advantage of the above method of addressing CCFs, to be referred to herein as beta(n) method, is that it is much simpler to use with fault trees than the MGL method. The

beta(n) method requires only one basic event to model the CCFs irrespective of the number of redundant components. Whereas the MGL method requires a much larger number of basic events to model the CCFs. For instance, with four redundant components eleven different CCF events have to be modeled in addition to the independent failures of the components. One should notice, however, that the beta(n) method is an approximate method for taking account of redundancy level. It is a convenient means for assessing the CCFs among redundant components when there are no CCF data available and the alternatives would be either to ignore the CCFs all together or use a fixed beta factor that does not vary with the redundancy level. It is needless to say that the beta(n) method is not a substitute for the more precise MGL method when reliable CCF data are available.

A better perspective of the present beta(n) method can be gained when one compares the failure probabilities of redundant components determined by this method and by the alternative MGL method. Three different generic models (labeled Model 1, 2, and 3) are selected for the MGL method calculations, see Table 7.36. The CCF parameters used in the three models are based on the beta and gamma values given in Table 3-5 of Ref 7.31 and Table 11 of Ref 7.32. The parameter delta is assigned a value of 0.8 in Models 1 and 2 to provide a smooth transition from gamma=0.5 to epsilon=1. A more gradual transition is assumed in Model 3 with gamma=0.2, delta=0.4, and epsilon=1. Although any of the three models could be justified as a generic CCF model for a group of redundant components, Model 1 appears to be an over-pessimistic model for components that have not shown in the past clear CCFs. Either Model 2 or Model 3 would more appropriately represent postulated CCFs for this type of redundant components or for redundant components with weak linkage. Model 3 is particularly important in view of a recent evaluation of CCFs among relays (PLG proprietary data). The MGL parameters of the relays indicate that Model 3 is not only realistic but conservative as well. Table 7.37 and Fig. 7.7 show the probability of failure of m components among n redundant components ($Q(m/n)$) according to the beta(n) method and according to the three models based on the MGL method. The results of the beta(n) method compares very favorably with the results of Models 2 and 3 in almost all cases. It is interesting to notice that the two alternative methods yield identical results for $n=6$ and $m=n-1$. Common cause failure at this high level of redundancy is particularly important for the EBR-II Reactor Shutdown System, see

Table 7.36. Parameters Used in Modeling CCF by $\beta(n)$ Method and MGL Method

<u>$\gamma(n)$ Method⁽¹⁾</u>		<u>MGL Method⁽²⁾</u>			
n	<u>$\beta(n)$</u>	<u>Parameter</u>	<u>1</u>	<u>2</u>	<u>3</u>
2	0.100	β	0.10	0.02	0.10
3	0.100	γ	0.50	0.50	0.20
4	0.020	γ	0.80	0.80	0.40
5	0.020	ϵ	1.00	1.00	1.00
6	0.008				

Notes

(1) $Q_t = \beta(n) Q_s + Q_s$

Q_t = Total Failure Probability of a Redundant Component

Q_s = Single Component Failure Probability

n = Number of Redundant Components

(2) Provides $Q(m/n)$, Probability of m Components failing among n Redundant Components. As an example:

$$Q(2/3) = \beta\gamma Q_s + (3/2)\beta(1-\gamma)Q_s + 3(1-\beta)^2 Q_s^2$$

Table 7.37. Comparison of Redundant Component Failures According to $\gamma(n)$ Method and MGL Methods

(m/n)	$\frac{Q(m/n)(1)}{Q_s}$						
	$\beta(n)$	MGL(2)			$\beta(n)/MGL$		
		<u>1</u>	<u>2</u>	<u>3</u>	<u>1</u>	<u>2</u>	<u>3</u>
(2/2)	0.101	0.101	0.021	0.101	1.00	4.81	1.00
(2/3)	0.103	0.127	0.028	0.142	0.81	3.68	0.73
(3/3)	0.100	0.050	0.010	0.020	2.00	10.00	5.00
(4/4)	0.020	0.054	0.011	0.024	0.37	1.82	0.83
(4/5)	0.020	0.040	0.008	0.008	0.50	2.50	2.50
(5/5)	0.020	0.040	0.008	0.008	0.50	2.50	2.50
(5/6)	0.008	0.040	0.008	0.008	0.20	1.00	1.00
(6/6)	0.008	0.040	0.008	0.008	0.20	1.00	1.00

Notes:

- (1) Q_s : Single Component Failure Probability ($Q_s = 10^{-3}$, used only with nonlinear terms)

$Q(m/n)$: Probability of m components failing among n redundant components

- (2) See Table 7.36 for the MGL parameters of each case.

Fig. 7.7. Comparison of Redundant Component Failure Probability According to Beta(n) and MGL Methods

Section 6.2.1 and Appendix C.1. The beta(n) method, however, appears to under-predict failure probability when one compares its results with Model 1 for $n = 4$. As mentioned earlier Model 1 is an over-pessimistic model for redundant components that have not shown CCF during their service in the past. The inability of the beta(n) method to differentiate between the various m values for a given n value appears to be significant only with $n=3$. The three MGL models show, as one would expect, larger failure probabilities with $Q(2/3)$ than with $Q(3/3)$; whereas the beta(n) method has virtually the same value for both types of failure logic. For redundant components that have 2-out-of-3 failure logic, this situation can be simply rectified by raising the beta-factor from 0.10 to 0.15. Because of the relatively small difference in the failure probability and in order to keep the beta(n) method as simple as possible it was found unnecessary to modify the method for this special case.

Table 7.38 shows the beta factors among the various electrical components that are used in the LOF and TOP fault trees of the RSS. The beta factors used in the accident sequence evaluations are the values listed under Analysis Model B. These beta factors are based on the previously described beta(n) method and cover all similar components in various channels or strings. Table 7.38 also shows beta factors for a different analysis model, designated Analysis Model A, with less extensive CCF modeling. In Analysis Model A, one assumes relatively large beta factors for similar components within the same channel or same shutdown string (larger than would have been used with beta(n) method) but ignores CCF between similar components in different channels or shutdown strings. The comparison between the results of Analyses A and B given in Section 6.2.1.5 provides insights into the effect of CCFs among the RSS critical components.

In addition to CCFs among electrical components, CCFs among the nine high-worth control and among the two safety rods have been given careful assessment. Table 7.39 shows the common cause events that are postulated and used in both fault trees to model the mechanical failure of the control and safety rods. The failure probability calculations begin with a generic failure per demand of $3 \cdot 10^{-5}$ for a single rod [7.25]. This is the failure probability of a control rod or safety exclusive of the failure of the magnetic clutch to release the rod. In the case of control rods, this generic failure probability is increased by a factor of three, to $9 \cdot 10^{-5}$ per

Table 7.38

LIST OF COMMON CAUSE FAILURES AMONG ELECTRICAL COMPONENTS -
FAULT TREE ANALYSIS OF SCRAM FAILURE UNDER LOF OR
TOP CONDITION

Description of CCF	Analysis Model	β Factor	CCF Event	Corresponding Independent Events
<u>ALL FAULT TREES</u>				
1. Armatures/coils of all CP-relays on shutdown string A(1)	A	0.30	RREC01QU	RRECPAQP, RRECC1QP, RRECP2QP, RRECPYQP, RRESR0QP, RRESRXQP
2. Contact of all CP-relays on shutdown string A(1)	A	0.30	RCZC01QU	RCZCPAQP, RCZCC1QP, RCZCP2QP, RCZCPYQP, RCZSR0QP, RCZSRXQP
3. Armatures/coils of all CP-relays on shutdown string B	A	0.30	RREC01QU	RRECB1QP, RRECB2QP, RRECB3QP
4. Contacts of all CP-relays on shutdown string B	A	0.30	RCZC02QU	RCZCB1QP, RCZCB2QP, RCZCB3QP
5. Armatures/coils of all CP-relays on both shutdown strings	B	0.008	RREC00QU	RRECPAQP, etc RRECB1QP, etc
6. Contacts of all CP-relays on both shutdown strings	B	0.008	RCZC00QU	RCZCPAQP, etc RCZCB1QP, etc

Table 7.38

LIST OF COMMON CAUSE FAILURES AMONG ELECTRICAL COMPONENTS -
FAULT TREE ANALYSIS OF SCRAM FAILURE UNDER LOF OR
TOP CONDITION (Cont'd)

Description of CCF	Analysis Model	β Factor	CCF Event	Corresponding Independent Events
<u>LOF FAULT TREE</u>				
<u>(i) Low-Flow Channels</u>				
1. Armatures/coils of two trip relays for both shutdown strings - each channel separate	A	0.20	RRE1A0QU	RRE1A1QP, RRE1A2QP
			RRE1B0QU	RRE1B1QP, RRE1B2QP
			RRE1C0QU	RRE1C1QP, RRE1C2QP
			RRE1D0QU	RRE1D1QP, RRE1D2QP
2. Contacts of two trip relays for both shutdown strings - each channel separate	A	0.20	RCZ1A0QU	RCZ1A1QP, RCZ1A2QP
			RCZ1B0QU	RCZ1B1QP, RCZ1B2QP
			RCZ1C0QU	RCZ1C1QP, RCZ1C2QP
			RCZ1D0QU	RCZ1D1QP, RCZ1D2QP
3. Armatures/coils of all trip relays in all channels	B	0.008	RRE1Z0QU	RRE1A1QP, etc RRE1B1QP, etc RRE1C1QP, etc RRE1D1QP, etc
4. Contacts of all trip relays in all channels	B	0.008	RCZ1Z0QU	RCZ1A1QP, etc RCZ1B1QP, etc RCZ1C1QP, etc RCZ1D1QP, etc

Table 7.38

**LIST OF COMMON CAUSE FAILURES AMONG ELECTRICAL COMPONENTS -
FAULT TREE ANALYSIS OF SCRAM FAILURE UNDER LOF OR
TOP CONDITION (Cont'd)**

Description of CCF	Analysis Model	β Factor	CCF Event	Corresponding Independent Events
5. Bistables for both shutdown strings - each channel separate	A	1.0	RBI1A0QW RBI1B0QW RBI1C0QW RBI1D0QW	N/A N/A N/A N/A
6. All bistables (Ronan X-82) in all channels(2,), (4)	B	0.10	RBI1Z0QU	RBI1A0QW, RBI1B0QW, RBI1C0QW
7. All MV/I converters (Ronan X-81) in all channels(4)	B	0.02	RVI1Z0QU	RVI1A1QW, RVI1A2QW, RVI1B1QW, RVI1C1QW
8. Electro-magnetic flow sensor of channels HPPF-2 and LPPF-2	B	0.10	RFM1Z1RU	RFM1A0RW, RFM1B0RW
(ii) SOT Channels				
1. Armatures/coils of two trip relays for both shutdown strings - each channel separate(5)	A	0.20	RRE3C0QU RRE3D0QU RRE3F0QU RRE3H0QU	RRE3C1QP, RRE3C2QP RRE3D1QP, RRE3D2QP RRE3F1QP, RRE3F2QP RRE3H1QP, RRE3H2QP

Table 7.38

LIST OF COMMON CAUSE FAILURES AMONG ELECTRICAL COMPONENTS -
 FAULT TREE ANALYSIS OF SCRAM FAILURE UNDER LOF OR
 TOP CONDITION (Cont'd)

Description of CCF	Analysis Model	β Factor	CCF Event	Corresponding Independent Events
2. Contacts of two trip relays for both shutdown strings - each channel separate(5)	A	0.20	RCZ3C0QU RCZ3D0QU RCZ3F0QU RCZ3H0QU	RCZ3C1QP, RCZ3C2QP RCZ3D1QP, RCZ3D2QP RCZ3F1QP, RCZ3F2QP RCZ3H1QP, RCZ3H2QP
3. Armatures/coils of all trip relays in all channels	B	0.008	RRE1Z0QU	RRE3C1QP, etc RRE3D1QP, etc RRE3F1QP, etc RRE3H1QP, etc
4. Contacts of all trip relays in all channels	B	0.008	RCZ1Z0QU	RCZ3C1QP, etc RCZ3D1QP, etc RCZ3F1QP, etc RCZ3H1QP, etc
5. All bistables (Ronan X-82) in all channels	B	0.008	RBI1Z0QU	RBI3C0QW, RBI3D0QW, RBI3F0QW, RBI3H0QW
6. All MV/I converters (Ronan X-81) in all channels	B	0.008	RVI1Z0QU	RVI3C0QW, RVI3D0QW, RVI3F0QW, RVI3H0QW
7. Thermocouples in all channels(5)	B	0.02	RTC300RU	RTC3C0RW, RTC3D0RW, RTC3F0RW, RTC3H0RW

Table 7.38

LIST OF COMMON CAUSE FAILURES AMONG ELECTRICAL COMPONENTS -
 FAULT TREE ANALYSIS OF SCRAM FAILURE UNDER LOF OR
 TOP CONDITION (Cont'd)

Description of CCF	Analysis Model	β Factor	CCF Event	Corresponding Independent Events
8. Reference junctions in all channels(5)	B	0.02	RRT300AU	RRT3C0AW, RRT3D0AW, RRT3F0AW, RRT3H0AW
<u>TOP FAULT TREE</u>				
<u>(i) High-Flux Channels</u>				
1. Armatures/coils of two trip relays for both shutdown strings - each channel separate	A	0.20	RRE2A0QU	RRE2A1QP, RRE2A2QP
			RRE2B0QU	RRE2B1QP, RRE2B2QP
			RRE2C0QU	RRE2C1QP, RRE2C2QP
2. Contacts of two trip relays for both shutdown strings - each channel separate	A	0.20	RCZ2A0QU	RCZ2A1QP, RCZ2A2QP
			RCZ2B0QU	RCZ2B1QP, RCZ2B2QP
			RCZ2C0QU	RCZ2C1QP, RCZ2C2QP
3. Armatures/coils of all trip relays in all channels	B	0.008	RRE2Z0QU	RRE2A1QP, etc RRE2B1QP, etc RRE2C1QP, etc
4. Contacts of all trip relays in all channels	B	0.008	RCZ2Z0QU	RCZ2A1QP, etc RCZ2B1QP, etc RCZ2C1QP, etc

Table 7.38

LIST OF COMMON CAUSE FAILURES AMONG ELECTRICAL COMPONENTS -
 FAULT TREE ANALYSIS OF SCRAM FAILURE UNDER LOF OR
 TOP CONDITION (Cont'd)

Description of CCF	Analysis Model	β Factor	CCF Event	Corresponding Independent Events
5. Neutron detectors(3) of all channels	A	0.0	None	RND2A0RW, RND2B0RW, RND2C0RW
6. Neutron detectors(3) of all channels	B	0.10	RND2Z0RU	RND2A0RW RND2B0RW RND2C0RW
7. HF- & LV-bistables for both shutdown strings - each channel separate	A	0.20	RBI2A0QU	RBI2A1QW, RBI2A2QW, RBI2A3QW, RBI2A4QW
			RBI2B0QU	RBI2B1QW, etc
			RBI2C0QU	RBI2C1QW, etc
8. All HF-bistables in all channels	B	0.008	RBI2Z1QU	RBI2A1QW, RBI2A2QW, RBI2B1QW, RBI2B2QW, RBI2C1QW, RBI2C2QW
9. All LV-bistables in all channels	B	0.008	RBI2Z2QU	RBI2A3QW, RBI2A4QW, RBI2B3QW, RBI2B4QW, RBI2C3QW, RBI2C4QW
10. Armatures/coils of K3, K4, K5, and K6 relays in each channel	A	0.20	RRE2A9QU	RRE2A3QP, RRE2A4QP, RRE2A5QP, RRE2A6QP
			RRE2B9QU	RRE2B3QP, etc
			RRE2C9QU	RRE2C3QP, etc

Table 7.38

LIST OF COMMON CAUSE FAILURES AMONG ELECTRICAL COMPONENTS -
 FAULT TREE ANALYSIS OF SCRAM FAILURE UNDER LOF OR
 TOP CONDITION (Cont'd)

Description of CCF	Analysis Model	β Factor	CCF Event	Corresponding Independent Events
11. Contacts of K3, K4, K5, and K6 relays in each channel	A	0.20	RCZ2A9QU RCZ2B9QU RCZ2C9QU	RCZ2A3QP, RCZ2A4QP, RCZ2A5QP, RCZ2A6QP RCZ2B3QP, etc RCZ2C3QP, etc
12. Armatures/coils of K3 and K4 relays (HF) in all channels	B	0.008	RRE2Z9QU	RRE2A3QP, RRE2A4QP, RRE2B3QP, RRE2B4QP, RRE2C3QP, RRE2C4QP
13. Armatures/coils of K5 and K6 relays (LV) in all channels	B	0.008	RRE2Z8QU	RRE2A5QP, RRE2A6QP, RRE2B5QP, RRE2B6QP, RRE2C5QP, RRE2C6QP
14. Contacts of K3 and K4 relays (HF) in all channels	B	0.008	RCZ2Z9QU	RCZ2A3QP, RCZ2A4QP, RCZ2B3QP, RCZ2B4QP, RCZ2C3QP, RCZ2C4QP
15. Contacts of K5 and K6 relays (LV) in all channels	B	0.008	RCZ2Z8QU	RCZ2A5QP, RCZ2A6QP, RCZ2B5QP, RCZ2B6QP, RCZ2C5QP, RCZ2C6QP

Table 7.38

LIST OF COMMON CAUSE FAILURES AMONG ELECTRICAL COMPONENTS -
FAULT TREE ANALYSIS OF SCRAM FAILURE UNDER LOF OR
TOP CONDITION (Cont'd)

Description of CCF	Analysis Model	β Factor	CCF Event	Corresponding Independent Events
<u>(ii) SOT Channels</u>				
1. Armatures/coils of all trip relays in all channels	B	0.008	RRE2Z0QU	RRE3C1QP, etc RRE3D1QP, etc RRE3F1QP, etc RRE3H1QP, etc
2. Contacts of all trip relays in all channels	B	0.008	RCZ2Z0QU	RCZ3C1QP, etc RCZ3D1QP, etc RCZ3F1QP, etc RCZ3H1QP, etc
3. All bistables (Ronan X-82) in all channels	B	0.02	RBI3Z0QU	RBI3C1QW, RBI3D1QW, RBI3F1QW, RBI3H1QW
4. All MV/I Converters (Ronan X-81) in all channels	B	0.02	RVI3Z0QU	RVI3C0QW, RVI3D0QW, RVI3F0QW, RVI3H0QW

Notes:

- (1) In Analysis Model A, relatively large β -Factors are used to emphasize CCFs between similar components on the same Shutdown STring or same Channel. In Analysis Model B, β -Factors are based only on the number of redundant components.
- (2) The bistables of channel UPFB are made by a different manufacturer.
- (3) Similar β -Factors are used with:
 - Cables and Connectors
 - Current amplifiers
 - + 15 V dc power supplies
 - - 680 V dc power supplies
- (4) The β -Factors are reduced to 0.008 when the SOT channels are part of LOF detection.
- (5) Similar β -Factors are used with the SOT channels in the TOP Fault Tree.

Table 7.39

BASIC EVENTS USED IN REPRESENTING MECHANICAL FAILURE OF
CONTROL AND SAFETY RODS

Event Name	Failure per Demand	Description and Causes of Binding
<u>CRs & CRDs</u>		
1 CCD30IRU	3.10^{-7}	CR Drives; Foreign material or particulates in Na exiting assemblies
2 CCD20WRV	1.10^{-7}	CR Drives; Failure of components located above rotating plug level
3 CCD10GRU	$1.3 \cdot 10^{-6}$	CR Drives; Failure of components located inside tank, mainly reactor vessel cover locking mechanism
4 CSR30IRU	1.10^{-7}	<u>CCF among CRs & SRs</u> ; Foreign materials or particulates in Na entering assemblies
5 CSR10GRU	1.10^{-7}	<u>CCF among CRs & SRs</u> ; Rods and Thimbles: Cross section distortion and bowing
6 CCL00PRU	$1.6 \cdot 10^{-7}$	Magnetic Clutches Fail to Release CRs
<u>SRs & SRDs</u>		
7 SCD300RI	6.10^{-5}	SR Drive; Foreign materials or particulates in Na
8 SCD200RW	6.10^{-5}	SR Drive; Failure of components located above rotating plug level
9 SCD100RG	6.10^{-5}	SR Drive; Failure of components located inside vessel
10 SSR300RI	6.10^{-5}	SRs and Thimbles; Foreign materials or particulate in Na existing assemblies
11 SSR100RG	6.10^{-5}	SRs and Thimbles; Cross section distortion, bowing
12 SSR100RG	2.10^{-4}	Magnetic Clutch Fails to Release SRs

demand, reflecting the experimental role of EBR-II operation. [This failure probability of a single control rod agrees with the probability that one would obtain based on the number of previous EBR-II scrams.] With the generic beta factor for a nine-rod redundancy, $\beta(9)=0.008$, the total failure probability becomes about $7 \cdot 10^{-7}$ per demand. As for the two safety rods, since they are laterally connected and are inserted in the core with the same drive mechanism, the failure rate of the two rods combined is conservatively estimated at $3 \cdot 10^{-4}$, which is 10 times larger than the generic value. Table 7.37 breaks down the common cause events of both the control rods and safety rods to their respective components. The purpose of this expansion, which is based on engineering judgement and familiarity with the system, is to arrive at a potential CCF between the two different types of rods. Among the five failure modes of each rod type, two have been identified as potential CCFs between the two types of rods with a combined failure probability of $2 \cdot 10^{-7}$ per demand. Obviously this CCF among the two types of rods sets a lower limit on the RSS availability. types of rods. Among the five failure modes of each rod type, two have been identified as potential CCFs between the two types of rods with a combined failure probability of $2 \cdot 10^{-7}$ per demand. Obviously this CCF among the two types of rods sets a lower limit on the RSS availability.

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8.0 HUMAN RELIABILITY ANALYSIS

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8.0 HUMAN RELIABILITY ANALYSIS

Because of the passive safety features of the EBR-II design, the contributions of human error to core damage sequences are those associated with failure to scram and failure to recover decay heat removal. The events which can effect scram reliability can be promulgated by: 1) errors in calibration and setting of trip points and 2) failure to effect manual scram. The human failure sequences lending to these errors have been examined in detail; the resulting relatively high human reliability is primarily due to reactor design, with strong contributions from EBR-II operator training, operating and maintenance procedures, and a high degree of professionalism demonstrated by EBR-II personnel. There are a few events which influence the decay heat removal capability. These events are predominantly failure to recover events because of the passive nature of the decay heat removal function.

8.1 General Evaluation of EBR-II Human Factors

8.1.1 Training

Operator training at EBR-II is extensive in terms of system and reactor knowledge presented. For example, current training emphasizes a thorough knowledge of functional design and characteristics of the Reactor Shutdown System (RSS). This knowledge is tested in both new and requalification training by random selection of test items that require operators to state consequences of failure of components and cascade effects of those failures, as well as expected indications associated with those failures. The operators of EBR-II have an extensive knowledge of the physics, operation, and design of the plant.

8.1.2 Procedures

Currently, the format and layout of operating and maintenance procedures are in line with updated MIL Standards [8.1] and NUREG-0899 [8.2]. The procedures are

excellent in terms of their requirements for check-off and sign-off actions. Observation indicates that these sign-off provisions are usable and are used. Extensive cross-checking of system operability required during start-up and approach-to-power sequences add to the safety of both these sequences and full-power operation. The net result of these procedural requirements is to make reactor startup of the EBR-II a cautious and deliberate enhancement to safety.

8.1.3 Maintenance

The EBR-II facilities are well lighted and spacious in general, especially when compared with most LWR facilities. Most areas in which maintenance personnel work have ample space and generally good environmental conditions, including low radiation levels. It is worth noting that most calibration and setting of trip-points for safety significant instrumentation occur in the cable-routing room. This room is limited access, with notification of the control room required. It is spacious and well lighted. The instruments, relays, etc. are located within locked cabinets in this room. The instruments themselves are clearly labeled and color coded. Instruments themselves are easily accessible with ample room for maintenance or replacement activities.

8.1.4 Maintenance History

A review of Unusual Occurrence Reports (UORs) for the last ten years was performed. The overall rate of human error-related UORs is low, on the order of 4 per year. There has been only an average of about one scram per year that was attributable to maintenance/equipment failures related to maintenance personnel error. The philosophy of performing most maintenance in shut-down conditions reduces the amount of lost power and challenges to the safety systems due to maintenance errors. The low number of maintenance errors that contribute to either UORs or scrams supports the observation that maintenance procedures are adequate and contribute minimal risk to the operation of this reactor.

8.2 Human Reliability Assessment

The investigation of human errors that could initiate core damage sequences at EBR-II was performed by means of personal interviews, reviews of relevant operating and maintenance procedures, reviews of the control console, review of internal documentation related to the PRA effort, and reviews of EBR-II Technical Specifications. Interviews were held with personnel from Training and Procedures, Critical Systems Maintenance, Instrumentation and Control Computer Applications and Program Development, and Operations Analysis. The analyses were performed by following the procedures outlined in NUREG/CR-1278, the "THERP Handbook". [8.3]. In particular, a human error probability of 10^{-6} was selected as a lower limit, as suggested in THERP.

8.2.1 Initiating Events

Three major classes of transient events related to human error were isolated as being most likely to lead to core vulnerability or plant unavailability. These events are:

1. Loss of Flow (LOF)
2. Loss of Heat Sink (LOHS)
3. Transient Over Power (TOP).

The first human reliability analysis that was performed was the evaluation of the contribution of human error to the initiation of these events. Operator actions are one of the primary contributors to risk in most reactor PRAs. The analysis was performed by review of operating procedures and discussions with operators and shift supervisors. The conclusion is that human error does not contribute to initiating sequences that are included in the cutsets.

There are several reasons for the lack of human contribution to initiation of these sequences. Most of the reasons are a result of the design of the reactor, while the remaining reasons derive from the operating philosophy and practices as embodied in the procedures. For example human error contribution to initiation of LOF or TOP events is largely precluded by the reactor design. Operators and maintainers cannot inadvertently trip the main pumps because switches and controls that shutoff these pumps are electrically locked out of operation when RSS is activated at reactor startup. Although operators can (and have) inadvertently added reactivity at full power, only one control rod can be inserted at one time, due to electrical interlocks.

The EBR-II reactor startup is a slow and deliberate procedure that is very safe. Startup is preceded by a full interlock check of all safety and operating systems. The startup procedure begins slowly at minimal (50 W) power and has hold points at every 10 MW stage for further checklist and calibration checks until reaching authorized power (typically 62.5 MW). These repeated hold-and-check points makes startup a lengthy process, but reduces risk to a very low value. These repeated checks also reduce the probability of undetected and uncorrected human errors to a very low value.

8.2.2 Human Error Contribution to Scram Failures

Human error during maintenance activities can contribute to failure of actuation of the automatic reactor shutdown system (scram system). Once the automatic system has failed, human operator actions may effect recovery actions (primarily manual scram).

There are two categories in which maintenance activities can contribute to failure of the scram system: 1) Instrumentation, and 2) Mechanical. Maintenance activities on instrumentation can cause or contribute to instrumentation failure through miscalibration and/or misadjustment of trip setpoints. Maintenance activities can cause or contribute to mechanical failure through faulty mechanical latching of control rod latch (release) rollers. These two categories will be treated separately below.

8.2.2.1 Miscalibration and Misadjustment of Safety-Significant Instrumentation

The primary instrumentation to initiate a scram following an LOF event are the Flow Sensors, which are backed up by the Sub-Assembly Outlet Temperature (SOT) instruments. That is, if flow is reduced or lost, sub-assembly outlet temperatures would immediately begin to rise. Thus, if flow indication was inaccurate or disabled, sub-assembly outlet temperatures would give indications of low or lost flow and initiate a scram. The primary instrumentation to initiate scram in case of a TOP event is the Wide Range Nuclear Channels (A, B, and C) - again, backed up by the SOT instruments. Increases in power will also increase temperatures within the core, leading to increased SOT indications. All calibration and adjustment procedures are performed in the cable routing room, a limited access (keycard) room, where telephone notification to the control room of entry and purpose is required.

8.2.2.2 Flow Sensors

Failure to Properly Calibrate Sensors, $p = \sim 0$

The flow sensors in the scram strings are the Low Pressure Plenum Flow for pump #2, High Pressure Plenum Flow for pump #2, the Total Reactor Outlet Flow at the outlet of the reactor coolant pipe, and the Upper Plenum Flow at the reactor upper plenum. The calibration procedures for these instruments were examined, but detailed analysis was not performed. The probability of undetected miscalibration was judged to be nearly zero for the following reasons:

1. The flow sensor instruments are set to relative readings (100%). Miscalibration would result in a power mismatch for the flow ratings at 100%, which would cause a halt in the approach to power and re-examination of several instrument calibrations.

2. The flow is checked during start up and again during approach to power and, finally, after reaching full power operation.
3. Flow readings are taken every four hours during normal operation.

Event RHU101QL, Failure to Properly Perform Flow Sensor Operability Check: $p = 1.5 \cdot 10^{-3}$ (Ref. Human Event Tree TFSO, Figure 8.1, Tables 8.1 and 8.2)

Sensor operability and trip point verification procedures were reviewed for these flow sensor instruments and a detailed Human Reliability Analysis (HRA) was judged to be warranted.

The operability check is performed at startup, and weekly while operating at authorized power. To perform this procedure, the technician requests that the operators increase flow by 2%. The technician observes and records the change in flow readings on all four channels. The probability of error in this check on one channel is $1.5 \cdot 10^{-3}$. This value is deemed appropriate because the operability test is repeated weekly during normal operations, reducing the probability of unnoticed loss of operability. This human error essentially increases the test interval of the checked components from one week to two weeks. As a result, the net effect on the failure probability of flow channels is insignificant.

HRA TREE DIAGRAM

ID: TFSO I & C Tech tests flow sensor operability. (STM-T-35(7)10/89)
 Sequence: RHU101QL
 Date modified: 26-MAR-90
 Analyst: Schuurman
 NOTES: This is a tech. spec. start up item. Performed in cable routing room which has access controlled by operators. It is a two-man job with one man acting as a checker on each step and signing off on the completion.

Fig. 8.1. Human Event Tree, TFSO

Table 8.1. HEPs for Human Event Tree TFSO

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A I&C Tech. fails to record existing flow readings.	0.004	Table 20-10, (9)	SBS	1		ZID	0.004 (3)
B Checker fails to check recording of existing flow readings.	0.1	Table 20-22, (1)	SBS	1		ZID	0.1 (5)
C I&C Tech. fails to call control room to request increased flow.	0.003	Table 20-7, (3)	SBS	1		ZID	0.003 (3)
D Checker fails to correct failure to call control room	0.1	Table 20-22 (1)	DYN	1		ZID	0.1 (5)
E I&C Tech. and checker fail to note flow increase.	0.003	Table 20-7 (3)	DYN	1		ZID	0.003 (3)
F I&C Tech. fails to record readings under increased flow condition.	0.004	Table 20-10 (9)	SBS	1		ZID	0.004 (3)
G Checker fails to note failure to record reading	0.1	Table 20-22 (1)	SBS	1		ZID	0.1 (5)
H I&C Tech. fails to note inconsistent flow reading changes	0.003	Table 20-10 (1)	SBS	1		ZID	0.003 (3)
I Checker fails to check evaluation of sensor response.	0.1	Table 20-22 (1)	SBS	1		ZID	0.1 (5)
J I&C Tech. fails to call CRO to reduce flow to original level.	0.003	Table 20-7 (3)	SBS	1		ZID	0.003 (5)
K Checker fails to note that CRO was not called.	0.1	Table 20-22 (1)	DYN	1		ZID	0.1 (5)
L I&C Tech. fails to record new flow readings.	0.001	Table 20-7 (1)	SBS	1		ZID	0.001 (3)
M Checker fails to note that readings are recorded.	0.1	Table 20-22 (1)	DYN	1		ZID	0.1 (5)

Table 8.2. Failure Paths and Total Failure/Success Probabilities for Human Event Tree TF50

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.004 x 0.1	0.0004		
2 .aCDE	0.003 x 0.1 x 0.003	0.		
3 .acFG	0.004 x 0.1	0.0004		
4 .acFH	0.003 x 0.1	0.0003		
5 .acHJK	0.003 x 0.1	0.0003		
6 .acHJLM	0.001 x 0.1	0.0001		
			0.0015	0.998499

Events RHU1A1OL, RHU1B1OL, RHU1C1OL and RHU1D1OL, Failure to Properly Verify Flow Sensor Trip Point Operability: $p = 3 \cdot 10^{-6}$ (Ref. Human Event Trees LPPF, HPPF, TSO, UPFB, Figures 8.2 to 8.5, Tables 8.3 to 8.10)

The flow sensor trip points are calibrated annually and are also verified during startup and weekly during authorized power. These verifications are performed by switching each instrument to CALIBRATE position and back to NORMAL, sequentially. Verification is performed by means of built-in test equipment. When any channel is in CALIBRATE position, it is tripped, and a TRIP signal is displayed in the control room - thus, the system is working on a 1 of 3 trip logic during this period. If the technician places a channel in CALIBRATE without returning the previous channel to NORMAL, the plant will trip on a low flow condition. Prior to startup, the trip points are operationally checked by lowering flow to 85% and verifying that the sensors all trip. Verification is made by means of a built-in test circuit on each channel when the reactor is critical. The probability of failure to verify trip points for each channel is $3 \cdot 10^{-6}$. These failures are judged to be independent, since each instrument is in a separate locked cabinet and one channel at a time is adjusted, then the cabinet relocked and the next cabinet unlocked. Credit was not taken for the fact that there is a checker assigned to observe the technician during these adjustments. Similar to the above, these human errors would essentially increase the test interval of the checked components from one week to two weeks. As a result, their net effect on the failure probability of their respective channels is insignificant. In the present LOF fault tree, these human errors are conservatively modeled as events that lead to channel failure but at a reduced probability of $1 \cdot 10^{-6}$.

HRA TREE DIAGRAM

ID: *HPPF Tech. fails to accurately set pump #2 HP plenum trip-point*
 Sequence: *RHULA1QL*
 Date modified: *18-JUN-90*
 Analyst: *Schurman*
 NOTES: *I & C Tech. fails to set the pump #2 HP plenum trip point at proper value. (STM-T-35, 10/31/89)*

Fig. 8.2. Human Event Tree, HPPF

Table 8.3. HEPs for Human Event Tree HPPF

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A Tech. fails to accurately set trip point during Annual Calibration.	0.003 (3)	Table 20-7(2)	SBS	1		ZID	0.003 (3)
B Pre-Startup Operational Interlock check fails to discover misadjustment.	0.001 (10)	Analyst's judgement	DYN	1		ZID	0.001 (10)

Table 8.4. Failure Paths and Total Failure/Success Probabilities for Human Event Tree HPPF

Tech. fails to accurately set pump #2 HP plenum trip-point			
Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability
1 AB	0.003×0.001	0.000003	
			0.999997

HRA TREE DIAGRAM

ID: *LPPF Tech. fails to accurately set pump #2 LP plenum trip-point*
Sequence: *RHU1BIQL*
Date modified: *10-APR-90*
Analyst: *Schurman*
NOTES: *I & C Tech. fails to set the pump #2 LP plenum trip point at proper value. (STM-T-35(7) 10/31/89)*

Fig. 8.3 Human Event Tree, LPPF

Table 8.5. HEPs for Human Event Tree LPPF

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A I&C Tech. fails to accurately set trip for Low Pressure Plenum on Pump #2.	0.003	Table 20-7(2)	SBS	1		ZD	0.003 (3)
B Pre-Startup Operability Interlock check fails to discover misadjustment.	0.001	Analyst's judgement SBS		1		ZD	0.001 (10)

Table 8.6. Failure Paths and Total Failure/Success Probabilities for Human Event Tree LPPF

Tech. fails to accurately set pump #2 LP plenum trip-point

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 AB	0.003×0.001	0.000003		
			0.000003	0.999997

HRA TREE DIAGRAM

ID: *TSOF Tech. fails to accurately set total flow trip-point*
Sequence: *RHU1C1QL*
Date modified: *10-APR-90*
Analyst: *Schurman*
NOTES: *I & C Tech. fails to set the trip point for the total-flow sensor.*

(STM-T-35(7) 10/31/89)

Fig. 8.4. Human Event Tree, TSOF

Table 8.7. HEPs for Human Event Tree TSOF

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A I&C Tech. fails to accurately adjust trips for Total Flow Rate.	0.003	Table 20-7(2)	SBS	1		ZD	0.003 (3)
B Pre-Startup Operability Interlock check fails to discover misadjustment.	0.001	Analyst's judgement SBS		1		ZD	0.001 (10)

Table 8.8. Failure Paths and Total Failure/Success Probabilities for Human Event Tree TSOF

Failure Path	Calculations	Tech. fails to accurately set total flow trip-point		
		Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 AB	0.003×0.001	0.000003		
			<u>0.000003</u>	<u>0.999997</u>

HRA TREE DIAGRAM

ID: *UPFB Tech. fails to accurately set upper plenum trip-point*
Sequence: *RHU1D1QL*
Date modified: *10-APR-90*
Analyst: *Schurman*
NOTES: *I & C Tech. fails to set the trip point for the upper plenum sensor. (STM-T-35(7) 10/31/89)*

Fig. 8.5. Human Event Tree, UPFB

Table 8.9. HEPs for Human Event Tree UPFB

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A I&C Tech. fails to accurately adjust trip for Upper Plenum flow.	0.003 (3)	Table 20-7(2)	SBS	1		ZD	0.003 (3)
B Pre-Startup Operability Interlock check fails to discover misadjustment.	0.001 (10)	Analyst's judgement	SBS	1		ZD	0.001 (10)

Table 8.10. Failure Paths and Total Failure/Success Probabilities for Human Event Tree UPFB

Failure Path	Tech. fails to accurately set upper plenum trip-point		
	Calculations	Resulting Failure Probability	Total Failure Probability
1 .AB	0.003 x 0.001	0.000003	
			0.000003
			0.999997

8.2.2.3 Sub-assembly Outlet Temperature Sensor Calibrations

The calibration and adjustment procedures were reviewed for the Sub-assembly Outlet Temperature (SOT) instruments. These procedures were found to be complex and required the calibration of several subunits for successful completion; furthermore no operability check is possible. Therefore detailed HRA of the calibration procedures was judged to be necessary.

Event RHU3AQL Calibration, Time Response Test of Core Subassembly Thermocouples: $p = 6 \cdot 10^{-4}$ (Ref. Human Event Tree SOTC in Figure 8.6, Tables 8.11 and 8.12)

The calibration procedure is performed annually. The procedure requires the adjustment of a number of amplifiers and detectors in series. Subsequent to calibration, timed tests of trip function are performed. The estimated probability of failing to perform the calibration accurately, and that subsequent check steps would not catch the error, is $6 \cdot 10^{-4}$. Since these calibration steps are performed separately on the four channels, three of four must be miscalibrated for consequences to occur, and since each channel is located in a separate, locked cabinet - the four channels are assumed to be independently calibrated.

Event RHU302QL, Set Sub-assembly Outlet Temperature Sensor Trip-points: $p = 2.5 \cdot 10^{-2}$ (Ref. Human Event Tree SOTT in Figure 8.7, Tables 8.13 and 8.14)

The setting of the trip points for the SOT channels consists of making a setting of each of the Ronan High Temperature Alarms to a value calculated by the control room operators and checked against estimates in the operating plan. It is performed at start up, at 50% power, and again when authorized power is reached. This activity is covered by a one-sentence procedure in the Startup Procedure (II-B). It is unlikely that the technician actually reads that procedure for each setting. There is often a second technician present who acts as a checker, but the checker is not procedurally required. The settings are made quickly and in order, so they cannot

HRA TREE DIAGRAM

ID: SOTC Calibrate SOT Channels (one at a time)
 Sequence: RHU3A1QL, RHU3B1QL, RHU3C1QL, RHU3D1QL
 Date modified: 18-JUN-90
 Analyst: Schumann
 NOTES: Calibration of Sub-Assembly Outlet Temperature Channels (4 ea.) one at a time. (STM-MC-61 Rev. 7, 2/21/89)

Fig. 8.6. Human Event Tree, SOTC

Table 8.11. HEPs for Human Event Tree SOTC

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source 'THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A Tech fails to properly set up test equipment.	0.01 (5)	Table 20-7(5)	SBS	1		ZD	0.01 (5)
B Tech fails to properly set test switches (2).	0.0011 (10)	Table 20-12(5) + (9)	SBS	1		ZD	0.0011 (10)
C Tech fails to properly perform "as found" check on system (R1-TC-503C).	0.001 (3)	Table 20-7(1)	SBS	1		ZD	0.001 (3)
D Tech fails to properly record "as found" readings. (Step 8.1.3)	0.004 (3)	Table 20-10(9)	SBS	1		ZD	0.004 (3)
E Administrative check fails to catch errors.	0.5 (5)	Table 20-22(8)	SBS	1		ZD	0.5 (5)
F Tech fails to properly check calibration on Newport DVM.	0.001 (3)	Table 20-7(1)	SBS	1		ZD	0.001 (3)
G Tech fails to properly record "as found" readings	0.004 (3)	Table 20-10(9)	SBS	1		HD	0.502 (3)
H Administrative check fails to catch errors.	0.5 (5)	Table 20-22(8)	SBS	1		ZD	0.5 (5)
I Tech fails to properly zero API meter (short, clear two-step procedure).	0.0005 (3)	Table 20-7(1)	SBS	1		ZD	0.0005 (3)
J Tech fails to properly record "as found" readings on API meter.	0.0004 (3)	Table 20-7(3)	SBS	1		HD	0.5002 (3)
K Administrative check fails to catch errors.	0.5 (5)	Table 20-22(8)	SBS	1		ZD	0.5 (5)
L Tech does not properly check calibration of Ronan High-Temp alarm.	0.001 (3)	Table 20-7(1)	SBS	1		ZD	0.001 (3)

Table 8.11. HEPs for Human Event Tree SOTC (cont'd)

M Tech fails to properly record "as found" readings on Ronan High-Temp alarm.	0.0004 (3)	Table 20-7(3)	SBS	1	IID	0.5002 (3)
N Check of recorded values fails to catch error.	0.5 (5)	Table 20-22(8)	SBS	1	ZD	0.5 (5)

Table 8.12. Failure Paths and Total Failure/Success Probabilities for Human Event Tree SOTC

Calibrate SOT Channels (one at a time)

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .aFGH	$0.001 \times 0.502 \times 0.5$			
2 .aΠJK	$0.0005 \times 0.5002 \times 0.5$	0.000251		
3 .afiLMN	$0.001 \times 0.5002 \times 0.5$	0.000125		
		0.0002501		
			0.0006261	0.9993738

HRA TREE DIAGRAM

ID: *SOTT Adjust trips for Subassembly Outlet Temperature (SOT)*
Sequence: *RHU302QL*
Date modified: *18-JUN-90*
Analyst: *Schurman*
NOTES: *Tech. sets trip points for Subassembly Outlet Temperature Instruments R1-TC-503 D, F, H, and C at temperatures calculated by operators.*

Fig. 8.7. Human Event Tree, SOTT

Table 8.13. HEPs for Human Event Tree SOTT

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A Tech. adjusts trips 'High' on SOT Instruments R1-TC-503 C, D, F, and H.	0.025 (5)	.5 x Table 20-7(5)	SBS	1	directional	ZD	0.025 (5)

Table 8.14. Failure Paths and Total Failure/Success Probabilities for Human Event Tree SOTT

Adjust trips for Subassembly Outlet Temperature (SOT)				
Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .A	0.025	0.025	0.025	0.975

be presumed to be independent. The probability of directional error in adjusting each channel is assumed to be 2.5×10^{-2} , which is one-half of the value given in NUREG/CR-1278, Table 20-7, item (5), [8.3]. One-half of the table value is used because misadjustment can occur in one direction only for an unsafe trip setting to result.

Since the SOT systems are designed to be a backup signal to primary trip systems, the procedures reflect this philosophy. However, if it is desired that this probability be reduced, procedurally requiring the checker to be present and making this adjustment a sign-off step would lower the estimated error probability by approximately two orders of magnitude.

8.2.2.4 Wide Range Nuclear Channel Trip Points

The linear drawer sections of each of the three wide-range nuclear channels (A, B, and C) provide flux measurement for the system. Since these channels are set to relative values (100%), are checked and reset several times during startup, and readings are checked and recorded every four hours during normal operation, it was determined that it would not be necessary to perform a detailed analysis on the calibration procedures.

The setting of trip points on these channels require that the scram trip points be set for "overpower" (high flux greater than 110%) and for loss of high voltage. An alarm trip point is also set for flux exceeding 104% of full power. This alarm trip point is important in providing additional time for the operators to respond to power increases (TOP events) but does not otherwise contribute to control of the event, so it was not included in the HRA event tree.

Event RHU201QL, Failure to Accurately Set High Flux Trip Point at 110% of Authorized Power: $p = 1.5 \times 10^{-6}$ (Ref. Human Event Tree BSPL, Figure 8.8, Tables 8.15 and 8.16)

HRA TREE DIAGRAM

ID: BSPL Tech. sets level trip points for linear drawers A,B, & C.
Sequence: RHU201QL
Date modified: 18-JUN-90
Analyst: Schurman
NOTES: Technician sets high level bistable trip points for linear amplifiers of channel A, B, and C. These settings are checked by the Interlock checks before start up of the reactor. (Proc. STM-MC-77 Rev. 8)

Fig. 8.8. Human Event Tree, BSPL

Table 8.15. HEPs for Human Event Tree BSPL

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A Tech. fails to set high level trip level to 110% +0% or -2%	0.0015 (3)	.5 x Table 20-7 (3)	SBS	1	directional	ZD	0.0015 (3)
B Interlock check fails to find misadjustment of high level setpoint.	0.001 (10)	Analyst's judgement	SBS	1		ZD	0.001 (10)

Table 8.16. Failure Paths and Total Failure/Success Probabilities for Human Event Tree BSPL

Tech. sets level trip points for linear drawers A,B, & C.

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.0015×0.001	0.000001		
			0.000001	0.999998

This procedure requires that the bistable trip on each channel (A,B & C) be set, by means of gain controls, to trip at 90% of authorized power during startup until power reaches 30MW. At 30 MW the gain controls are reset to trip at 110% of authorized power (calculated). The gain control accuracy is verified one hour after full power has been attained. The accuracy of the trip point settings at 110%, $\pm 0\%/-2\%$, is verified by built in tests prior to each numbered run. The probability that the bistable trips will be set inaccurately in either direction is 3×10^{-6} . This is a conservative estimate, since there is a 50:50 chance of setting the trip point too low (producing a scram or alarm before it is needed) rather than too high (failing to produce a scram or alarm when it is needed).

The probability of misadjusting scram trip points for the channels is conservatively assumed to be non-independent. Although the drawers are in separate locked cabinets, the cabinets are transparent and the trip points are set one after the other.

Event RHU202QL, Failure to Accurately Set High Voltage-Low Trip Point: $p = 1.5 \times 10^{-6}$
(Ref. Human Event Tree BSPV, Figure 8.9, Tables 8.17 and 8.18)

This procedure requires that the bistable trip on each channel (A, B & C) be set to -680 ± 20 VdC. The probability that the bistable trips will be set inaccurately in either direction is 3×10^{-6} . This is a conservative estimate, since there is a 50:50 chance of setting the trip point too high (producing a scram or alarm before it is needed) rather than too low (failing to produce a scram or alarm when it is needed).

The probability of misadjusting scram trip points for the channels is conservatively assumed to be non-independent. Although the drawers are in separate locked cabinets, the cabinets are transparent and the trip points are set one after the other.

HRA TREE DIAGRAM

ID: *BSPV Tech. sets voltage trip points for linear drawers A,B, & C.*
Sequence: *RHU202QL*
Date modified: *18-JUN-90*
Analyst: *Schurman*
NOTES: *Technician sets high voltage bistable trip points for linear amplifiers of channels A, B, and C. These settings are checked by the Interlock checks before startup of the reactor. (Proc. STM-MC-77 Rev. 8).*

Fig. 8.9. Human Event Tree, BSPV

Table 8.17. HEPs for Human Event Tree BSPV

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A Tech. fails to properly set high voltage trip point.	0.0015 (3)	.5 x Table 20-7 (3)	SBS	1	directional	ZD	0.0015 (3)
B Interlock check fails to discover misadjustment of high voltage trip.	0.001 (10)	Analyst's judgement SBS	SBS	1		ZD	0.001 (10)

Table 8.18. Failure Paths and Total Failure/Success Probabilities for Human Event Tree BSPV

Tech. sets voltage trip points for linear drawers A,B, & C.

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.0015×0.001	0.000001	0.000001	0.999998

8.2.2.5 Subassembly Outlet Temperature Instruments

The SOT system is designed as a backup signal to the wide range nuclear channels as well as for the flow sensors. The calibration and trip point errors are the same as discussed, above, under the paragraphs on Flow Sensor instruments.

8.2.2.6 Failure to Remove Mechanical Latches from Control Rod Release Rollers

There are several procedures, having to do with maintenance on the control rod drive systems, that call for placing mechanical latches on the roller latches (usually held closed by magnetic clutches). The roller latches are spring-loaded against the shaft of the control rod drive mechanism and are held closed by the action of magnetic clutches. Loss or removal of power allows the weight of the control rod drive mechanisms (assisted by pneumatic pressure) to open the roller latches. The control rods then drop from the force of gravity - assisted by an air driven piston. In order to perform maintenance on the control rod drive assemblies, these roller latches are mechanically held closed by U-shaped mechanical latches. In each of these procedures, the placement and removal of the mechanical latches are called out in separate steps. Procedures calling for mechanical latching of one or more control rod roller latches occur an average of four times per year (approximately 96 times in the 25 years of operation). In most procedures, the control rod drive being worked on and two adjacent drives are latched at the same time. Supervision procedures (and good mechanical practice) call for work on only one control rod drive assembly at a time.

Nonetheless, it is conceivable, if unlikely, that two control rod drive assemblies could be worked on at the same time. In such case, as many as six assemblies could be mechanically held closed and disabled from dropping. In order for failure to scram to occur, all but two of the control rods would have to be mechanically latched, since any two control rods will reduce reactivity to subcritical (see Section 7 for a more detailed explanation). The post-maintenance procedures call for a performance check of the control rod drop system. The performance check is performed by timing control rod drop without the air assist. Failure of one or more rods to

fall to the bottom is indicated by a red light and annunciator in the control room. The drop times are recorded on a pre-startup interlock checksheet and must be less than 0.45 sec.

The probability of any control rod remaining mechanically latched after maintenance and that latching being undetected is judged to be negligible.

8.2.2.7 Manual Scram

Event RHU00ZCL, Failure to Manually Scram Safety Rods after Scram Signal: $p = 7.5 \times 10^{-3}$ (Ref. Human Event Tree SCRM, Figure 8.10, Tables 8.19 and 8.20.)

The operators are trained to press the control rod scram button immediately upon receiving a scram alarm and the probability of one of the two console operators failing to take this action approaches zero. The event of concern here, however, is the operator failure to scram the safety rods in the case the control rods fail to shutdown the reactor. An emergency procedure for manual or automatic scram of EBR-II requires the operators to verify that a reactor scram has actually occurred after the operators press the control rod scram pushbutton. If the operators can not verify the shutdown, they are instructed to immediately press the safety rod scram pushbutton. This is failure path .abD in Figure 8.20 and has probability 7.5×10^{-3} .

Event RHU001CL, Failure to Manually Scram in the Absence of a Scram Signal:
Screening $p = 5 \times 10^{-2}$

Event RHU002CL, Failure to Manually Scram in the Absence of a Scram Signal:
Screening $p = 1 \times 10^{-1}$

HRA TREE DIAGRAM

ID: SCRM Operators use full manual SCRAM procedure on failure of auto
Sequence: RHU00ZCL
Date modified: 16-APR-90
Analyst: Schurman
NOTES: Operators go through full manual SCRAM procedure given that automatic SCRAM mechanisms fail, but SCRAM is annunciated. (Proc. EP 2-1 8/8/89)

Fig. 8.10. Human Event Tree, SCRM

Table 8.19. HEPs for Human Event Tree SCRM

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A Operators fail to detect SCRAM announcement.	0.000001	p. 11-48	SBS	3		ZD	0.000004 (10)
B Operators (2) fail to press Control Rod Scram button (Step 1.1.1)	0.	ASEP Table 7-1(d)	SBS	3		ZD	0. (10)
C Operators (2) fail to verify SCRAM has occurred (Step 1.1.2)	0.001	Table 20-22(9)	SBS	3		ZD	0.003 (5)
D Operators (2) fail to push Safety Rod SCRAM button (Step 1.1.3.1)	0.0025	Table 20-2(2) LB	SBS	3		ZD	0.0075 (10)
E Operators (2) fail to turn OFF key switches 1PK & 1PRS (Step 1.1.3.2)	0.025	Table 20-2(2)	SBS	3		ZD	0.075 (10)
F Operators (2) fail to verify SCRAM has occurred (Step 1.1.3.3)	0.001	Table 20-22(9)	SBS	3		ZD	0.003 (5)

Table 8.20. Failure Paths and Total Failure/Success Probabilities for Human Event Tree SCRM

Operators use full manual SCRAM procedure on failure of auto

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .A	0.000004	0.000004		
2 .abD	0.0075	0.0075		
3 .abdEF	0.075 x 0.003	0.000225		
			0.007729	0.99227

In the case of failure of all scram logic, no scram signal occurs at the control console. Determination of the precise human error probability for this scenario is situation-dependent, since different sets of alarms will be present in the control room for each of the top events in the fault trees. A screening value of $1 \cdot 10^{-4}$, using the ASEP values from NUREG/CR-4772 [8.4], Table 8.2, was initially assigned using two assumptions:

- 1) the operators at EBR-II, unlike commercial operators, are thoroughly trained in the philosophy, "When in doubt, scram the reactor!"
- 2) all of the annunciators, therefore, signal the same action - scram the reactor - and can be assumed to be equivalent to a single unequivocal annunciation.

However, more conservative values can be more appropriate, particularly for the stressful situation in which the operators have to determine the need to scram. For instance the DAS, which could be useful in alerting the operators, keeps track of more than 900 parameters and shows on a terminal the parameters that exceed their specified range. However, if a large number of less important parameters appears on the terminal, the operators could be distracted from the parameters that indicate the need to scram. Accordingly, a failure to manually scram in the event of LOF, RHU001CL, is conservatively estimated to be $5 \cdot 10^{-2}$, whereas, in the event of TOP, RHU002CL, the failure is estimated to be $1 \cdot 10^{-1}$. The availability of an emergency procedure for abnormal operation of the primary pumps supports the use of a smaller value in the event of LOF.

In the contrast, the situation for the Loss of Heat Sink is quite different. In this case there is no scram signal that is to be expected and the procedures dictate a manual scram upon diagnosis of the event. The symptoms are clear, slowly developing and the operators are trained to recognize the characteristics of such an event. Since there are two operators the value used is the minimum of $1 \cdot 10^{-6}$ for the event; failure to manually scram in a LOHS, RHU006CL.

8.3 Insufficient Subcriticality Margin during Refueling

A possible vulnerability is evaluated for the EBR-II when it is not at power. The event that can occur during reactor shutdown is criticality occurring during the "Unrestricted Fuel Handling" process. During this activity, fuel and experimental subassemblies are moved around or replaced in the reactor core between almost every reactor run due largely to the research nature of this reactor. Although the reactor is shut down, care must be taken to maintain the core at a subcritical level of reactivity, plus a safety margin of subcriticality.

Subcriticality margins are maintained, according to the Technical Specifications, in order to preclude reactivity of the core attaining criticality while refueling is in progress. Technical Specifications require that the core remain subcritical with both safety rods fully inserted by at least 3.0 \$. There are two major Human Errors that could be involved in a criticality event during refueling: 1) Errors in developing the core loading plan; and, 2) Errors in observing neutron count rates that could lead to failure to terminate fuel handling prior to actuation of the engineered safety systems.

In addition, the engineered safety systems and the operator observation (and tracking) of changes depend on the wide-range nuclear channels A, B, and C. The lower level measuring systems of these channels, Log Count Rate (LCR) drawers, provide information regarding reactivity increases. These instruments also provide safety rod trips in case of high count rates or low count rates from a calibrated neutron source (indicating loss of detector function). Thus, calibration of the LCR drawers of the wide-range systems and setting of the low and high count rate trips are important parts of the safety system for unrestricted fuel handling. For this reason, evaluation of the human error probabilities in performing these functions was required for this analysis.

Event CFA005RL, Errors in Developing Core Loading Plan: Screening $p = 4 \times 10^{-5}$, (Ref. Human Event Tree CLPE, Figure 8.11, Tables 8.21 and 8.22)

The development of the core loading plans are tightly controlled, technically and administratively. The purpose of the plan is to accomplish refueling in such a manner that the reactivity and flux envelope of the core is known and appropriate after the removal and insertion of each subassembly. This refueling activity is categorized as "Unrestricted Fuel Handling" and may involve movement of fuel subassemblies from one location to another within the core, removal of spent fuel subassemblies, and/or insertion of fresh fuel subassemblies. The reactivity worth, burn-up level, and other characteristics of the constituent subassemblies are calculated and balanced in the core "Loading Plan" by the EBR-II Chief Physicist and checked by the Operations Physicist. This loading plan is indirectly checked by the calculation of fissiles and fission rates using a proprietary program in the Argonne, IL, location.

The Operations Physicist performs an independent calculation of expected reactivity at each step of the loading plan. The loading plan is loaded into the Fuel Handling computer at EBR-II and copies are distributed to the main personnel involved in the fuel handling sequence. The operations managers involved in the fuel handling sequence, including the Fuel Management Supervisor and the Reactor Neutronics and Chemistry Manager, independently review and check the loading plan before the fuel handling begins. An ASEP screening value analysis was performed on this sequence of activities and the screening value for human error in the development and implementation of the loading plan was found to be on the order of 4×10^{-5} , as an upper confidence bound. This value was judged to be too low to warrant further analysis.

Event FHU001RZ, Errors in Observing Neutron Count rates: $p = 7.3 \times 10^{-4}$ (Ref. Human Event Tree FHHU, Figure 8.12, Tables 8.23 and 8.24)

During the Unrestricted Fuel Handling process, a Control Room Operator (CRO) is stationed in the control room with the sole job of tracking neutron count rates to ensure that the core does not trend toward criticality. The CRO has the authority to halt the fuel handling

HRA TREE DIAGRAM

ID: *CLPE Error is made in developing Core Loading Plan*
Sequence: *unk*
Date modified: *08-JUN-90*
Analyst: *Schurman*
NOTES: *ASEP screening values used.*

Fig. 8.11. Human Event Tree, CLPE

Table 8.21. HEPs for Human Event Tree CLPE

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A Core Loading Plan miscalculated by physicists.	0.00003 (16)	ASEP Table 4-3	HCR	1	Case VII (2 checks)	ZD	0.00003 (16)
B Core Loading Plan incorrectly trans- lated into loading instructions.	0.00001 (10)	ASEP Table 4-2	HCR	1	rated "negligable"	ZD	0.00001 (10)

Table 8.22. Failure Paths and Total Failure/Success Probabilities for Human Event Tree CLPE

Error is made in developing Core Loading Plan

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .A	0.00003	0.00003		
2 .aB	0.00001	0.00001		
			0.00004	0.99996

HRA TREE DIAGRAM

ID: FHHU Operator fails to follow fuel handling Count-rate Procedure.
 Sequence: Unknown
 Date modified: 04-MAY-90
 Analyst: Schuman
 NOTES: When notified by Fuel-Handling Console Operator - Control Room Operator fails to use and follow Count-rate Surveillance Procedure.

Fig. 8.12. Human Event Tree, FHHU

Table 8.23. HEPs for Human Event Tree FHHU

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A Operator fails to initiate count rate surveillance when requested.	0.000001 (3)	Table 20-5(2)	SBS	1		ZD	0.000001 (3)
B FHCO fails to notify CRO to initiate surveillance.	0.001 (5)	Table 20-6(2)	SBS	1		ZD	0.001 (5)
C CRO misreads Channel A count rate LOW.	0.0005 (5)	.5 x Table 20-10(2)	SBS	1		ZD	0.0005 (5)
D CRO misreads Channel B count rate LOW.	0.0005 (5)	.5 x Table 20-10(2)	SBS	1		MD	0.143285 (5)
E CRO misreads Channel C count rate LOW.	0.0005 (5)	.5 x Table 20-10(2)	SBS	1		MD	0.143285 (5)
F CRO does not catch reading errors when recording data from each channel.	0.05 (10)	Page 11-13/An. Judgment	SBS	1		MD	0.185714 (10)
G CRO fails to calculate CO/CN ratio for Channel A.	0.003 (3)	Table 20-5(1)	SBS	1		ZD	0.003 (3)
H CRO fails to calculate CO/CN ratio for Channel B.	0.003 (3)	Table 20-5(1)	SBS	1		HD	0.5015 (3)
I CRO fails to calculate CO/CN ratio for Channel C.	0.003 (3)	Table 20-5(1)	SBS	1		HD	0.5015 (3)
J CRO fails to write down CO/CN ratios.	0.05 (3)	Table 20-10(10)	SBS	1		HD	0.525 (3)
K CRO fails chart CO/CN ratio for Channel A. (LCB because of written prior step)	0.000333 (10)	Table 20-7(LCB)	SBS	1		ZD	0.000333 (10)
L CRO fails chart CO/CN ratio for Channel B. (LCB because of written prior step)	0.000333 (10)	Table 20-7(LCB)	SBS	1		HD	0.500166 (10)

Table 8.23. HEPs for Human Event Tree FHHU (cont'd)

M CRO fails chart CO/CN ratio for Channel C. (LCB because of written prior step)	0.000333 (10)	Table 20-7(LCB)	SBS	1	HD	0.500166 (10)
N CRO fails to note CO/CN ratio for Channel A is trending out of range.	0.05 (5)	Table 20-10(11)	SBS	1	ZD	0.05 (5)
O CRO fails to note CO/CN ratio for Channel B is trending out of range.	0.05 (5)	Table 20-10(11)	SBS	1	MD	0.185714 (5)
P CRO fails to note CO/CN ratio for Channel C is trending out of range.	0.05 (5)	Table 20-10(11)	SBS	1	MD	0.185714 (5)
Q FHCO fails to warn CRO that count rates are trending out of range.	0.006 (5)	Table 20-10(3)	SBS	1	LD	0.0557 (5)
R CRO fails to terminate fuel handling prior to Safety Rod SCRAM.	0.003 (3)	Table 20-7(2)	SBS	1	ZD	0.003 (3)
S CRO fails to attend to high count rate alarm.	0.001 (3)	Table 20-23(1)	SBS	2	LD	0.0519 (3)

Table 8.24. Failure Paths and Total Failure/Success Probabilities for Human Event Tree FHHU

Operator fails to follow fuel handling Countrate Procedure.

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.000001 x 0.001	*		
2 .aCDEF	0.0005 x 0.143285 x 0.143285 x 0.185714	0.000001		
3 .acGHIJ	0.003 x 0.5015 x 0.5015 x 0.525	0.000396		
4 .acgKLM	0.000333 x 0.500166 x 0.500166	0.000083		
5 .acgNOPQ	0.05 x 0.185714 x 0.185714 x 0.0557	0.000096		
6 .acgnRS	0.003 x 0.0519	0.000155		
			0.000733	0.999266

process if reactivity changes are not occurring according to the loading plan. He/She communicates with the Fuel Handling Console Operator (FHCO) who is in charge of the actual fuel-handling procedure.

The CRO takes a baseline count (referred to as C_0 , in the procedures) that identifies the reactivity of the core before fuel loading begins. He then takes a three-minute countrate reading from each of the wide-range nuclear channels after each insertion of a subassembly (referred to as C_n , in the procedures). The CRO records these readings, then calculates the C_0/C_n ratio. The CRO then plots these readings, for each channel, on a graph and visually verifies that the ratio is not tending toward less than 0.7 on the graph. Thus, this CRO serves as a safety barrier to terminate fuel handling before core reactivity increases to the point that would actuate the automatic safety systems (at 1500 - 2000 counts per second). The human reliability probability for this task is approximately $7.3 \cdot 10^{-4}$.

Although this task appears simple and straightforward, there are a number of error-likely steps involved, such as accurate transcription of meter readings, accurate calculations, and accurate transfer of the calculations to graphs. Therefore, estimated human error probability is higher than was found for the rest of the analyzed tasks.

Event CND001RM, Errors in Calibration for Log Count Rate Instrumentation:

$p = 3 \cdot 10^{-5}$ (Ref. Human Event Tree FHCR, Figure 8.13, Tables 8.25 and 8.26)

Calibration of the wide-range nuclear channels Log Count Rate (LCR) drawers is very important, since both the accuracy of information upon which to perform countrate surveillance and the actuation of automatic safety systems are both dependent upon these instruments.

The calibration of the LCR instruments is part of the calibration of all ranges of the wide-range nuclear channels (STM-MC-77, Rev. 8). However, the calibration for this portion of the channels is separately checked by use of the "Unrestricted Fuel Handling Interlock Checks"

HRA TREE DIAGRAM

ID: *FHCR Calibrate Log Count Rate on channels A, B, & C (fuel handling)*
Sequence: *Unknown*
Date modified: *02-APR-90*
Analyst: *Schurman*
NOTES: *Tech. calibrates instruments for log count rate on Wide Range Nuclear Channels A, B, and C. Checked by the Interlock check prior to Fuel Handling.*

Fig. 8.13. Human Event Tree, FHCR

Table 8.25. HEPs for Human Event Tree FHCR

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A Tech. fails to set adjust power supply properly.(Step 9.1)	0.0005 (3)	.5 x Table 20-7 (1)	SBS	1		ZD	0.0005 (3)
B Interlock check fails to find misadjustment.(STM-MC-14, Log 1-E (H.))	0.005 (5)	Table 20-22 (4, 10)	SBS	1		ZD	0.005 (5)
C Tech fails to adjust discriminator properly. (Step 9.2)	0.0005 (3)	.5 x Table 20-7 (1)	SBS	1		ZD	0.0005 (3)
D Interlock check fails to discover misadjustment.	0.005 (5)	Table 20-22 (4, 10)	SBS	1		ZD	0.005 (5)
E Tech. fails to calibrate Calibrator Signal Select properly. (Step 9.3)	0.0015 (3)	.5 x Table 20-7 (2)	SBS	1		ZD	0.0015 (3)
F Interlock check fails to discover misadjustment. (STM-MC-14)	0.005 (5)	Table 20-22 (4, 10)	SBS	1		ZD	0.005 (5)
G Tech. fails to calibrate Log Amplifier properly. (Step 9.4)	0.001 (3)	Table 20-7 (1)	SBS	1		ZD	0.001 (3)
H Interlock check fails to find misadjustment. (STM-MC-14)	0.005 (5)	Table 20-22 (4, 10)	SBS	1		ZD	0.005 (5)
I Technician fails to perform LCR alignment properly. (Step 9.4)	0.001 (3)	Table 20-7 (1)	SBS	1		ZD	0.001 (3)
J Interlock check fails to catch misadjustment. (STM-MC-14)	0.005 (5)	Table 20-22 (4, 10)	SBS	1		ZD	0.005 (5)
K Tech. fails to properly align LCR Buffer. (Step 9.8)	0.0015 (3)	.5 x Table 20-7 (2)	SBS	1		ZD	0.0015 (3)
L Interlock check fails to catch misadjustment. (STM-MC-14)	0.005 (5)	Table 20-22 (4, 10)	SBS	1		ZD	0.005 (5)

Table 8.26. Failure Paths and Total Failure/Success Probabilities for Human Event Tree FHCR

Calibrate Log Count Rate on channels A, B, & C (fuel handling)

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.0005 x 0.005			
2 .aCD	0.0005 x 0.005	0.000002		
3 .aceF	0.0015 x 0.005	0.000002		
4 .aceGH	0.001 x 0.005	0.000007		
5 .acegIJ	0.001 x 0.005	0.000005		
6 .acegiKL	0.0015 x 0.005	0.000007		
			0.00003	0.99997

(STM-MC 14, Rev. 22) and checked again by the Technical Specification Compliance Log No. 1-E (Rev. 28). The probability of an undetected error in performing this procedure is low (3×10^{-5}).

Event FHU002RZ, Error in Setting Trip Points for High and Low Log Count Rates:
 $p = 5 \times 10^{-6}$ (Ref. Human Event Tree FHSP, Figure 8.14, Tables 8.27 and 8.28)

Similarly, the actuation of the automatic safety systems is dependent upon the accurate setting of the trip points for high count rate. Knowledge of instrument failure is dependent upon the accuracy of the low count rate settings. The trip point settings of the LCR instruments is part of the procedure for calibration of all ranges of the wide-range nuclear channels (STM-MC-77, Rev. 8). However, the setting of trip points for this portion of the channels is separately checked by use of the "Unrestricted Fuel Handling Interlock Checks" (STM-MC 14, Rev. 22) and checked again by the Technical Specification Compliance Log No. 1-E (Rev. 28). The probability of an undetected error in setting the trip points is low (5×10^{-6}).

8.4 Other Human Errors During Operation

In the systems analyses a number of other human errors were identified. Since these errors do not constitute safety significant errors, screening values are used.

8.4.1 Valves and Dampers

In the Shield Cooling System, the basic events listed below concern manual isolation valves which are located in the freon flowpath for backup cooling and are subject to a latent human error of improper valve line-up by being left closed. Since there is no flow in these freon lines during normal operation, the error is assumed to be undetectable until the backup cooling is in operation.

HRA TREE DIAGRAM

ID: FHSP Tech. sets bistable trip points for LCR A,B, & C.(fuel hand)
Sequence: Unknown
Date modified: 03-APR-90
Analyst: Schurman
NOTES: Technician sets LCR bistable trip points for log count rate of channels A, B, and C. These settings are checked by the Interlock checks before start up of the reactor. (Proc. STM-MC-77 Rev. 8: 2/24/87)

Fig. 8.14. Human Event Tree, FHSP

Table 8.27. HEPs for Human Event Tree FHSP

Tech. sets bistable trip points for LCR A,B, & C.(fuel hand)

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.0005×0.005	0.000002		
2 .aCD	0.0005×0.005	0.000002		
			0.000005	0.999995

Table 8.28. Failure Paths and Total Failure/Success Probabilities for Human Event Tree FHSP

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A Tech. fails to set adjust power supply properly.(Step 9.1)	0.0005 (3)	.5 x Table 20-7 (1)	SBS	1		ZD	0.0005 (3)
B Interlock check fails to find misadjustment.(STM-MC-14, Log 1-E(H.))	0.005 (5)	Table 20-22 (4, 10)	SBS	1		ZD	0.005 (5)
C Tech fails to adjust discriminator properly. (Step 9.2)	0.0005 (3)	.5 x Table 20-7 (1)	SBS	1		ZD	0.0005 (3)
D Interlock check fails to discover misadjustment.	0.005 (5)	Table 20-22 (4, 10)	SBS	1		ZD	0.005 (5)

HXV958SL	COIL 715 #2 ISOLATION LEFT CLOSED R8-V-958
HXV959SL	STBY COILS 718 ISOLATION LEFT CLOSED R8-V-959
HXV960SL	COIL 715 #1 ISOLATION LEFT CLOSED R8-V-960
HXV961SL	COIL 718 #1 ISOLATION LEFT CLOSED R8-V-961
HXV962SL	COIL 718 #2 ISOLATION LEFT CLOSED R8-V-962
HXV963SL	STBY COILS 715 ISOLATION LEFT CLOSED R8-V-963
HGV735SL	COIL 715 #2 EXPANSION VALVE LEFT CLOSED R8-VC-735
HGV736SL	COIL 715 #1 EXPANSION VALVE LEFT CLOSED R8-VC-736
HGV737SL	COIL 718 #1 EXPANSION VALVE LEFT CLOSED R8-VC-737
HGV738SL	COIL 718 #2 EXPANSION VALVE LEFT CLOSED R8-VC-738

The mean value used is $1.0 \cdot 10^{-4}$ based on the IREP number for manual valves failure [8.5] to operate which was identified as being dominated by human error (error factor 3). Since the data did not divide the human error and mechanical failures, an error factor of 10 was used. This same number was also applied to the basic events for manual isolation dampers in the shield cooling system listed below.

HDM741SL	ISOLATION DAMPER 741 LEFT CLOSED R8-DM-741
HDM742SL	ISOLATION DAMPER 742 LEFT CLOSED R8-DM-742
HDM746SL	ISOLATION DAMPER 746 LEFT CLOSED R8-DM-746
HDM765SL	ISOLATION DAMPER 765 LEFT CLOSED R8-DM-765
XV962SL	COIL 718 #2 ISOLATION LEFT CLOSED R8-V-962

There is also a basic event for one isolation damper in the Thimble Cooling System which falls in this category.

TDM751SL	ISOLATION DAMPER 751 LEFT CLOSED R8-DM-751
----------	--

8.4.2 Alarms

The human error of failing to respond to a control room alarm is 3×10^{-6} based on NUREG/CR-1278 (THERP handbook) chapter 11, page 48, [8.3] with two operators present. However, recent PRA's have used a value of 1×10^{-4} for this type of error. Therefore, the value used for event LIN12MBZ is 1×10^{-4} .

CSWMS2CL OPERATORS FAIL TO RELEASE MSW WITHIN 1 MINUTE AFTER ALARM

The value in NUREG/CR-1278 page 10-15 is 10^{-5} for two operators failing to respond to an annunciator. This value is increased by a factor of 10 to account for the alarm being caused by a previous error by one of the operators involved and by a second factor of 10 for the possibility that the secondary operator may not be a qualified primary operator. The value used for CSWMS2CL is 1×10^{-3} per demand.

If the alarm is a low priority alarm which persists, it may be considered by the operators to be a nuisance alarm. In that case, the probability to fail to respond other than to silence the alarm may be as high as 1×10^{-2} for the two basic events:

DAN82LCZ HUMAN ERROR BULK SODIUM LOW TEMP ALARM IGNORED

DAN500CZ HUMAN ERROR ANY T/C REF JUNCT > 152F ALARM IGNORED

8.4.3 Maintenance and Testing

The Continuous Power System is subject to a human error in the testing of the UPS batteries and the DC electrical system is subject to the same type of error in the testing of the station battery. The IREP value for this type of maintenance activity of 1×10^{-3} is applied to the basic events:

UINV1BYL	135 VDC UPS 1 BATTERY CAPACITY LOSS NOT DETECTED
UINV2BYL	135 VDC UPS 2 BATTERY CAPACITY LOSS NOT DETECTED
LIN12CBL	125 VDC STATION BATTERY CAPACITY LOSS NOT DETECTED

8.4.4 Acts of Commission

In general actions of commission are not included in a PRA but selected inadvertent actions are included here due to potential consequences, particularly in delaying recovery actions.

In the Balance of Plant (Sec Na & Steam), a possible human error action is an inadvertent operator action to dump secondary sodium. This basic event is:

NSW00CZ INADVERTENT OPERATOR ACTION TO DUMP SECONDARY NA

A mean of $1 \cdot 10^{-4}$ is used. This is the result of starting with the screening number of $1 \cdot 10^{-3}$ and reducing it one order of magnitude to account for the effect of the procedural requirements.

Another area where acts of commission are considered is the reactor control console. Operation of some of the switches has occurred and based on plant data [8.6], the following events are considered:

CSWMSICZ MSW SWITCH HELD UP BY UNAUTHORIZED PERSON

The frequency of CSWMS1CZ is 0.21 per year or $2.4 \cdot 10^{-5} \text{ hr}^{-1}$. For this to happen the console access restriction must be violated.

CSWCONCZ CONSOLE ACCESS RESTRICTION VIOLATED

The probability for CSWCONCZ used in the EBR-II PRA is 5×10^{-2} demand. This value is based on the NUREG/CR-1278 Table 20-20 [8.3] value of 1×10^{-2} and an estimate of 5 demands per year for EBR-II.

CSWMSZCZ OPERATOR HOLDS UP MSW UNTIL ALARM OCCURS

A value of $2 \times 10^{-2} \text{ yr}^{-1}$ or $2.3 \times 10^{-6} \text{ hr}^{-1}$ is based on plant data [8.6].

CSWMSCCL MSSW NOT SET TO NEUTRAL POSITION

NUREG/CR-1278 page 13-9 has a value of 3×10^{-3} per demand for a similar error [8.3]. This value is converted to an hourly value of $1.67 \times 10^{-4} \text{ hr}^{-1}$ to account for the demand to set the switch to neutral every 18 hours.

In the Shutdown Cooler System there is an air operated valve, PA-154, which could cause inadvertent excess heat removal through the shield cooling system if inadvertently operated. This valve appears in the system notebook only and not in the accident sequence evaluation since excess shield cooling heat removal is an operational problem affecting plant efficiency and not an initiating event. The basic event is:

DHU154PZ HUMAN ERROR IMPROPER VALVE OPERATION PA-154

The value used is 1×10^{-4} which is a screening number.

8.5 Human Errors During Recovery

The design of LMRs allows significant time for recovery action, particularly of decay heat removal systems. The minimal dependance on electrical systems allows recovery of many of these systems also.

8.5.1 Recovery of Electrical Systems

In the electrical systems, there are a number of alternate power supplies to various busses which involve either the closing of manual breakers or the switching of manual transfer switches. The human error in these cases is dominated mostly by the ability of the operator to diagnose the problem rather than to take the specific action. Loss of the normal power to an electrical bus is annunciated and the cause is usually obvious. For this reason the screening value of $1 \cdot 10^{-4}$ was used. The breaker related human error basic events are as follows:

ECB11BYZ	OPERATOR ERROR IN CONNECTING E-1 TO R-1B
ECB1PBYZ	OPERATOR ERROR IN CONNECTING E-1 TO P-1B
ECB2DHYZ	OPERATOR ERROR FAILS TO CLOSE E-1 BREAKER 2D
JCBN3AVZ	OPERATOR FAILS TO CLOSE NORM BUS NO.1 BRKR 3A

The manual transfer switch basic events are:

ESWE1RYZ	OPERATOR FAILS TO EFFECT E-1 TO R-2B LINE ON DEMAND
ESWP4NYZ	OPERATOR FAILS TO SWITCH P-4E MAN. TRANSF. SW TO NORM. PWR
LSWME1BZ	OPERATOR FAILS TO CLOSE 125 VDC MTS 1 EMERG BUS SIDE
LSWME2BZ	OPERATOR FAILS TO CLOSE 125 VDC MTS 2 EMERG BUS SIDE
LSWME3BZ	OPERATOR FAILS TO CLOSE 125 VDC MTS 3 EMERG BUS SIDE

LSWME4BZ OPERATOR FAILS TO CLOSE 125 VDC MTS 4 EMERG BUS SIDE
LSWUPTBZ OPERATOR ERROR FAILS TO SWITCH TO UPS 1 BAT.

8.5.2 Recovery of Decay Heat Removal Systems

The primary recovery action considered is in the event of the irrecoverable loss of both shutdown coolers while the secondary system is drained.

Event BOO01ZOO, Failure to Recover from loss of Decay Heat Removal ($p = 4 \times 10^{-6}$)
(Ref. Human Event Tree, Figure 8.15, Tables 8.29 and 8.30)

This event is presumed to result from the loss of both shutdown coolers during shutdown, with the secondary sodium loop drained. There are two operator actions that will recover from this situation: refill secondary loop to institute cooling from Balance of Plant (requires 48 hours to complete from cold pipes - or about 3 1/2 hours if the system is heated) or start shield/thimble cooling. Any one of these two actions will serve to delay heat-up to unacceptable temperatures while other measures are instituted or additional problem-solving help (engineering staff) is obtained.

These two actions provide opportunity for two human errors of omission in addition to failure to respond to the annunciators, failure to perform these actions could result in overheating. The Human Error Probability (HEP) for each of two three actions is listed below. The total failure probability, given above, is the product of these two individual probabilities. These probabilities are estimated conservatively from Table 20-1 of NUREG-1278 [8.3], using the assumption that only 60 minutes is available for diagnosis and action.

HRA TREE DIAGRAM

ID: ARG1 OPERATORS FAIL TO RECOVER FROM LOSS OF CORE COOLING
Sequence: [TBD]
Date modified: 25-SEP-90
Analyst: SCHURMAN
NOTES: THIS EVENT OCCURS WITH THE LOSS OF BOTH SHUTDOWN COOLERS WITH THE SECONDARY SODIUM LOOP DRAINED

Fig. 8.15. Human Event Tree, ARG1

Table 8.29. HEPs for Human Event Tree ARG1

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A OPERATORS FAIL TO DIAGNOSE PROBLEM WITHIN 60 MIN. AFTER LOSS IS ANNUNCIATED	0.000003 (30)	TABLE 20-3 (5)	LBDYN	1	CLEAR SIG.	ZD	0.000003 (30)
B OPERATORS FAIL TO START REFILL OF SECONDARY LOOP	0.004 (5)	TABLE 8-5 (3)	LB DYN	1	PRACTICED	ZD	0.004 (5)
C OPERATORS FAIL TO START SHIELD COOLING	0.02 (5)	TABLE 8-5 (3)	DYN	1		ZD	0.02 (5)
D OPERATORS FAIL TO INCREASE SODIUM PURIFICATION FLOW	0.02 (5)	TABLE 8-5 (3)	DYN	1		ZD	0.02 (5)

Table 8.30. Failure Paths and Total Failure/Success Probabilities for Human Event Tree ARG1

OPERATORS FAIL TO RECOVER FROM LOSS OF CORE COOLING

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .A	0.000003	0.000003		
2 .aBCD	$0.004 \times 0.02 \times 0.02$	0.000001		
			0.000004	0.999995

The operators are trained on this situation and trained to use shield cooling and the cooling effects from increased sodium purification flow to maintain shutdown cooling margins. They are trained and practiced on refilling the secondary loop and viewing the secondary loop as a cooling factor in controlling core temperature (although the usual case is throttling secondary loop circulation to prevent overcooling during normal shutdowns). Nonetheless, the operators are trained, in case of loss of cooling, to use or return to use the secondary sodium system.

Operators fail to diagnose problem within 60 minutes after loss of shutdown cooling is annunciated, ($p = 3 \times 10^{-6}$, Table 20-3(5), Lower Bound)

The lower bound is used because operators, in fact, have much more than 60 minutes to respond and make a diagnosis and because the diagnosis is quite clear.

Operators fail to refill Secondary Loop, ($p = 4 \times 10^{-3}$, ASEP (NUREG/CR-4772) Table 8-5 Item (3), Lower Bound)

This action is practiced and the time required is limited by the temperature of the secondary loop pipes and their heat-up time. The lower bound was used because the action is well-learned.

Operators fail to start Shield Cooling, ($p = 2 \times 10^{-2}$, ASEP (NUREG/CR-4772) Table 8-5 Item (3))

The median value is used in this case because the task is trained during initial training and discussed periodically, but not retrained or practiced.

The recoverable loss of shutdown coolers is modeled in the decay heat removal trees and dominated by the failure to diagnose the problem 10^{-6} as shown in the analysis above.

DHU000OL HUMAN ERROR - FAILURE TO RECOVER COOLERS

Action to restore the recoverable loss of shutdown coolers is simple and may be accomplished by a number of diverse actions (e.g. disconnecting electric power to air solenoids, closing off instrument air to the cooler actuators, inputting high temperature test signal, venting air from shutdown cooler actuators, or removing linkage pins). Many of these actions are practiced. The temperature test signal is input before every reactor run. The closing of instrument air and venting actuators is done at least twice per year during the cooldowns associated with the long shutdown.

In other systems such as shield cooling, a spurious breaker trip of a component resulting in a signal based on current, voltage, or power has a higher probability of noncorrective action than an alarm based on flow which indicates a more urgent need for action. In both cases, the values used depend on the existence of specific procedures as opposed to operating in the troubleshooting mode. The following values are used: (1) Flow alarm with recovery procedure, $1 \cdot 10^{-2}$; (2) current, voltage, power alarm with recovery procedure, $1 \cdot 10^{-1}$; (3) Flow alarm in troubleshooting mode, $5 \cdot 10^{-1}$; (4) current, voltage, power alarm in troubleshooting mode, $5 \cdot 10^{-1}$.

In the Shield Cooling System and Thimble Cooling System the value of $2 \cdot 10^{-2}$ is used for the basic events,

HHU002YL NO RECOVERY ACTION TO RESET CIRCUIT BREAKER

THU002YL NO RECOVERY ACTION TO RESET CIRCUIT BREAKER

Failure of recovery actions for shifting the filters in the event of a clogged filter is $2 \cdot 10^{-2}$ since this action is performed on an annual basis for filter replacement.

HHU01BSZ NO RECOVERY ACTION FOR CLOGGED FILTER R8-FL-717

The motor dampers in both the Shield Cooling System, the Thimble Cooling System, and the Stack Exhaust System have integral handwheels in the motor operators for use in the event that the motor fails. The actions to use these handwheels are taught during operator qualification on the systems. There is no repeat training on these values but there is training on the steam

isolation valve located in the sodium boiler building which is of similar construction and operation but different in size. The value used therefore is $1 \cdot 10^{-2}$ for the basic events for Shield and Thimble Cooling dampers.

HHU003SL NO RECOVERY ACTION TO USE DAMPER MANUAL HANDWHEEL

HHU003YL NO RECOVERY ACTION TO USE DAMPER MANUAL HANDWHEEL

THU003SL NO RECOVERY ACTION TO USE DAMPER MANUAL HANDWHEEL

Dampers in the stack exhaust system are operated by non-EBR-II personnel and a higher failure rate of $1 \cdot 10^{-1}$ was used.

HHU004SL NO RECOVERY ACTION TO USE MOVE DAMPER MANUALLY

The pneumatic dampers in both the Shield Cooling System and the Thimble Cooling System require either a connecting pin or bolt to be removed for manual operation if the pneumatic operators fails. These actions to remove the pin or bolt are taught during operator qualification on the systems and there is annual training on the actions. The value used for failure to recover is $1 \cdot 10^{-1}$ for the basic event,

HHU004SL NO RECOVERY ACTION TO USE MOVE DAMPER MANUALLY

In the event that flow in the Thimble Cooling System is lost due to failure of either the exhaust fans or exhaust flowpath, the system can be cross-connected to the Shield Cooling System by opening a manual damper. For failure to perform this recovery action a number of $2 \cdot 10^{-2}$ is used for the basic event,

THO01BSZ NO RECOVERY ACTION TO CROSS-CONNECT SHIELD COOLING

8.5.3 Recovery of Control Rod Malfunctions

There are several recovery actions associated with control rod malfunctions.

CSWMSWCZ OPERATOR FAILS TO FREE STUCK SWITCH

A value of 3×10^{-1} per demand is used although a failure such that the operator would be physically unable to open the up contacts appears to be extremely unlikely, based on examination of the switch. The operator did free the switch in the one event when it stuck [8.6]. It is a natural reaction for him to try.

CSWPSBCL OPERATOR FAILS TO PUSH PSB2

A value of $1.67 \times 10^{-4} \text{ hr}^{-1}$ is used based on NUREG/CR-1278 page 13-9 [8.3] value of 3×10^{-3} per demand for a similar error and converted to 1 hr^{-1} to use the code input option that can account for the demand to set the switch to neutral every 18 hours.

CSWMS4CZ OPERATOR HOLDS MSW UP INSTEAD OF DOWN

CMWMS4CZ is based on the NUREG/CR-1278 Table 12-1 value of 5×10^{-4} with a Performance Shaping Factor (PSF) of 50 to account for a stress level between moderately high and extremely high page 17-17 [8.3]. This gives a mean for CSWMS4CZ of 2.5×10^{-2} per demand.

CSWMSICL OPERATORS FAIL TO FREE MSW FROM UNAUTHORIZED PERSON

A value of 5×10^{-2} is used.

NUREG/CR-1278 Table 20-16 states that failure to detect an error of commission is the basic HEP $\times 10$. Since CSWCONCZ is based on $0.01 / \text{d}$, this implies 0.1. However, the high

flux alarms should reduce the failure probability by at least a factor of 2. No credit is taken for the fact that two operators are in a position to intervene.

CSWMS3CZ OPERATOR CLOSES RELAY, THEN FAILS TO COMPENSATE WITH
CONTROLLING ROD

The probability of compensation depends on whether the controlling rod is #4. Assuming the operator does not take the unusual action of bringing another rod into play if #4 is the controlling rod. For #4 to be the controlling rod the probability is $1 \cdot 10^{-1}$. If #4 is not the controlling rod (probability $9 \cdot 10^{-1}$) the operator will be expecting to balance rod #4 with another rod. Under normal conditions this action is similar to item 7 on page 13-6 of NUREG/CP-1278, which has an HEP of 0.003. But rod #4 will move before it is expected to, which can be accounted for by a PSF. The stress level is judged to be moderately high, which introduces a PSF of 5. The failure role is therefore:

$$(0.1) (1) + (0.9) (3 \cdot 10^{-3}) (5) = 1.1 \cdot 10^{-1}$$

CHU0P1CL OPERATOR DECIDES TO COMPENSATE WITH 2ND ROD

Value used is 0.1 per demand. There is no procedure for such action. It is inconsistent to make this probability large when corrective action has been assumed not to occur with the causes that dominate run-in frequency for the first rod.

8.6 Summary

The human error probabilities derived in this section have been used in quantifying the accident sequence trees. The results of the quantification are given in Sections 11, 12 and 13 while a complete listing of the fault tree used in the quantification can be found in Appendix B.

The results of the human reliability analysis described here show a very limited dependence of EBR-II on human error. Those events which do have significance are related to failure to scram, one maintenance error possibility and one dynamic error possibility, and failure

to recover from a failure of decay heat removal. The last event has a low probability but is important nonetheless because the decay heat removal system is otherwise reliable.

Section 8.0 References

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- 8.2 "Guidelines for the Preparation of Emergency Procedures," NUREG-0899, USNRC, August 1982.
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- 8.4 A. D. Swain, "Accident Sequence Evaluation Program: Human Reliability Analysis Procedure," NURGEG/CR-4772, Sandia Laboratories, February 1987.
- 8.5 R. J. Borkowski, et.al., "The In-Plant Reliability Data Base for Nuclear Plant Components: Interim Data Report - The Valve Component," NUREG/CR-3154, December 1983.
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9.0 SEISMIC ANALYSIS

9.0 SEISMIC ANALYSIS

(Preparation of this section in progress; publication date is TBD)

10.0 EXTERNAL EVENTS

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10.0 EXTERNAL EVENTS

10.1 Introduction

External events, as assessed in PRAs, typically include seismic activity, weather-related phenomena such as tornadoes, high winds, and lightning along with floods and aircraft impact, to evaluate their potential contribution to plant risk. In addition to these natural and man-induced hazards, the scope for some PRAs is extended to include internal flooding, fires, and turbine missiles. Although these latter events are not external events in the strict sense, they have the potential to defeat multiple plant safety systems, and are therefore, generally included under the external events category.

10.1.1 Identification of External Events

Selection of the external events to be considered for the EBR-II PRA involved a two-tier screening process. The first level of screening was based upon guidance in the PRA Procedures Guide (NUREG/CR-2300) [10.1], which provides a suggested list of natural and man-made external events that should be considered in a PRA study. It also suggests applying a set of screening criteria to identify only those significant external events to be included for further analyses. Table 10.1 is the list from the PRA procedures guide. The screening criteria suggested by the PRA procedures guide were:

1. The event is of equal or lesser damage potential than the events for which the plant has been designed.
2. The event has a significantly lower mean frequency of occurrence than other events with similar uncertainties and could not result in worse consequences than those events.
3. The event cannot occur close enough to the plant to affect it.
4. The event is included in the definition of another event.

After applying these criteria as shown in Table 10.1, the following external events were subjected to the second tier of the screening process:

- Fires (Liquid Metal and Conventional)
- Tornadoes and High Winds
- Volcanism
- External Floods
- Lightning
- Aircraft Impact
- Accident at Other INEL Facilities
- Turbine Missiles
- Internal Floods
- Seismic

10.1.2 Screening Methodology

Although external initiating events may impact facility operation and result in plant unavailability, they are only considered important initiating events if they can require plant shutdown, either automatically or manually, in response to the external event, and lead to a core vulnerable condition at a frequency commensurate with those sequences initiated by internal events. While seismic events are generally expected to dominate the risk contribution from external events, some PRA studies have found events such as fires and internal floods to significantly contribute to overall plant risk usually through the total loss of electrical zones.

The following sections describe the specific analysis approach used to address each external event except seismic. Seismic events, due to their relative importance, are covered in Section 9.

Table 10.1. Natural and Man-induced External Events
to be Considered in PRA Studies

Event	Applicable screening criterion [10.1]	Remarks
Aircraft impact	--	Requires further analysis
Avalanche	3	Can be excluded for EBR-II site
Coastal erosion	3	Can be excluded for EBR-II site
Drought	1	Excluded since the ultimate heat sink (shutdown coolers) is not affected by drought
External flooding	--	Requires further analysis
Extreme winds and tornadoes	--	Requires further analysis
Fire	--	Requires further analysis
Fog	4	Considered under man-made hazards involving aircraft
Forest fire	1, 3	Fire cannot propagate to the site because the site is cleared
Frost	1	Snow and ice govern
Hail	1	Other missiles govern
High tide, high lake level, or high river stage	4	Included under external flooding
High summer temperature	1	Ultimate heat sink is designed for at least 45 days of operation, even under high summer temperatures
Hurricane	3	Can be excluded for EBR-II site
Ice cover	1, 4	Ice blockage of river included in flood
Industrial or military facility accident	--	Requires further analysis
Internal flooding	--	Requires further analysis
Landslide	3	Can be excluded for EBR-II site
Lightning	1	Considered in plant design and in analysis

Table 10.1. Natural and Man-induced External Events
to be Considered in PRA Studies (Cont'd)

Event	Applicable screening criterion [10.1]	Remarks
Low lake or river water level	1	Ultimate heat sink is designed for at least 45 days of operation; does not require cooling water
Low winter temperature	1	Ultimate heat sink standby operation precludes effects of icing and thermal stresses
Meteorite	2	Insignificant frequency of occurrence
Pipeline accident (gas, etc.)	3	Can be excluded for EBR-II site
Intense precipitation	4	Included under external and internal flooding
Release of chemicals in onsite storage	1, 3	Limited quantities of hazardous chemicals quantities which could effect plant safety stored onsite. Chemical processing plant several miles distant
River diversion	1, 4	Considered in the evaluation of external flooding
Sandstorm	1	Potential blockage of air intakes with particulate matter is generally considered in plant design
Seiche	3	Can be excluded for EBR-II site
Seismic activity	--	Requires detailed study
Snow	1, 4	Ultimate heat sink standby operation precludes effects of icing and thermal stresses; snow melt causing river flooding is included under external flooding
Soil shrink-swell consolidation	1	Site-suitability evaluation and site development for the plant are designed to preclude the effects of this hazard

Table 10.1. Natural and Man-induced External Events
to be Considered in PRA Studies (Cont'd)

Event	Applicable screening criterion [10.1]	Remarks
Storm surge	4	Included under external flooding
Transportation accidents	2,3	Vehicle traffic on-site controlled, closest highway over 3 miles away
Tsunami	3	Can be excluded for EBR-II site
Toxic gas	3	Can be excluded for EBR-II site
Turbine-generated missile	--	Requires further analysis
Volcanic activity	--	Requires further analysis
Waves	3	Can be excluded for EBR-II site

10.2 Sodium Fires

Sodium is the primary coolant in EBR-II, and is also used for secondary system heat transport to the plant steam evaporators and superheaters. NaK (sodium-potassium) coolant is used for the shutdown cooler natural circulation loops and in the cold trap cooling system.

Although of different chemical compositions, both sodium and NaK may burn in air. At room temperature, NaK is a liquid which ignites on contact with air. Sodium is a solid which does not undergo spontaneous combustion at room temperature. The elevated temperatures in EBR-II during normal operation are such that the sodium remains in a molten state, and may ignite on contact with oxygen, if a leak should occur.

Therefore, leaks or spills of sodium or NaK coolant in areas of the plant where the atmosphere is not inerted, may result in a fire and the release of sodium aerosols and reaction product aerosols, principally sodium oxide. Sodium oxide reacts when in contact with air and water vapor in air to form sodium carbonate and caustic sodium hydroxide which may cause corrosion of structural materials. Sodium oxide will also react when in contact with the water present in concrete to create sodium hydroxide, which is highly corrosive in nature, and hydrogen, [10.2]. Incidences of sodium fires have occurred at EBR-II, as discussed further in Section 10.2.3.5.

10.2.1 Sodium-Fire Methodology

The purpose of the analytical method discussed in this section is to aid in the identification of a list of the dominant accident sequences that are initiated by fire and then to assess the frequency of occurrence for each. This process requires information about several important aspects of a fire (e.g., ignition, progression, detection and suppression, characteristics of materials under fire conditions) as well as the plant safety functions and their behavior under accident conditions.

The objective of a PRA-based fire-risk analysis is to estimate the frequency of fire-induced radionuclide releases of varying magnitudes. These releases would occur as the result of core damage and containment failure, both of which, potentially, could be initiated by the same fire.

Following the lead of NUREG/CR-2300, "PRA Procedures Guide" [10.1], the method for quantifying the risk due to fires can be divided into four somewhat independent steps: the fire-hazard analysis, where critical plant areas are identified and the frequency of fires assessed; the fire-propagation analysis, where the behavior of fires in the critical areas is modeled; the plant-systems analysis, where the likelihood of the fires leading to plant-damage states is estimated; and the release-frequency analysis, where the frequencies of accident sequences leading to radionuclide releases are derived from the results of the preceding analyses.

The first step in any risk analysis is to identify the specific sequences of concern. In the case of fires, sequences of concern are those in which a fire has caused an initiating event and in addition resulted in a sufficient amount of failures to cause some degree of core damage. Release of radionuclides outside of the plant containment can then be estimated based upon an analysis of fire-induced damage to the containment boundary.

Given that a fire-related initiating event requiring plant shutdown has occurred, failure which could result in core damage can be divided into three general classifications of critical safety function failure, as follows:

- 1) Failure to reach and maintain a condition of negative reactivity.
- 2) Failure to monitor and control primary system coolant inventory.
- 3) Failure to remove decay heat.

A working knowledge of the plant and its system interactions is necessary to ensure that a thorough investigation of failures which fall into one of these three categories is performed.

Thus, the fire-risk analysis is generally performed in parallel with or near the end of the event tree and fault tree analysis performed for the internal events portions of the PRA. That is, once the event tree and fault tree models have been completed to represent the expected plant response to internal initiating events and resulting dominant accident sequences, these models can be utilized to determine the impact (e.g., fragility analyses) of the external initiating events.

10.2.2 Screening by Location

In many of the previous PRA studies that analyzed external events (e.g., Zion [10.21], and Indian Point [10.22]), the fire-induced loss of safety system instrumentation or control was judged to dominate fire-induced hardware losses. Therefore, the critical locations investigated should be those which contain electric cables and/or switchgear for many safety and non-safety systems, the failures of which initiate a requirement for reactor shutdown and fall into one of the three critical safety function failure categories previously described.

In order to greatly reduce the effort required for detailed external initiating event analysis, e.g., for fire or internal floods, a screening approach can be used to limit the events that have to be examined in detail. This approach relies upon PRA-based core vulnerability arguments. That is, although external initiating events may impact facility operation and result in plant downtime, they are only considered important initiating events as assessed by the PRA if they can require plant shutdown either automatically and/or manually in response to the external event, and lead to a core vulnerable condition at a frequency commensurate with those sequences initiated by internal events.

10.2.2.1 Negative Reactivity Functional Failure

The inability to trip the reactor, i.e., in the case of the EBR-II reactor, drop the control rods out of the core through automatic action of the reactor protection system during a rapid loss of flow (LOF) or transient overpower condition (TOP) could result in a short-term fuel vulnerable condition. Therefore, fire-related initiating events which could affect the ability to scram the reactor, and/or result in a positive reactivity insertion or loss of

flow are potentially significant initiators for EBR-II. To achieve this type of impact, a sodium-fire initiator would have to occur within containment in the vicinity of the control rod drive mechanisms, reactor shutdown system circuitry, or primary pump control cables.

10.2.2.2 Primary Coolant Inventory (LOSA) Functional Failure

There are no credible mechanisms by which a sodium fire could induce any of the postulated LOSA initiators discussed previously in Section 5 of the internal initiating events analyses. However, the converse case of fire resulting from postulated LOSA initiators has been considered. These fall into the categories of air ingress, small leak, and large rupture.

In the case of air ingress, the oxidation of sodium at the sodium surface in the primary tank is inhibited by the argon cover gas so that the reaction is slowed. The problem then becomes one of high sodium plugging temperature which is an operational problem if the amount of oxygen exceeds the capacity of the cold trap.

Small leaks in the sampling system or test equipment are limited by the systems pumping capacity and would occur in confined spaces, and therefore would not lead to core damage. A small leak from the primary vessel would be limited to the annulus space between the primary vessel and guard vessel which is filled with argon gas. In order for a small leak to react in air, both primary and guard vessels would have to breach. The probability of primary tank leak event is $1 \cdot 10^{-8}$. Breach of both primary and guard vessel is below the 10^{-10} cutoff.

The only remaining LOSA initiator is the large rupture (e.g. one caused by overpressurization of the primary tank and/or annulus. This initiator leads to direct core damage and therefore is not included in external events to avoid double counting.

EVENT TREE FOR: Liquid Metal Fire

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Fig. 10.1. Event Tree For Sodium Fire Sequences

In particular, this would include the piping beneath the reactor building operating floor deck plates for the sodium primary purification system and Radioactive Sodium Chemistry Loop (RSCL), the secondary system sodium piping to/from the IHX within containment, the shutdown cooler NaK piping, the Fuel Element Rupture Detection (FERD) system piping and the Breached Fuel Test Facility (BFTF).

The Breached Fuel Test Facility (BFTF) uses primary system sodium for cooling and for transport of fission products from breached fuel experiments to the neutron detector that is at an elevation above the bottom of the primary tank. The head is provided by the primary system coolant pumps. In order to pump sodium out of the primary vessel, two barriers would have to be breached. The first is the detector dry well and the second is the seal in the instrument box. Both the dry well and the instrument box have leak detectors. Corrective action in case of a leak would be to shut down the reactor and trip the primary coolant pumps. The instrument box is also pressurized and monitored so that a leak would be detected early. The pressure at the seal in the instrument box for a high flow rate subassembly, e.g., XY24, would be about 8.8 psig after correcting for elevation loss. The leak rate would be from 1.4 to 14 gpm depending on the area of the breach.

The Fuel Element Rupture Detection (FERD) system removes primary sodium from the outlet region of the IHX for the purpose of monitoring the level of delayed neutron emitting fission products in the primary coolant as a measure of fuel element failure. Piping carries the sodium out of the primary system through the E-3 nozzle plug to the detector system located in the depressed area of the reactor building operating floor. The flow is through a magnetic flowmeter, the detector assembly, a DC electromagnetic pump, and back to the primary tank, discharging below the sodium surface. The head for the system flow is provided solely by the EM pump which is designed to provide a 5 psig head at the rated 100 gpm flow. A leak in the piping carrying the primary sodium to the detectors would not provide a means of sodium pumpout if the leak was located before the EM pump. Sodium could possibly spill out, but there would be no pumping force to remove additional sodium from the tank. A leak after the pump would continue the sodium flow as long as the pump was powered. The reactor building smoke

detectors would respond to a large sodium spill in the depressed area, and the system could be shutdown by turning off the pump.

The primary sodium purification system provides for the continuous purification of the primary sodium to control impurities to lessen the corrosion rate of reactor containment and reactor materials. The primary sodium is drawn to a surge tank in the purification system by applying a vacuum at the top of the surge tank. The sodium flows from the primary tank, out the piping through the "V" nozzle, and enters the side of the surge tank. The sodium leaves the bottom of the surge tank and flows to the primary purification cell, via a DC powered EM pump which provides the flow through the purification system before returning the sodium to the primary tank through the "V" nozzle. A syphon-break system prevents syphoning of primary tank sodium should a leak occur since other than the surge tank, most of the primary purification system and RSCL piping is located below the sodium level in the primary tank.

The secondary sodium system provides for heat transport from the primary sodium via the IHX to the superheaters and evaporators, and then into a sodium surge tank. The secondary flow is provided by an electromagnetic pump in the secondary system which returns cooled sodium from the surge tank back to the IHX. A leak in the secondary system would require the operators to scram the reactor, trip the secondary system EM pump and dump the secondary sodium to the storage tank.

There are two shutdown coolers in the EBR-II system, each completely independent and providing for heat removal from the primary tank when the secondary sodium system is not in operation. NaK coolant flows by natural circulation from a bayonet-type heat exchanger that extends through a nozzle in the primary tank cover down into the bulk sodium, to an outside air heat exchanger where it is cooled and returned to the bayonet heat exchanger. Since this system is completely sealed and driven by natural circulation, any breach removes the driving force that provides the circulation.

10.2.3.2 Initiating Event Frequency

The probability of piping failures in these systems leading to a loss-of-coolant-accident from the primary vessel was examined previously in Section 5 of the internal initiating event discussion. The LOSA analysis was primarily concerned with events that could lower sodium in the primary tank below the suction intakes of the primary coolant circulation pumps. However, initiating event frequencies corresponding to leaks in the piping systems exiting the primary tank cover region as discussed previously were determined based upon a piping failure rate of $3 \times 10^{-9} \text{ hr}^{-1} \text{ ft}^{-1}$ per foot of piping and a typical reactor run time of 2400 hours, assuming that there are approximately three reactor runs per year. This yielded initiating event frequencies for a piping leak for each of the previous systems described above as follows: BFTF at 7.2×10^{-4} per year; FERD at 9.1×10^{-4} per year; primary sodium purification system (excluding in-cell piping) at 2.2×10^{-3} per year; secondary sodium system piping within containment at 1.5×10^{-3} per year; and the shutdown coolers system piping had a frequency of 4.3×10^{-3} per year for a leak in either system.

However, as noted earlier, for a primary sodium leak to occur in the BFTF requires failure of two barriers: first the detector dry well and then the seal in the instrument box. Should this occur, the dry-well leak detector provided in the system should alert the operators to the problem. Because the only driving force for the flow in the BFTF are the primary coolant pumps, any action that would stop or reduce the primary coolant flow will also stop or reduce the flow in the BFTF and therefore the LOSA potential. For these reasons, the BFTF leak is not considered a significant initiator for a sodium fire that would threaten the integrity of the RSS and primary pump cabling.

Most fire studies assume that if a fire is of sufficient magnitude, that all components or cabling will eventually be damaged due to either the heat generation of the fire or propagation of the fire within that fire zone. The radial beams supporting the reactor vessel as shown in Fig. 10.2 effectively divide the area below the reactor building operating floor deck plates into various sectors, and serve as a barrier to limit the effect of a sodium fire to that sector. As shown by Fig. 10.2, the relative location of the FERD and primary sodium purification system

ZONE 1 DETECTORS LOCATED 15' ABOVE DECKPLATES
 DETECTOR 1-07 FLAME DETECTOR, ON CEILING LIGHTS
 ZONE 7 DETECTORS LOCATED IN DEPRESSED AREA
 ZONES 9-14 LOCATED UNDER DECKPLATES

Fig. 10.2. Reactor Building (767) Operating Floor Fire Detectors

piping is opposite the sector containing cables for the CRDM's and primary pumps. Therefore, a fire due to a leak in the FERD or primary sodium purification system piping would not be expected to propagate or effect the cables for the CRDM's and primary pumps due to shielding from the thermal effects of the fire provided by the radial beam structures.

However, since the piping for the secondary sodium system and the shutdown coolers exits the primary tank cover region in the same relative location as the CRDM's and primary pump cables, the effect of a sodium or NaK fire due to a leak in a piping system in this area could fail these cables due to the burning of cable insulation and shorting of circuit conductors. While there is separation of cables in this sector of the compartment below the reactor building operating floor deck plates, the intensity of the fire due to a sodium or NaK leak is assumed to affect all cabling within this compartment area. This simplifying assumption is conservative since it implies that any fire which starts in this area will propagate and affect all safety-related cables in this area unless detected and extinguished. It is also conservative in that it bases the shutdown coolers NaK leak on the total length of piping rather than the part of piping within containment.

The initiating event frequency is therefore, based upon the occurrence of a leak in either of the shutdown coolers system piping ($4.3 \cdot 10^{-3}$ per year) or the secondary sodium system piping ($1.5 \cdot 10^{-3}$ per year) that could impact components of the reactor shutdown system, primary pumps, or decay heat removal systems.

10.2.3.3 Sodium Fire Detection

Referring to the event tree presented as Figure 10.1 for analyzing the effect of a sodium fire, it can be seen that separate events are included to account for detection and suppression methods for limiting the impact of fire.

The two obvious methods of detection in any area are manual (i.e., visual) or automatic (with smoke/flame detection). To estimate the probability of visual detection requires the determination of the frequency of routine walkthroughs, test or maintenance, in this case, in the

reactor building. Since this frequency is not readily determined, the availability of automatic smoke detectors is relied upon to determine if adequate detection capability exists. Eight redundant photoelectric smoke detectors are located in six separate zones underneath the reactor building operating floor (see figure 10.2). This fire detection system was installed in 1980 [10.3] and provides alarms in the main control room and at the fire station to alert the operators if a fire in any of these areas should occur. The sensors are smoke tested and calibrated every six months [10.4]. None have failed since the original installation. There are also two ionization detectors in the depressed area and six ionization detectors and one flame detector above the operation floor to provide backup means of sodium fire detection in the reactor building (See Fig. 10.2). In addition to the fire detectors, increasing levels of airborne radioactive contaminants in the reactor building would provide a diverse means of indication. Based upon the various number of redundant smoke detectors to alarm the operators to a fire in the reactor building, the probability of detection for a significant sodium or NaK fire is estimated to be approximately 1.0. A screening value of 1.0×10^{-6} for failure to detect a liquid metal fire prior to core and structural damage is used in the sequence quantification.

10.2.3.4 Reactor Shutdown

If detection of the fire under the operating deck plates is successful, the operators are procedurally instructed by Emergency Procedure 1-4 [10.5] to scram the reactor and monitor the reactor building radiation monitors. Failure of the operators to scram the reactor could potentially result in a core vulnerable condition due to a Loss-of-Flow condition when the power cables to the primary coolant pumps fail, if the reactor shutdown system does not automatically respond to the LOF condition in time to prevent exceeding clad temperature limits. In 1981, MCN-387 [10.6] provided for the replacement of the primary pump cables with 500MCM, 600 volt cables with an asbestos braid rated at 125°C. The cable runs under the deck plates in the reactor building were placed in flexible conduit, where each conduit holds the three electrical cables for the pump. However, both conduits are routed together for a portion of the routing under the operating floor deck plates next to the piping for one of the shutdown coolers. Since the fire rating of the cables is low compared to temperatures expected from a sodium or NaK fire ($\sim 900^\circ\text{F}$), the cable insulation would be expected to fail

eventually, shorting out electrical power to the primary pumps. If this were to occur, pumping power would be lost resulting in a rapid pump coastdown similar to that of opening the MG set output circuit breakers.

The postulated initiating event could also cause similar effects to the cables for the CRDM's and reactor shutdown circuitry, since these cables are routed in cable trays that are not protected, and may in fact, result in a reactor trip due to loss of power to the CRDM's. A study conducted by Factory Mutual Research Corporation in December 1987 [10.7] concluded that the only situation that would result in an unsafe condition would be a burn through of the cable to the control rods in such a manner that each of the two conductors of the circuit came in contact with conductors of an approximately equal DC power source. There are only three DC power cables in the reactor building cable trays. The possibility of this occurring to one control rod is extremely low. It would not be credible for this to occur to enough control rods at the same time to prevent safe shutdown, since normally only one or two control rods are required to shutdown the reactor and four (see Section 7) control rods are the limiting condition.

Since the deck plates are normally in place except during outages for maintenance, the effect of the sodium fire would not affect the ability to scram the reactor using the control rods, and/or safety rods. Although sodium aerosols would be released during the fire, there is no credible mechanism that could be identified to prevent the control rods from being inserted during the initial time period when the fire has been detected. Failure to achieve manual reactor shutdown following detection of the sodium fire is dominated by the human error for the operators failing to scram the reactor as required by procedure EP 1-4. However, in the event of failure of the cables to primary pumps, the reactor shutdown system would automatically respond to the loss-of-flow condition. Failure of both manual and automatic response to scram in the event of a LOF has been calculated in Section 6 to be 4.0×10^{-6} .

10.2.3.5 Limited Leak vs Rupture

The initiating frequency for liquid metal fires is based on the leak rates for the associated system piping. Due to the low pressure in the systems and in the case of shutdown coolers lack of even a pump head other than natural convection, a differentiation is made between limited leaks and rupture. A leak in a low pressure system would be a weeping type of leak. The liquid metal could oxidize as smoke if it reacted with air in the insulation or seeped out of the insulation. A rupture would release liquid metal at a sufficient rate to allow rapid oxidation and/or fire. Analysis of pipe breaks using probabilistic fracture mechanics and LWR experience [10.8] shows a leak to rupture ratio of 10:1 for 1" pipes and 1000:1 for pipes > 10". NUREG/CR-4407 [10.9] gives a leak to guillotine ratio break of 1000:1. A slightly older reference [10.10] gives a leak to rupture ratio of 30:1 for all size piping. Combining these references, a leak to rupture ratio of 99:1 was selected to give the branch point conditional probability of 1.0×10^{-2} that the initiator is a rupture rather than a pipe leak. The experience at EBR-II has been that all breaches have been leaks. A sodium fire in 1960 in the boiler building involved an inventory of approximately 80 gallons of sodium. The primary effect of this fire was sodium aerosols coating the metal surfaces of the secondary pump controls and other instrumentation and controls in the control wing of the SBB. An incident involving a small leak from a sodium-filled instrumentation line under the operating floor deck plates in the reactor building resulted in a small, self-limiting fire and smoke. No damage to other components occurred from this fire. UOR 81-4, February 22, 1981, was a leak in the radioactive chemistry loop which resulted in only instrument failure with no fire. UOR 81-8, June 19, 1981 was a leak from the secondary sodium sampling loop which was sufficient to cause a limited sodium fire which was contained. The most recent leak (UOR 89-8) was of NaK from a pressure instrument which produced only a wisp of smoke but no fire. A secondary sodium leak (UOR 90-5) from a vent valve caused an alarm indication but was so small that it was found only after hourly surveillance had been established. Thus the EBR-II operation experience supports the assumption that sodium and NaK piping failures lead to leaks and not ruptures.

10.2.3.6 Prevention of Cable Damage

A sodium leak inside of the reactor building requires the operators to perform the applicable steps of Procedure EP 3-1 [10.11] in conjunction with the previously mentioned EP 1-4 procedure. Procedure EP 3-1 instructs the operators for actions to be taken to terminate leaks in the primary purification system, RSCL or FERD systems. In addition, it cautions the plant operators that entry into the reactor building will require Scott Air-Paks and anti-C's. Radiation Safety Technician coverage is required and visibility may be impaired due to sodium aerosol and smoke. Since remote capability for suppression of a fire in this location is not provided at this time, the requirements for reactor building entry make it unlikely that manual fire suppression actions could be taken in time to prevent cable damage. Even if the fire is successfully extinguished, extensive damage to cables or other equipment and secondary sodium dump may have already occurred. Therefore, limited credit is given for the ability for early fire suppression to limit cable and instrumentation damage, and failure of suppression is estimated to be $5.0 \cdot 10^{-1}$ for pipe rupture. For pipe leak, a value of $1.0 \cdot 10^{-2}$ is used.

In the event of significant damage to cabling that is required to allow the operators to monitor the status of the reactor and locate the system which initiated the fire, it is assumed that the secondary system sodium may be dumped, precluding the use of the normal balance-of-plant system for decay heat removal. Also, if the leak is from the secondary sodium system, the secondary sodium is assumed to be dumped regardless of the status of the cabling.

10.2.3.7 Operation of Unaffected Systems

The operation of the decay heat removal systems in the event of a secondary sodium dump is dependent on the effect of aerosols generated by the liquid metal fire. The shield/thimble cooling system is considered inoperable due to the clogging of the system intake filters due to aerosols released by an unsuppressed rupture. However, short-term decay heat removal relies only on the successful operation of the remaining shutdown cooler(s). For long-term decay heat removal, fourteen days are available for recovery of the shield and thimble cooling systems by removal and/or replacement of the clogged thimble inlet filters and the

HEPA filters once the fire has been extinguished. Shutdown cooler system failure is dominated by the louvers failing to open, which is an easily recoverable event outside containment. Plant Operations indicates that the BOP could be restored for long-term decay heat removal (14-45 days) but could not be recovered within 8 hours for short-term decay heat removal if the secondary sodium and steam plant water are dumped.

10.2.3.2.2

For any leak or suppressed rupture of secondary sodium, both shutdown coolers, are initially available. One of the two shutdown coolers is required for fourteen days to assure success with a probability of $3.3 \cdot 10^{-8}$. For a rupture of the secondary sodium with an unsuppressed sodium fire, the shutdown cooler piping next to the secondary sodium piping is assumed to be affected to an extent to render that cooler inoperative. Also the effectiveness of the second cooler may be reduced due to thermal effects from the fire that may reduce natural circulation. Therefore, the probability of failure of the secondary coolers is increased from $3.3 \cdot 10^{-8}$ to a single cooler value of $1.35 \cdot 10^{-4}$ and then further increased to $1.0 \cdot 10^{-1}$ to account for possible thermal effects.

10.2.3.2.3

In the case of a suppressed shutdown cooler NaK leak, the remaining shutdown cooler is required for short-term decay heat removal. The one of one logic gives a failure probability of $1.35 \cdot 10^{-4}$. For the case of an unsuppressed rupture of shutdown cooler piping, the probability of failure of the remaining cooler is increased from $1.35 \cdot 10^{-4}$ to $1.0 \cdot 10^{-1}$ to account for possible thermal effects.

10.2.3.3

10.2.4 Analysis Results

10.2.4.1

The simplifying assumptions employed in examination of the postulated sodium fire accident scenario were conservative. Therefore, several potentially significant ($\geq 10^{-10}$) core vulnerable sequences result.

10.2.4.2

10.2.4.3

10.2.4.4

The event tree shown in Fig. 10.3 assumes the initiating event is a secondary sodium fire in containment (SSFR). The significant sequences are SSFR-6, 7, and 8. SSFR-6 is an uncontained rupture with a frequency of $1.5 \cdot 10^{-6} \text{ yr}^{-1}$. Failure of suppression is considered to be 1.0 since, for this initiating event, the operators are assumed to dump secondary sodium even if suppression is successful. Thus sequences SSFR-1 and SSFR-4 are not allowed and no credit is taken for recovery of the BOP. SSFR-7 and SSFR-8 are failure to scram or detect, respectively, with frequencies of $6.0 \cdot 10^{-9} \text{ yr}^{-1}$ and $1.5 \cdot 10^{-9} \text{ yr}^{-1}$.

The event tree as shown by Fig. 10.4 assumes the initiating event is a leak in a shutdown cooler NaK piping loop which causes a fire underneath the operating floor deck plates. For sequences SDFR-3 and SDFR-6, the fire is immediately detected and the operators scram the reactor as required by procedure but fail to manually suppress the fire before extensive cable damage occurs. The balance-of-plant heat removal is either purposely or inadvertently rendered unavailable by dumping the secondary system sodium. In the case of a leak, SDFR-3, the remaining shutdown cooler is not affected due to the small size of the fire. The sequence frequency is $5.8 \cdot 10^{-9} \text{ yr}^{-1}$. In the case of a rupture, the remaining cooler failure probability is increased from $1.35 \cdot 10^{-4}$ to $1.0 \cdot 10^{-1}$. This sequence (SDFR-6) has a frequency of $2.1 \cdot 10^{-6} \text{ yr}^{-1}$. For sequence SDFR-7, the fire is detected but the operators fail to scram the reactor. The frequency for this sequence is $1.7 \cdot 10^{-8} \text{ yr}^{-1}$. For SDFR-8, the failure to detect, the frequency is $4.3 \cdot 10^{-9} \text{ yr}^{-1}$.

The above sequences, although not likely to result in core damage, illustrate the relative importance for not being able to immediately suppress the fire upon detection. Even if decay heat removal is successful, the damage to plant cables and other non-safety equipment could preclude future operation of the plant.

10.2.5 Planned Modifications to Address Vulnerability

Modifications to reduce plant vulnerability to cable fires and sodium fires in the reactor building have been previously identified and are described in the February 1988 report, "Addendum #1 to Conceptual Design Report for Fire Safety Improvements" [10.12].

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Fig. 10.3. Event Tree Results For Secondary Sodium Fire in Containment

EVENT TREE FOR: Shutdown Cooler NaK Fire in Containment

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Shutdown Cooler NaK Fire in Containment	Liquid Metal Fire Detection	Reactor Shutdown	Limited Leak	Fire Suppression: No Secondary Sodium Dump	Systems in Other Locations Operate as Required	Sequence	Class	Fuel Damage	Frequency
SDFR	LMFD	RFIR	LLRF	SSSD	SYSO	SDFR-1			
						SDFR-2			
						SDFR-3	P2	CSD	5.8 E-9
						SDFR-4			
						SDFR-5			
						SDFR-6	P2	CSD	2.1 E-6
						SDFR-7	P3	CSD	1.7 E-8
						SDFR-8	P3	CSD	4.3 E-9

Fig. 10.4. Shutdown Cooler NaK Fire in Containment

The Fire Safety Improvements, Addendum #1 to the document W7500-0343-ES, entitled "Conceptual Design Report for Fire Safety Improvements", dated April, 1987 consists of alterations and additions to existing Argonne-West facilities that will upgrade them to meet the Department of Energy's Improved Risk Criteria, incorporate Technical Safety Appraisal items, Factory Mutual Research Corporation recommendations, and satisfy Life Safety Code recommendations.

Two of these modifications that pertain to this analysis are described below:

EBR-II Cable Protection

This project provides for protection of cables in grouped cable trays in the cable tunnel, the power plant, and reactor building. As recommended by Factory Mutual Research Corporation the protection will be accomplished by covering cable trays with solid covers top and bottom and installing fire stops at 15 ft intervals. Flammastic cable coating may also be used where covering the trays is not practical. Protecting the cables in this manner will limit the potential damage in these areas to an acceptable level.

Reactor Building - ARGON System for Under the Operation Floor Deck Plates, Bldg. 767

This project provides for the installation of a remote-manually actuated fixed pipe inert gas (argon) system which will service each of the six sectors formed by the primary tank support structure beneath the operating floor deck plates. The volume to be inerted is estimated to be 2300-cu ft. However, because of the impracticality of isolating the support structure sectors from an adjacent area in the northwest quadrant of the reactor building and the depressed area, an additional volume of 4700-cu ft will be inerted (depressed area: 3900-cu ft, northwest quadrant: 800-cu ft). Each sector will be provided with fire detectors that will alarm in the reactor building and the EBR-II control room. In the event of a fire, all of the sectors will be inerted. Because of the asphyxiating hazard of argon to personnel, actuation of the inert gas system will only be permitted after all personnel have exited the reactor building. Actuation of the inert gas system will be from the EBR-II control room. The area underneath the deck plates

contain instrumentation and control cabling and mechanical components that are essential to the operation of the EBR-II reactor. This project will minimize the potential fire damage and the impact on reactor operation.

These modifications are scheduled to be completed over the next two years. Completion of the modification for the ARGON system would significantly improve the probability for suppression of the fire since detection is highly reliable. Failure to remotely actuate the ARGON system from the EBR-II control room given detection should be no worse than RHU001CL in the RSS fault tree for failure to manually scram in the absence of a SCRAM signal, i.e., 5×10^{-2} . Reducing the probability of fire suppression failure to 5×10^{-2} from 5×10^{-1} reduces SDFR-6 sequence frequency to 2.6×10^{-7} , sequence SSFR-6 to 1.5×10^{-7} and sequence SDFR-3 to $\leq 10^{-10}$. Additional analysis, once the system has been installed, should allow a further reduction in these sequence probabilities. As stated previously, this illustrates the relative importance of the proposed modification.

10.3 Fires

In the previous section, the risk to EBR-II from sodium-related fires was examined and quantified. Non-sodium related fires also have the potential to affect multiple plant systems and must be considered as a separate initiating event. The methodology to examine plant vulnerabilities due to fires is the same as that considered previously in Section 10.2.1 for sodium fires. That is, identifying those locations which contain electrical cables for instrumentation, power, and control for multiple systems whose failure would initiate a requirement for reactor shutdown and impact plant safety by impacting the ability to: reach and maintain a condition of negative reactivity; monitor and control primary coolant system inventory; or remove decay heat. As stated previously, although an external initiating event such as a fire can impact facility operation and result in plant downtime, it is only considered as an important initiator as assessed by the PRA if it requires plant shutdown and leads to a core vulnerable condition at a frequency commensurate with those sequences initiated by internal events. It can be shown that the potential for core damage due a non-sodium fire initiator is bounded by the previous analysis for sodium fires, or by internal initiating events that have the same effect and are more frequent

initiators than fires involving safety-related equipment. Due to the low dependence on electrical power and other support systems, the worst case non-sodium fire is assumed to be a fire that disables the constant power supply to the primary pump clutches. Loss of constant power has a frequency of 0.1 per year which bounds the case for fire.

10.3.1 Plant Impacts From Fire Initiators

The most important reason for the conclusion that non-sodium fires are insignificant contributors to plant risk is that there are no credible fire initiating events that can impact the ability to remove decay heat with the shutdown cooling system following reactor shutdown. The shutdown cooling system removes heat from the primary sodium tank through two natural convection NaK loops to the atmosphere. The shutdown cooling system does not require support systems for its operation. The system does not contain any active components for system operation other than louvers which open to increase heat removal capacity. Instrumentation and control for automatic opening of the louvers on high primary bulk sodium tank temperature is not required since the louvers can be opened manually outside the reactor building, and at least 8 hours are available to take this action. While there are a number of plant locations where a fire could impact loss of the primary heat sink, i.e., the balance-of-plant system, e.g., fires associated with electrical switchgear or components in the sodium boiler building, the power plant, cooling towers, etc., a fire in these locations can not affect the ability of the plant to remove decay heat with the shutdown cooler system, and would be expected to occur much less frequently than other internal initiators that cause loss of the primary heat sink.

Similarly, there are no credible fire initiators that could affect the ability to shutdown the reactor with the control rods either automatically or manually in response to a fire. Other than the cables themselves, there are no combustibles in the reactor containment building that could ignite in the vicinity of the control rod drive mechanisms, reactor shutdown system circuitry or primary pump control cables. Since there are no ignition sources, the frequency of a cable fire starting and propagating to affect several cables is less likely than a sodium fire initiator in this area. A sodium fire due to a leak as analyzed previously also has potentially more severe consequence due the effect on multiple plant safety systems and is therefore, more limiting than

a cable fire in the reactor building. As discussed in Section 10.2.3.4 of the sodium fire analysis, a fire in a cable tray containing cables for the CRDM's and reactor shutdown circuitry would most likely result in a reactor trip due to loss of power to the CRDM's. As referenced in that section, a study conducted by Factory Mutual Research Corporation in December 1987 [10.10] concluded that the only unsafe condition would be a burn through of a cable to the control rods in such a manner that each of the two conductors of the circuit came in contact with conductors of an approximately equal DC power source, since power is required to keep the rods latched. The possibility of this occurring to one control rod is extremely low. It would not be credible for this to occur to enough control rods at the same time to prevent safe shutdown, since for normal core loadings only two control rods are required to shutdown the reactor. In the limiting case, four control rods are sufficient to shutdown the reactor.

Areas that were considered vulnerable to the effects of a fire due to the impact on operations were identified in the Factory Mutual Research Corporation study as the cable routing room, the cable tunnel, and the vertical cable tray runs between them, since a fire in one cable tray could affect all trays in that area. Nuclear system and primary sodium instrumentation important to the plant operators for control and surveillance during operation and after shutdown would be affected. If undetected, a fire in these areas would most likely lead to a loss of normal power and reactor scram, and inability to control heat removal with the BOP. However, the result would be a LOF/LOHS sequence similar to that described for LONP but much less frequent as an initiator. The shutdown cooler system would be required for decay heat removal and would not be impacted by a cable fire in these areas. Although status of the primary system and reactor may be unknown in this situation, decay heat would continue to be removed via natural circulation within the primary system and the shutdown coolers.

10.3.2 Ongoing Fire Safety Improvements

While risk to EBR-II from a fire is relatively low compared to other initiators, previous studies and appraisals conducted at the facility have resulted in a number of improvements to address fire protection. The report entitled "Addendum No. 1 To Conceptual Design Report for Fire Safety Improvements" [10.12] identified a number of modifications that

will upgrade ANL-W facilities to meet DOE's Improved Risk Criteria, incorporate items from the Technical Safety Appraisal and fire protection survey by Factory Mutual Research Corporation that were conducted in 1987, and satisfy Life Safety Code recommendations.

There are eleven subprojects which will require the utilization of wet sprinkler systems; protection of cable trays in the cable tunnel, power plant and reactor building with solid covers, fire stops and/or use of Flamastic cable coating; Halon gas, argon gas and dry chemical fire suppression systems; fire detection additions; redundant water supply systems, and diesel fire pump or a combination of these methods to establish an "improved risk" level of fire protection. The basic philosophy has been to provide automatic systems where possible for fire protection for those components required for reactor operation and increased emphasis on automatic fire suppression systems. Some of these items have already been incorporated on a priority basis, e.g., the addition of automatic wet-pipe sprinkler system in the cable tunnel and cable routing room, while others are scheduled for completion during plant outages over the next few years. The work that is already underway and scheduled will ensure that risk from fires at EBR-II continues to be maintained at a low level.

10.4 Tornadoes and High Winds

This section describes the evaluation of the hazard to EBR-II from tornadoes and high wind. Tornadoes and high winds can affect structures at the ANL-W site in two ways. If the wind forces exceed the load capacity of the structure, the walls might collapse or the structure may fall on other buildings. Also, the winds may be capable of lifting objects against the walls of structures. If a wind-propelled missile broke through a structure's wall, critical equipment inside the structure or the structure itself might be damaged [10.2].

10.4.1 Tornadoes

The INEL is in the region of the country where tornado frequency and damage potential are the lowest in the 48 contiguous states. In more than 23 years of observation at the INEL, no tornadoes have been reported within its boundaries. Two low-intensity tornadoes and

10 vortex clouds (where the funnel does not touch the ground) have been sighted within 10 miles of the INEL. A tornado was sighted near the Pocatello airport (approximately 50 miles distant from ANL-W) in August, 1990.

Studies conducted for other INEL facilities indicate that the annual probability of a tornado striking the INEL is low, 7.8×10^{-5} per year [10.14]. The Environmental Assessment for the Fuel Processing Restoration Project at the Idaho Chemical Processing Plant [10.17] referenced studies that indicated the probability of a Design Basis Tornado with a maximum windspeed of 175 mph striking that facility was estimated at 5×10^{-7} per year. A tornado with the associated windspeeds and missiles could potentially result in loss of the primary heat sink by causing failure of the secondary sodium yard piping or condenser cooling tower. Depending upon the path of the tornado, the outside containment portion of one of the shutdown coolers is also assumed to be damaged. Only one shutdown cooler is assumed to fail because they are oriented approximately 90° apart around the containment and are not susceptible to the same missile. Only a missile generated by a Design Basis Tornado could damage a shutdown cooler since the shutdown cooler chimney and boxes are constructed of 1/4-inch steel and are structurally sound, as noted in the seismic analysis in Section 9. The reactor containment building shell, consisting of a carbon steel cylinder with a hemispherical top, which is up to one inch thick and lined with reinforced concrete inside the cylindrical portion of the shell to a depth of 12 inches, would not be impacted by high winds or missiles. Components within the reactor containment building would not be impacted. Short-term decay heat removal would be dependent upon operation of the remaining shutdown cooler system. Figure 10.5 depicts the accident sequences.

The initiator probability for the tornado is 5×10^{-7} per year. Reactor shutdown is initiated manually or is assumed to occur upon loss of normal power. Short-term decay heat removal requires that the remaining shutdown cooler be operable within the 8 hour to 14 day time phase.

Quantification of the event tree for the postulated tornado scenario does not yield any significant core damage sequences as shown in Fig. 10.5. Failure either to shutdown or failure of the remaining shutdown cooler result in sequences below the 1×10^{-10} cutoff frequency.

EVENT TREE FOR: Tornado

13-JUN-91

Tornado	Reactor Shutdown	Decay Heat Removal, Short Term	Sequence	Class	Fuel Damage	Frequency
TNDO	RFIR	DSRC				
5.0 10 ⁻⁷		1.35 10 ⁻⁴	TNDO-1			
			TNDO-2	P2	CSD	6.8 E-11
			TNDO-3	P3	CSD	< 1.0 E-11

Fig. 10.5. Event Tree for Tornado

Therefore, the risk to EBR-II from an external event initiator due to a tornado is considered to be an insignificant contributor.

10.4.2 High Winds

The report "Design and Evaluation Guidelines for Department of Energy Facilities Subjected to Natural Phenomena Hazards", [10.18], provides guidance and criteria for design of new facilities and for evaluation, modification, or upgrade of existing facilities to ensure that DOE facilities can safely withstand the effects of natural phenomena. This report indicates that rather than tornadoes, the viable wind hazard to ANL-W specifically, comes from straight winds and associated gusts or missiles carried by these winds. The Design Basis Wind (DBW) for a high hazard facility (defined as those which pose a potential hazard to the safety of general public and/or the environment due to the presence of radioactive or toxic materials within the facility) such as EBR-II was determined to be 95 mph for ANL-W based upon a 10,000 year return period [10.18]. This is in general agreement with the wind speed records that have been accumulated for 32 years at the Central Facilities Area (CFA). These records indicate that peak wind gusts of 78 mph at a height of 20 ft, and 89 mph at 250 ft have occurred. The extreme values for wind speeds for a 100-year return period for INEL correlate to a 82 mph gust at 20 feet and a 91 mph gust at 250 feet [10.13].

Site Engineering for EBR-II indicates that the reinforced concrete of the sodium boiler building and steel siding/masonry block walls of the power plant were designed to resist these wind forces.

The missile criteria specified for objects or debris that could be picked up by high winds is a 2x4 inch timber plank weighing 15 lbs. Its impact speed is 50 mph at a maximum height of 50 feet above the ground. This missile is expected to perforate sheet metal siding, or wood siding up to 3/4 inch thick, but will not perforate unreinforced concrete masonry or brick veneer walls [10.18]. The safety assessment at ANL-W concluded that missiles which can cause significant damage to the facility will not be generated at windspeeds below 100 mph [10.13]. Small missiles light in weight could be generated at velocities which would penetrate a light

building shell, but these missiles would not have the energy required to do structural damage or breach containment. Therefore, damage from such a missile would be limited to the power plant or cooling tower resulting in the loss of the balance-of-plant system as the primary heat sink. Since the loss of primary heat sink as an initiator (TSEC and TBOP) is more frequent (-1/yr) than the probability of exceeding the DBW in an 10,000-year return period (10^{-4} yr), the hazard to EBR-II from high winds is bounded by the analysis for the normal Loss of Primary Heat Sink event tree.

10.5 Volcanism

The INEL is located in an area built principally by volcanic eruptions; therefore, there is a potential for resumption of volcanic activity. Volcanic processes that might affect facilities at the INEL are lava flows, earthquakes associated with volcanism, and explosive eruptions.

Volcanics on the Eastern Portion of the Snake River Plain (ESRP) consists of rhyolites and basalts. As shown by Fig. 10.6, rhyolite domes on the ESRP include Big Southern Butte, East Butte, Middle Butte and Cedar Butte. Vents that produced surface basalt flows occurred along a series of rift zones. Fig. 10.6 shows three well-defined rift zones (Arco - Big Southern Butte, Lava Ridge - Hell's Half Acre, and Circular Butte - Kettle Butte), as well as the (off-INEL) Great Rift [10.13].

Explosive rhyolitic volcanism associated with the large calderas of the ESRP has not occurred in the vicinity of the INEL for about 4-6 million years. The possibility of flows from the potentially more active Yellowstone region reaching the INEL is essentially nonexistent because of the distances involved and intervening topographic barriers. The possibility of airfall ash from such an eruption impacting the ANL-W site is also remote because the recurrence interval of the major eruptions is on the order of 0.5-1 million years and the prevailing winds have generally distributed the ash to the east or southeast of Yellowstone [10.14].

The age of several basalt flows located on, and adjacent to, the INEL have been evaluated and range from 12,000 to 400,000 years old. Just off the east boundary of the INEL,

Fig. 10.6. Volcanic Structures and Rift Zones near the INEL Site

the most recent Hell's Half Acre flow has been dated at 4100 years [10.13]. Away from the rift zones, individual areas are inundated at less frequent intervals. Studies of cores from wells at ANL-W have determined that the ANL-W site has experienced 5 flows during the past 500,000 years. The likelihood of a given eruption reaching the site is once every 80,000-100,000 years [10.14].

Earthquakes that might be expected in association with volcanic eruptions at the INEL are of two types: (1) earthquakes due to magma movement and (2) sympathetic tectonic earthquakes. Earthquakes due to magma movement are typically of low magnitude (2 to 5 Richter magnitude), and restricted to areas near the magma source. Sympathetic tectonic earthquakes are caused by forceful injection of magma into existing rift zones and as such are generally confined to the rift areas. Because of the low magnitude and localized effects from these types of earthquakes, safety-related problems due to sympathetic tectonic earthquakes and earthquakes due to magma movement are not anticipated [10.14].

The recurrence interval for lava flows would suggest that the ANL-W site is susceptible to hazards from lava flows since recurrence frequencies are in the range of 10^{-5} . Precursory events such as ground deformation, seismic activity, and fuming would likely occur near vents prior to any possible future volcanic activity on the ESRP. At that time, geologic and geophysical monitoring should help to predict the likelihood and probable location of an eruption as well as the possible threat to individual facilities at the INEL. If a particular facility like EBR-II were in danger, orderly shutdown procedures could be initiated, including removing the fuel from the reactor. Lava flow control or diversion methods could be instituted to allow six to twelve months for defueling the reactor. Therefore the likelihood of any release due to volcanic activity would be insignificant.

10.6 External Floods

The INEL is situated on a high arid plain with flat or gently rolling terrain. These characteristics, combined with the low precipitation and the permeable soils that are prevalent, result in little or no concern for external hazards from flooding. The waters of the Big and

Little Lost Rivers and Birch Creek drain through the Eastern Snake River Plain to sinks in the vicinity of the Test Area North (TAN), significantly contributing to the Snake River Aquifer. The Big Lost River, as the most significant water source, has two major flood control features, the Mackay Dam and the INEL flood-diversion system. The INEL flood-control system reduces the threat of floods at INEL from any abnormally large stream flows in the Big Lost River, by diverting stream flow out of the main channel to INEL spreading areas.

There is no history of flooding at the ANL-W site. The afore-mentioned rivers do not constitute a flood threat to the EBR-II facility since the elevation above mean sea level (MSL) of ANL-W is approximately 5120 ft. At its closest approach to the site, the Big Lost River is 11 miles away at a lower elevation of 4850 ft MSL. This elevation differential (+270 ft higher than the river channel) and distance from the river protect the site from even a worst case scenario involving complete failure of the Mackay Dam due to a seismic event [10.13].

10.7 Lightning

Air-to-ground lightning strikes occur infrequently at the INEL. However, the lack of natural targets and the poor conductivity of the lava rock and desert soil cause manmade structures at the INEL to be highly susceptible to lightning strikes. Therefore, lightning protection systems are required for structures that could be susceptible to damage from lightning.

On two separate occasions, the EBR-II plant has tripped due to lightning strikes. While at power in June of 1977, a lightning strike caused both 2400V and 480V transformer secondary circuit breakers to open and the 13.8 kV incoming and feeder breakers to open. Voltage and current transients associated with the lightning strike also caused the constant uninterruptible power system (UPS) units No. 1 and No. 2 to transfer to the 480V AC bypass power which was failed since the 480V power line had already failed. This resulted in an automatic reactor scram and loss of power to the reactor control room. However, the emergency diesel generators started and assumed the loads. It was later determined that significant ground currents were flowing in the power plant's grounding network and substantial overcurrent flowed through the UPS inverter and control logic as a result of the lightning surge. A later incident occurred in

1984 as power was lost to the main feedwater pumps when a EBR-II 13.8 kV incoming line breaker failed to open upon an electric power system transient due to lightning. The reactor was manually scrammed after the operators received a steam drum low-level alarm.

Lightning rods above the power plant stack and weather tower, as well as for the metal surfaces of the power plant and sodium boiler buildings, are connected by large grounding cables to the site ground system. It was determined by plant engineering that lightning strikes which hit these rods or structures could eventually reach the grounding system and create disturbances similar to those observed during the June 1977 incident. Therefore, Engineering Change Notice 161 and 161B [10.15] provided for the installation of surge suppression devices on the input and bypass power lines for continuous power UPS No. 1 and 2, and the DAS UPS. The continuous power system provides 120V AC power to critical instruments and components throughout EBR-II. The surge suppressors are installed on the 480V AC input power feeders for the UPS units. These surge suppressors prevent damage to the UPS components due to spikes on the 480V AC distribution system, caused primarily by lightning strikes on the INEL loop transmission lines.

Even with the design change, lightning strikes can still result in the requirement for reactor shutdown, either automatically or manually, due to the effect of loss of normal power to operating components, and therefore, contribute to the frequency for the Loss of Normal Power initiating event. However, since the modification ensures reliable power for the nuclear and process instrumentation and control equipment in the event of a lightning strike, and the shutdown coolers do not require power to remove decay heat, any accident sequences initiated by lightning are bounded by the evaluation for LONP, in Section 11.

10.8 Aircraft Impact

One possible threat to EBR-II is from accidents associated with aircraft. There are five commercial airports located within 100 miles of INEL: Pocatello (50 miles), Idaho Falls (29 miles), Twin Falls (80 miles), Hailey (70 miles), and Jackson, Wyoming (85 miles).

Several smaller, gravel-surface landing strips near INEL are used for emergency landings, charter flights, crop dusting, and other such intermittent activities. The closest such landing strip to ANL-W is at Atomic City, just outside the south boundary of INEL. Only one large commercial airline flight (Salt Lake to Calgary) is known to over-fly the INEL routinely, although several commuter flights (Idaho Falls to Boise) may be routed to traverse the southern portion of the INEL. Aircraft are warned away by high-intensity strobe lights on ANL-W facilities (such as the 240-ft meteorological tower). The location of these lights and other restrictions are designated on current aeronautical charts. Increased security surveillance at all INEL sites, and the need to upgrade security forces, resulted in helicopter pads being installed at most INEL sites. The ANL-W helipad is just south of the main parking lot. Routine helicopter overflights over ANL-W are not allowed [10.13].

Because overflying aircraft activity is very limited and airports are relatively distant from the site, hazards to EBR-II due to aircraft are not a significant threat. An upper bound estimate on the total frequency of aircraft impact hazards for EBR-II can be estimated based upon criteria established in the U.S. NRC Standard Review Plan (SRP), Section 3.5.1.6, which addresses the frequency of aircraft crashing into a nuclear plant [10.16].

According to the guidelines in the SRP, the crash frequency at a nuclear plant for aircraft using a federal airway can be estimated from the expression:

$$P_{FA} = C * N * A/W$$

where

P_{FA} = the frequency of aircraft crashing into a nuclear power plant structure (crashes/year),

C = the inflight crash rate for aircraft using the airway (crashes/aircraft-mile),

N = the annual number of flights along that segment of the airway (aircraft/year),

A = the effective area of the plant, i.e., the area that includes all plant structures which, if struck, could have an adverse affect on safety (square miles), and

W = the width of the federal airway (miles).

The SRP recommends using an inflight crash rate of 4.0×10^{-10} crashes per aircraft-mile. The Federal Aviation Administration, Idaho Falls, projects that 2920 aircraft per year operating under instrument flight rules (IFR) use a fixed route between Salt Lake City and Calgary, Canada which passes over the INEL. There are also approximately 100 Vortex Test Flights conducted by the NOAA-FAA at the site per year. Additionally, visual flight rules (VFR) smaller aircraft, while restricted to maintaining 8000' MSL elevation, are allowed to pass over the INEL. The FAA estimates there are approximately 100 such operations per month, giving a total number of flights of 4220. The effective area of the EBR-II structures is estimated to be 0.01 square miles. The width of the fixed airway route was estimated by the FAA to be approximately 2 miles. All of these operations are assumed to take place within this fixed route. Therefore, the estimated frequency of an aircraft flying this route and crashing into EBR-II is approximately 8.4×10^{-9} crashes per year. Since this evaluation only considers the frequency of aircraft impacts without investigating the capabilities of the EBR-II plant to withstand crashes, it can be assumed that this hazard poses a negligible risk to EBR-II.

10.9 Accidents at Other INEL Facilities (Industrial and Military)

Activities or accidents at other INEL facilities do not pose a physical hazard to ANL-W since the nearest facility is nine miles distant. However, airborne release of radioactive materials during emergency conditions at these other facilities could conceivably pose a hazard to ANL-W site personnel. The DOE Warning Communications Center (WCC) in Idaho Falls maintains continuous site-wide surveillance of all INEL facilities, and transmits site-wide warnings of any unsafe conditions. Upon receipt of such warnings, the ANL-W Emergency Plan would be activated. Operations at EBR-II can be safely terminated, and the personnel evacuated almost immediately [10.13].

10.10 Turbine Missiles

An analysis of external events often includes an examination of the impact of a missile from the turbine-generator striking a critical plant component. The postulated missile is a result of mechanical components being hurled at high velocity from self-destruction and breakup of the

turbine-generator during plant operation. The failure mechanism for such an initiator is based upon either metallurgical and/or design errors, environmental effects such as stress corrosion, overspeed, or a combination of these factors. The probability of significant damage to critical nuclear reactor components due to turbine failure has been assessed in previous studies of commercial reactors by determining the combined probabilities of turbine failure and ejection of an energetic missile that escapes the turbine casing; the probability the missile strikes a critical component, and the probability that significant damage occurs to the component [10.19]. An example of such a scenario would be a missile that strikes the reactor containment, and penetrates and damages key components in the primary coolant system. Studies conducted for Westinghouse and General Electric LWRs indicate the overall probability of serious consequences from such an initiator is very low, on the range of 10^{-10} to 10^{-12} per year, and is even further reduced if the turbine axis is oriented on a radius of the containment, since missile direction would be radial about the axis of the turbine [10.19].

The turbine generator for EBR-II is located on the turbine deck floor of the power plant, as shown previously by Figure 3.8 in Section 3 of this report. The positioning of the turbine-generator relative to the reactor building precludes missile damage to the containment or the shutdown coolers. It would be expected that balance-of-plant heat removal would be impacted requiring reactor shutdown but the ability to remove decay heat with the shutdown coolers would not be affected. However, since the EBR-II main control room is adjacent to the turbine, a more immediate concern would be missiles that could strike the main control room since there are no barriers to such a missile other than the concrete wall providing separation. Control room personnel and equipment would be vulnerable to the effects of such an event. The probability of a turbine generator missile initiator was determined to be in the order of $3.1 \cdot 10^{-4}$ to $3.5 \cdot 10^{-5}$ per turbine year for failures observed in turbine population (nuclear as well as non-nuclear) corrected to be relevant to nuclear reactors [10.20]. The previously referenced study indicated that the probability of striking a critical target is approximately $1.0 \cdot 10^{-3}$. This would result in a worst-case probability on the order of $3.1 \cdot 10^{-7}/\text{yr}$ for a turbine missile striking the control room. The effect of such a missile upon the ability to achieve reactor scram would require further analysis, but the engineering judgement of plant operations indicate that most likely, either due to the initiator effect on the electrical grid or the loss of control room

instrumentation or control power by the missile would result in an automatic reactor scram, and that manual scram would not necessarily be precluded, resulting in safe shutdown and decay heat removal via the shutdown coolers. Therefore, turbine missiles are considered to be an insignificant contributor to the overall frequency of core damage at EBR-II, but do pose a risk to plant personnel should they occur.

10.11 Internal Floods

The reactor containment building does not contain any water sources such as tanks or piping. Water sources are purposely excluded in the reactor building to prevent any possibility of a sodium-water reaction from occurring due to a spill or accident. An internal flood initiator could occur in the power plant since the balance-of-plant power conversion cycle involves the condensate and feedwater systems and the condenser and associated cooling water systems. An internal flood initiator involving these systems would only serve to disrupt normal operations by disabling balance-of-plant heat removal, requiring plant shutdown. There are no credible internal flood events that can impact the ability to remove decay heat with the shutdown cooling system following reactor shutdown. Plant shutdown with the primary heat sink unavailable would be expected to occur more frequently due to other internal initiators, e.g., loss of feedwater flow, than the frequency for a flood-induced event, and therefore, would not significantly contribute to plant risk.

There is a potential for water accumulation in the cable tunnel between the main control room and the reactor building due to groundwater seepage or inadvertent actuation of the wet pipe fire protection sprinkler system that has been recently installed. The most likely consequence of water accumulation in the cable tunnel would be an inadvertent reactor scram due to electrical shorts of seismic detectors located on the floor of the cable tunnel. The cables would not be affected by water immersion or spray since they are sealed and routed in cable trays a minimum of two to three feet above the floor of the tunnel. Cable connections may be susceptible to immersion but are located a minimum of 4-5 feet above the floor. The cable tunnel itself has sumps in the event of water accumulation with sump pumps that automatically start to transfer water out of the sumps. Operation of the sump pumps is annunciated in the

main control room. Inadvertent operation of the wet pipe fire protection system would result in a trouble alarm code at the site fire department indicating water delivery flow in the cable tunnel. Operations personnel would be dispatched to determine the cause for the alarm and water flow would be terminated.

Inadvertent actuation of the wet pipe fire protection sprinkler system in the remainder of the plant other than the BOP could impact the 2400 VAC switchgear and the primary pump clutch control panel. The spray shields above the 2400 VAC switchgear are designed to prevent water intrusion. The failure rate of sprinkler heads for inadvertent actuation is calculated based on three environmental failures (high temperature due to steam) and five due to maintenance in a population of 60,000 for 30 years [10.23]. The calculated value is $4.7 \cdot 10^{-6} \text{ yr}^{-1}$. Even if several sprinklers could effect the switchgear, the LONP resulting would be much smaller than the value of 10^{-1} from other causes. Therefore, this event is subsumed into LONP.

Water in primary pump clutch control could cause a loss of both primary pumps due to loss of power to the clutches, resulting in a LOF condition. Since there is no spray shield on the clutch control cabinet, the initiating frequency is assumed to be the same as the failure of the single sprinkler in the area, $4.7 \cdot 10^{-6} \text{ yr}^{-1}$. This is three orders of magnitude below the LF2B internal initiating event frequency of $4.7 \cdot 10^{-3} \text{ yr}^{-1}$. The LF2B conditional probabilities of core and structural damage (CSD) and core damage (CD) are $8.3 \cdot 10^{-9}$ and $4.6 \cdot 10^{-6}$, respectively. Therefore the core damage frequency for inadvertent sprinkler actuation is below the 10^{-10} threshold.

10.12 Summary

Of the external events considered in this section, only liquid metal fires have sequence frequencies which are above the 10^{-10} screening cutoff. The total frequency for secondary sodium fire damage of any type is $1.5 \cdot 10^{-6} \text{ yr}^{-1}$. The total frequency for shutdown cooler NaK fire is $2.1 \cdot 10^{-6} \text{ yr}^{-1}$. Detailed results for all sequences are given in Table 10.2. All other sodium fires have sequences which do not lead to core damage or which fall significantly below the screening cutoff.

Table 10.2 Summary of External Event Initiating Events and Sequences

<u>External Event</u>	<u>Initiating Event Frequency</u>	<u>Sequence</u>	<u>Class</u>	<u>Fuel Damage</u>	<u>Frequency</u>
Fires					
BFTF					
FERD					
Primary Sodium Purification Out of Cell [1]					
Secondary Sodium Within Containment	7.2 10 ⁻⁴	SSFR-3	P2	CSD	< 10 ⁻¹⁰
	9.1 10 ⁻⁴	SSFR-6	P2	CSD	1.5 10 ⁻⁶
	2.2 10 ⁻³	SSFR-7	P3	CSD	
	1.5 10 ⁻³	SSFT-8	P3	CSD	
Shutdown Coolers					
	4.3 10 ⁻³	SDFR-3	P2	CSD	5.8 10 ⁻⁹
		SDFR-6	P2	CSD	2.1 10 ⁻⁶
		SDFR-7	P3	CSD	1.7 10 ⁻⁸
		SDFR-8	P3	CSD	4.3 10 ⁻⁹
Tornado					
	5.0 10 ⁻⁷	TNDO-2	P2	CSD	< 10 ⁻¹⁰
		TNDO-3	P3	CSD	< 10 ⁻¹⁰
High Wind [2]	1.0 10 ⁻⁴				
Volcanism	1.0 10 ⁻⁵				
any	< 1.0 10 ⁻⁸				
affecting EBR-II	< 1.0 10 ⁻⁸				
External Floods					
Lightning	[3] 8.2 10 ⁻⁹				
Aircraft Impact	3.1 10 ⁻⁴ to				
Turbine Missiles	3.5 10 ⁻⁵				
					< 10 ⁻¹⁰ [5]
Internal Floods					
	< 10 ⁻¹⁰ [6]		P3	CD	< 10 ⁻¹⁰

Notes:

- [1] limited by syphon break system
- [2] bounded by TSEC, TBOP and LONP
- [3] bounded by LONP
- [4] reference 10.19
- [5] personnel risk; not core or plant damage
- [6] damage from unprotected loss of primary pump clutches, remainder of sequences bounded by LOHS

The only other sequence which is above the 10^{-10} cutoff is turbine missiles. Depending on the reference, a value of $3.1 \times 10^{-7} \text{ yr}^{-1}$ is found but this value is limited to personnel injury only. A lower value is assumed for equipment damage. Another reference for equipment damage [10.19] gives a value below the cutoff of 10^{-10} .

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11.0 SEQUENCE QUANTIFICATION

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11. SEQUENCE QUANTIFICATION

11.1. Introduction

The accident sequence quantification task consists in integrating the models developed for plant systems, the models developed for accident progression, and the data compiled for the basic events and the initiating events. The frequency of an accident sequence is calculated by determining the dominant minimal cutsets of the sequence and evaluating their frequency using the initiator frequency and the probabilities of the basic events. The sum of the frequencies of the minimal cutsets is the point estimate of the sequence frequency, and the sum of the sequence frequencies is the point estimate of the overall frequency of core damage.

The products of the accident sequence analysis are the estimated frequencies of each sequence, a list of the sequences and their relative contribution to the overall damage frequency, and a list of the most significant minimal cutsets and basic events contributing to the dominant sequences. The results reported in this section refer only to the sequences from internal initiating events. Although uncertainty coupling was studied in some systems, the calculation for accident sequences have been performed without correction for uncertainty coupling among basic event probability distributions.

The process of quantifying accident sequences, using the event tree and fault tree models developed, in practice becomes a complex task when all the system dependencies and interconnections are taken into account. The most important aspects to be considered during sequence quantifications are:

- Effects of the initiating event on the different plant systems and their response.

- Logical loops that appear when a system depends upon a support system that in turn depends upon the original system.
- Effect of linking faulted and successful systems in a sequence, with the possible appearance of incompatibilities, such as minimal cutsets in the faulted systems that are supersets of those of the systems that have not failed.
- Dependencies and commonalities between systems, such as failure of a common support system or the failure of a shared component.
- Inclusion of recovery actions.

11.2 Sequence Quantification Methodology

With the exception of two direct damage initiators, the accident sequences in EBR-II are formed by a combination of an initiating event and one or more system failures and/or successes. Following the criteria developed in Section 5, the accident sequences leading to some sort of fuel damage in EBR-II have been selected and analyzed. Table 11.1 provides a listing of all the sequences that have been quantified, grouped by fuel damage category. It lists the potential sequences leading to fuel damage (CD, CSD, MCD, PED) plus those sequences that, although not resulting in fuel damage, have the potential for major plant disruption (ND). For each sequence listed in the tables, the failed and success top events are identified by the 4-letter acronyms that have been used in the event trees. Table 11.2 provides a description of these event tree top events. The initiating event has not been explicitly listed under the failed top events because it is part of the sequence identification, which also provides direct correspondence between the sequences and the event trees of Section 5.

When all the relevant sequences had been identified and properly described, the quantification process began by building the composite fault trees. This task consisted primarily in including the support system trees in the main system trees, in particular the decay heat

Table 11.1 Accident Sequences Analyzed, Listed by Damage Category

DAMAGE CLASS: CD

SEQUENCE	FAILED TOP EVENTS:	SUCCESSFUL TOP EVENTS:
LCDL-4	PKRU	
LF2B-6	FROD	FSIG
LF2B-7	FSIG	
LF2D-2	FROD	FSIG
LF2D-3	FSIG	
LF2E-6	FROD	FSIG
LF2E-7	FSIG	
LOCP-6	LROD	
MDLF-4	NPPD SLPD NSSP	DETP SHTS
MDLF-6	NPPD SLPD SHTS NSSP	DETP
MDLF-8	NPPD SLPD DETP NSSP	
MELF-4	NPPE SLPE NSSP	DETP SHTS
MELF-6	NPPE SLPE SHTS NSSP	DETP
MELF-8	NPPE SLPE DETP NSSP	
OELF-4	NPPO SLPE NSSP	DETP SHTP
OELF-6	NPPO SLPE SHTP NSSP	DETP
OELF-8	NPPO SLPE DETP NSSP	
RIFL-1		
RIFS-7	RROD PRUN	RSIG
RISA-7	RROD PRUN	RSIG
RISB-7	RROD PRUN	RSIG
RISC-6	RROD	RSIG PRUN
RISC-7	RROD PRUN	RSIG
RISC-8	RSIG	
RISD-6	RROD	RSIG PRUN
RISD-7	RROD PRUN	RSIG
RISD-8	RSIG	
SFLE-5	SNPB NPPO SLPE NSSS	DETP SHTS
SFLE-7	SNPB NPPO SLPE SHTS NSSS	DETP
SFLE-9	SNPB NPPO SLPE DETP NSSS	
SFPB-5	SNPB NPPB SLPE NSSS	DETS SHTS
SFPB-7	SNPB NPPB SLPE SHTS NSSS	DETS
SFPB-9	SNPB NPPB SLPE DETS NSSS	

Table 11.1 Accident Sequences Analyzed, Listed by Damage Category (Cont'd)

DAMAGE CLASS: MCD

SEQUENCE	FAILED TOP EVENTS:	SUCCESSFUL TOP EVENTS:
LF1B-7	FROD PRUN	FSIG
LF1C-7	FROD PRUN	FSIG
LF1D-7	FROD PRUN	FSIG
LF2C-6	FROD	FSIG
LF2C-7	FSIG	
LF2D-1		FSIG FROD
LF2F-6	FROD	FSIG
LF2F-7	FSIG	
MDLF-5	NPPD SLPD SHTS	DETP NSSP
MDLF-7	NPPD SLPD DETP	NSSP
MELF-5	NPPE SLPE SHTS	DETP NSSP
MELF-7	NPPE SLPE DETP	NSSP
OCSL-9	RROD SPTR	RSIG
OCSL-10	RSIG	
OELF-5	NPPO SLPE SHTP	DETP NSSP
OELF-7	NPPO SLPE DETP	NSSP
RIFS-6	RROD	RSIG PRUN
RIFS-8	RSIG	
RISB-6	RROD	RSIG PRUN
RISB-8	RSIG	
SFLE-6	SNPB NPPO SLPE SHTS	DETP NSSS
SFLE-8	SNPB NPPO SLPE DETP	NSSS
SFPB-6	SNPB NPPB SLPE SHTS	DETS NSSS
SFPB-8	SNPB NPPB SLPE DETS	NSSS

Table 11.1 Accident Sequences Analyzed, Listed by Damage Category (Cont'd)

DAMAGE CLASS: CSD

SEQUENCE	FAILED TOP EVENTS:	SUCCESSFUL TOP EVENTS:
ANTC-3	BPHR DHRS	
EVTR-4	SPRS DUME RIHX	FSDG SMLK LKDT
EVTR-6	LKDT SPRS	FSDG SMLK
EVTR-10	SMLK SPRS PRRE RIHX	FSDG
EVTR-13	FSDG SPRS PRRE	
LCDL-2	DHRL	PKRU DHRS
LCDL-3	DHRS	PKRU
LF1A-2	DHRL	FSIG FROD BPHR
LF1A-4	BPHR DHRL	FSIG FROD DHRS
LF1A-5	BPHR DHRS	FSIG FROD
LF1B-2	DHRL	FSIG FROD BPHR
LF1B-4	BPHR DHRL	FSIG FROD DHRS
LF1B-5	BPHR DHRS	FSIG FROD
LF1C-2	DHRL	FSIG FROD BPHR
LF1C-4	BPHR DHRL	FSIG FROD DHRS
LF1C-5	BPHR DHRS	FSIG FROD
LF1D-2	DHRL	FSIG FROD BPHR
LF1D-4	BPHR DHRL	FSIG FROD DHRS
LF1D-5	BPHR DHRS	FSIG FROD
LF2A-2	DHRL	FSIG FROD BPHR
LF2A-4	BPHR DHRL	FSIG FROD DHRS
LF2A-5	BPHR DHRS	FSIG FROD
LF2B-2	DHRL	FSIG FROD BPHR
LF2B-4	BPHR DHRL	FSIG FROD DHRS
LF2B-5	BPHR DHRS	FSIG FROD
LF2C-2	DHRL	FSIG FROD BPHR
LF2C-4	BPHR DHRL	FSIG FROD DHRS
LF2C-5	BPHR DHRS	FSIG FROD
LF2E-2	DHRL	FSIG FROD BPHR
LF2E-4	BPHR DHRL	FSIG FROD DHRS
LF2E-5	BPHR DHRS	FSIG FROD
LF2F-2	DHRL	FSIG FROD BPHR
LF2F-4	BPHR DHRL	FSIG FROD DHRS
LF2F-5	BPHR DHRS	FSIG FROD
LOCP-2	DHRL	LROD BPHR
LOCP-4	BPHR DHRL	LROD DHRS

Table 11.1 Accident Sequences Analyzed, Listed by Damage Category (Cont'd)

DAMAGE CLASS: CSD (cont'd)

SEQUENCE	FAILED TOP EVENTS:	SUCCESSFUL TOP EVENTS:
LOCP-5	BPHR DHRS	LROD
LONP-2	DHRL	FSIG FROD DHRS
LONP-3	DHRS	FSIG FROD
OCPP-2	DHRL	RSIG RROD BPHR
OCPP-4	BPHR DHRL	RSIG RROD DHRS
OCPP-5	BPHR DHRS	RSIG RROD
OCPP-7	RROD DHRL	RSIG SPTR DHRS
OCPP-8	RROD DHRS	RSIG SPTR
OCSL-2	DHRL	RSIG RROD BPHR
OCSL-4	BPHR DHRL	RSIG RROD DHRS
OCSL-5	BPHR DHRS	RSIG RROD
OCSL-7	RROD DHRL	RSIG SPTR DHRS
OCSL-8	RROD DHRS	RSIG SPTR
OCSP-2	DHRL	RSIG RROD BPHR
OCSP-4	BPHR DHRL	RSIG RROD DHRS
OCSP-5	BPHR DHRS	RSIG RROD
OCSP-7	RROD DHRL	RSIG SPTR DHRS
OCSP-8	RROD DHRS	RSIG SPTR
RIFS-2	DHRL	RSIG RROD BPHR
RIFS-4	BPHR DHRL	RSIG RROD DHRS
RIFS-5	BPHR DHRS	RSIG RROD
RISA-2	DHRL	RSIG RROD BPHR
RISA-4	BPHR DHRL	RSIG RROD DHRS
RISA-5	BPHR DHRS	RSIG RROD
RISB-2	DHRL	RSIG RROD BPHR
RISB-4	BPHR DHRL	RSIG RROD DHRS
RISB-5	BPHR DHRS	RSIG RROD
RISC-2	DHRL	RSIG RROD BPHR
RISC-4	BPHR DHRL	RSIG RROD DHRS
RISC-5	BPHR DHRS	RSIG RROD
RISD-2	DHRL	RSIG RROD BPHR
RISD-4	BPHR DHRL	RSIG RROD DHRS
RISD-5	BPHR DHRS	RSIG RROD
SCRM-2	DHRL	BPHR
SCRM-4	BPHR DHRL	DHRS
SCRM-5	BPHR DHRS	

Table 11.1 Accident Sequences Analyzed, Listed by Damage Category (Cont'd)

DAMAGE CLASS: CSD (cont'd)

SEQUENCE	FAILED TOP EVENTS:	SUCCESSFUL TOP EVENTS:
SHDL-2	DHRL	DHRS
SHDL-3	DHRS	
SHDS-3	BPHR DHRS	
SPTR-3	DUMS RIHX	FSDG MODS LKDT SPRS
SPTR-6	SPRS DUMS RIHX	FSDG MODS LKDT
SPTR-9	LKDT DUMS RIHX	FSDG MODS SPRS
SPTR-10	LKDT SPRS	FSDG MODS
SPTR-13	MODS PRRS RIHX	FSDG LKDT SPRS
SPTR-16	MODS SPRS PRRS RIHX	FSDG LKDT
SPTR-19	MODS LKDT PRRS RIHX	FSDG SPRS
SPTR-21	MODS LKDT SPRS PRRS	FSDG
SPTR-23	FSDG PRRS	
TAIR-2	DHRL	MRSS DHRS
TAIR-3	DHRS	MRSS
TBOP-2	DHRL	MRSS DHRS
TBOP-3	DHRS	MRSS
TSDC-2	DRLA	BPHR
TSDC-4	BPHR DRLA	DRSC
TSDC-5	BPHR DRSC	
TSEC-2	DHRL	MRSS DHRS
TSEC-3	DHRS	MRSS
TSHC-2	DRLB	BPHR
TSHC-4	BPHR DRLB	DRSB
TSHC-5	BPHR DRSB	

Table 11.1 Accident Sequences Analyzed, Listed by Damage Category (Cont'd)

DAMAGE CLASS: PED

SEQUENCE	FAILED TOP EVENTS:	SUCCESSFUL TOP EVENTS:
LCDL-1		PKRU DHRS DHRL
LF1A-7	FROD PRUN	FSIG
LF2A-6	FROD	FSIG
LF2A-7	FSIG	
LONP-4	FROD	FSIG
LONP-5	FSIG	

DAMAGE CLASS: ND

SEQUENCE	FAILED TOP EVENTS:	SUCCESSFUL TOP EVENTS:
EVTR-3	SPRS DUME	FSDG SMLK LKDT RIHX.
EVTR-9	SMLK SPRS PRRE	FSDG RIHX
LF1A-6	FROD	FSIG PRUN
LF1A-8	FSIG	
LF1B-6	FROD	FSIG PRUN
LF1B-8	FSIG	
LF1C-6	FROD	FSIG PRUN
LF1C-8	FSIG	
LF1D-6	FROD	FSIG PRUN
LF1D-8	FSIG	
OCPP-9	RROD SPTR	RSIG
OCPP-10	RSIG	
OCSP-9	RROD SPTR	RSIG
OCSP-10	RSIG	
RISA-6	RROD	RSIG PRUN
RISA-8	RSIG	
SPTR-2	DUMS	FSDG MODS LKDT SPRS RIHX
SPTR-5	SPRS DUMS	FSDG MODS LKDT RIHX
SPTR-8	LKDT DUMS	FSDG MODS SPRS RIHX
SPTR-12	MODS PRRS	FSDG LKDT SPRS RIHX
SPTR-15	MODS SPRS PRRS	FSDG LKDT RIHX
SPTR-18	MODS LKDT PRRS	FSDG SPRS RIHX
TAIR-4	MRSS	
TBOP-4	MRSS	
TSEC-4	MRSS	

Table 11.2 Description of the Event Tree Top Events

TOP EVENT	DESCRIPTION
BPHR	Balance of plant decay heat removal
DBOP	Balance of Plant (Sec Na & Steam)
DETP	Detect in Time to Shutdown and Prevent Propagation
DETS	Detect in Time to Shutdown and Prevent Propagation
DHRL	Decay heat removal long term
DHRS	Decay heat removal short term
DRLA	Decay heat removal long term (1 sd cooler + shield cooling)
DRLB	Decay heat removal long term (2 shutdown coolers)
DRSB	Decay heat removal short term (2 shutdown coolers)
DRSC	Decay heat removal short term (1 sd cooler)
DUME	Water and Sodium Dump Before Leak Escalation
DUMS	Water and Sodium Dump Before Leak Escalation
FROD	Sufficient control or safety rods drop during LOF event
FSDG	Less than 2 DEG Failures within First 0.2 Seconds
FSIG	Scram signal during LOF event
LKDT	Leak Detection by HMLDs
LROD	Sufficient control or safety rods drop during LOF event
MODS	Leak Escalates within First Minute
MRSS	Manual SCRAM Under LOHS Conditions
NPPB	No Pin-to-Pin Propagation
NPPD	No Pin-to-Pin Propagation
NPPE	No Pin-to-Pin Propagation
NPPO	No Pin-to-Pin Propagation
NSSP	No Subassembly-to- Subassembly Propagation
NSSS	No Subassembly-to- Subassembly Propagation
PKRU	Pumps keep running for 1 hour
PRRE	Pressure Relief by Secondary Boundary Failure
PRRS	Pressure Relief by Secondary Boundary Failure
PRUN	Primary pumps keep running
RIHX	Reactor Shutdown Before IHX Failure
RROD	Sufficient control or safety rods drop during TOP event
RSIG	Scram signal during TOP event
SHTP	Shutdown of Reactor
SHTS	Shutdown of Reactor
SLPD	Self-Limiting Pin Propagation
SLPE	Self-Limiting Pin Propagation
SMLK	Small or Medium Leak
SNPB	No Pin Breach
SPRS	Pressure Relief System Activates
SPTR	Secondary pump trips

removal trees. A logic loop had to be broken in the electrical support system trees. The next step was the generation of the minimal cutsets for each sequence. Because the balance of plant (BPHR) and the decay heat removal trees (DHRS and DHRL) are the largest in terms of minimal cutsets and appear in many different sequences, an intermediate sequence evaluation (qualitative determination of minimal cutsets only) was performed for the heat removal sequences. The heat removal sequences evaluated are depicted in the event tree of Fig. 11.1. Only three of the intermediate sequences were analyzed (DKHR-2, DKHR-4, DKHR-5), since the other two lead to successful decay heat removal. The minimal cutsets for the three relevant partial sequences were found and then used in the calculations for the complete accident sequences.

A fault tree linking method was used in generating the minimal cutsets of the accident sequences. Logically inconsistent minimal cutsets, those sequence cutsets of failure trees that were also supersets of a minimal cutset of the success trees, were automatically eliminated. The cutset generation was truncated by the use of a probability cutoff. The probability cutoff was selected at 10^{-15} for all the sequence calculations except for those of the local faults and evaporator and superheater tube rupture event trees, where no probability cutoff was used at all. The cutsets of the fault trees involved in the accident sequences had previously been generated with a probability cutoff of 10^{-11} or 10^{-12} , depending on the top event probability and the computation times involved. After the minimal cutsets were determined the frequency was estimated. Importance and uncertainty calculations were also performed for each sequence, as reported in the next section. For two initiators (loss of piping integrity, LOPI, and rapid loss of circulation, LCRA) the frequency of the initiating events was less than $1.0 \times 10^{-11} \text{ yr}^{-1}$, and their sequences were not quantified.

11.3 Sequence Quantification Results

The accident sequence frequency values reported in this section are point mean values, estimated using only mean values for the initiator frequencies and the basic event

EVENT TREE FOR: Decay heat removal sub-tree

01-JUN-91

Decay heat removal sub-tree	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
DKHR	BPHR	DHRS	DHRL				
				DKHR-1			
				DKHR-2	P2	CSD	
				DKHR-3			
				DKHR-4	P2	CSD	
				DKHR-5	P2	CSD	

Fig. 11.1. Decay Heat Removal Sub-Tree

probabilities. Section 12, on the other hand, reports the frequency distributions. Depending on the type of transient and damage produced, the accident sequences have been classified in different bins, as defined in Section 5.

There are sequences that lead to some form of fuel damage or core and structural damage, grouped under classes CD, MCD, and CSD, sequences that lead only to potential experiment damage, assembled under PED, and finally sequences that, without causing damage to the core, are severe or disruptive enough as to be included in the frequency evaluation (collected under class ND). Within the ND class, a clear distinction can be made among sequences belonging to two different transient types. Sequences in the ATWS (P3) and ND class represent the unprotected LOF, TOP, and LOHS transients that result in passive shutdown and heat removal in EBR-II. The ND sequences in class P5 (evaporator or superheater tube rupture) are those that, although a safe shutdown is achieved, lead to potential severe damage or contamination of the IHX.

The following subsections summarize the annual frequency estimates for the accident sequences. The sequence annual frequencies are compared on the bases of their relative contribution to the different types of transient and to the different types of damage. When the frequency results are presented, the differentiation between damage and total annual frequency is made; damage sequences refer to those sequences that lead to either fuel or structural damage or potential experiment damage (classes CD, MCD, CSD, and PED). Similarly, the damage frequency is the annual frequency of classes CD, MCD, CSD, and PED combined, while total frequency refers to the annual damage frequency plus the frequency of the ND class.

11.3.1 Results by Accident Sequence

The dominant sequences have been defined as those with a point mean frequency above $1.0 \cdot 10^{-11} \text{ yr}^{-1}$. The dominant sequences leading to fuel or experiment damage are listed in Table 11.3 and the contribution of the most important sequences is graphically shown in Fig. 11.2. The first 10 sequences account for over 82 % of

Table 11.3 Summary of Dominant Sequences Contributing to Fuel and Experiment Damage

TOTAL DAMAGE: $1.57 \cdot 10^{-6} \text{ yr}^{-1}$		
ACCIDENT SEQUENCE	FREQUENCY yr^{-1}	PERCENTAGE %
LONP-4	$4.55 \cdot 10^{-7}$	28.9
OCSL-10	$3.28 \cdot 10^{-7}$	20.9
TSDC-2	$1.25 \cdot 10^{-7}$	8.0
LF2A-6	$9.55 \cdot 10^{-8}$	6.1
SPTR-3	$8.40 \cdot 10^{-8}$	5.3
SPTR-13	$8.00 \cdot 10^{-8}$	5.1
RISB-6	$3.50 \cdot 10^{-8}$	2.2
SHDL-3	$3.33 \cdot 10^{-8}$	2.1
SPTR-23	$3.20 \cdot 10^{-8}$	2.0
RIFL-1	$2.91 \cdot 10^{-8}$	1.9
RISB-8	$2.53 \cdot 10^{-8}$	1.6
LF2B-6	$2.14 \cdot 10^{-8}$	1.4
TSEC-3	$2.10 \cdot 10^{-8}$	1.3
LOCP-6	$2.02 \cdot 10^{-8}$	1.3
LONP-5	$1.77 \cdot 10^{-8}$	1.1
TSDC-5	$1.37 \cdot 10^{-8}$	0.9
SFPB-5	$1.26 \cdot 10^{-8}$	0.8
SFPB-6	$1.26 \cdot 10^{-8}$	0.8
TBOP-3	$1.23 \cdot 10^{-8}$	0.8
LCDL-1	$1.10 \cdot 10^{-8}$	0.7
SHDL-2	$8.40 \cdot 10^{-9}$	0.5
TAIR-3	$8.32 \cdot 10^{-9}$	0.5
SCRM-2	$5.88 \cdot 10^{-9}$	0.4
TSEC-2	$5.29 \cdot 10^{-9}$	0.3
LF1A-7	$4.55 \cdot 10^{-9}$	0.3
RIFS-6	$4.55 \cdot 10^{-9}$	0.3
MELF-5	$4.50 \cdot 10^{-9}$	0.3
OELF-4	$4.50 \cdot 10^{-9}$	0.3
OELF-5	$4.50 \cdot 10^{-9}$	0.3
LF1A-2	$4.20 \cdot 10^{-9}$	0.3
TSHC-2	$3.92 \cdot 10^{-9}$	0.2
LF2A-7	$3.72 \cdot 10^{-9}$	0.2
SHDS-3	$3.51 \cdot 10^{-9}$	0.2
LONP-3	$3.33 \cdot 10^{-9}$	0.2
RIFS-8	$3.28 \cdot 10^{-9}$	0.2
TBOP-2	$3.11 \cdot 10^{-9}$	0.2

Table 11.3 Summary of Dominant Sequences Contributing to Fuel and Experiment Damage (Cont'd)

ACCIDENT SEQUENCE	FREQUENCY yr^{-1}	PERCENTAGE %
LF2D-1	2.60×10^{-9}	0.2
OCPP-2	2.52×10^{-9}	0.2
LF1C-7	2.27×10^{-9}	0.1
TSDC-4	2.20×10^{-9}	0.1
LF1C-2	2.10×10^{-9}	0.1
TAIR-2	2.10×10^{-9}	0.1
LF2F-6	1.64×10^{-9}	0.1
LF1B-7	1.36×10^{-9}	0.1
SFPB-8	1.26×10^{-9}	0.1
LF1B-2	1.26×10^{-9}	0.1
LF2C-6	1.00×10^{-9}	0.1
LONP-2	8.40×10^{-10}	0.1
OCSL-2	8.40×10^{-10}	0.1
LOCP-2	8.40×10^{-10}	0.1
LF2B-7	8.33×10^{-10}	0.1
SFLE-5	5.40×10^{-10}	0.0
SFLE-6	5.40×10^{-10}	0.0
OCSL-9	4.55×10^{-10}	0.0
MELF-4	4.50×10^{-10}	0.0
OELF-7	4.50×10^{-10}	0.0
SCRM-5	4.09×10^{-10}	0.0
EVTR-4	3.96×10^{-10}	0.0
SPTR-6	3.33×10^{-10}	0.0
LF1A-5	2.92×10^{-10}	0.0
ANTC-3	2.92×10^{-10}	0.0
OCSP-2	2.77×10^{-10}	0.0
RISA-7	2.73×10^{-10}	0.0
RISA-2	2.52×10^{-10}	0.0
LF1D-7	2.45×10^{-10}	0.0
LF1D-2	2.27×10^{-10}	0.0
LF2A-2	1.76×10^{-10}	0.0
OCPP-5	1.75×10^{-10}	0.0
LF1C-5	1.46×10^{-10}	0.0
SFPB-7	1.26×10^{-10}	0.0
SCRM-4	1.03×10^{-10}	0.0
LF1B-5	8.76×10^{-11}	0.0
LF1A-4	7.35×10^{-11}	0.0
RISB-7	7.00×10^{-11}	0.0

Table 11.3 Summary of Dominant Sequences Contributing to Fuel and
Experiment Damage (Cont'd)

ACCIDENT SEQUENCE	FREQUENCY yr^{-1}	PERCENTAGE %
TSHC-4	6.87×10^{-11}	0.0
RISB-2	6.47×10^{-11}	0.0
LF2F-7	6.38×10^{-11}	0.0
LOCP-5	5.83×10^{-11}	0.0
OCSL-5	5.83×10^{-11}	0.0
MELF-7	4.50×10^{-11}	0.0
OCPP-4	4.41×10^{-11}	0.0
LF2B-2	3.95×10^{-11}	0.0
LF2C-7	3.90×10^{-11}	0.0
LF1C-4	3.67×10^{-11}	0.0
SPTR-16	3.52×10^{-11}	0.0
LF2E-6	3.18×10^{-11}	0.0
SPTR-9	2.94×10^{-11}	0.0
SPTR-19	2.80×10^{-11}	0.0
LF1B-4	2.20×10^{-11}	0.0
OCSP-5	1.92×10^{-11}	0.0
TSHC-5	1.92×10^{-11}	0.0
RISA-5	1.75×10^{-11}	0.0
LF1D-5	1.57×10^{-11}	0.0
EVTR-6	1.54×10^{-11}	0.0
LOCP-4	1.47×10^{-11}	0.0
OCSL-4	1.47×10^{-11}	0.0
EVTR-13	1.41×10^{-11}	0.0
SPTR-10	1.29×10^{-11}	0.0
SFPB-9	1.26×10^{-11}	0.0
LF2A-5	1.22×10^{-11}	0.0

DAMAGE SEQUENCES

Fig. 11.2. Contribution of Dominant Sequences to Damage Frequency in EBR-II

frequency of damage, while the contribution of the first 20 sequences is about 93 % of the total. The frequencies of the two most dominant sequences, LONP-4 and OCSL-10, are $4.55 \cdot 10^{-7} \text{ yr}^{-1}$ and $3.28 \cdot 10^{-7} \text{ yr}^{-1}$ respectively, and together they account for almost 50 % of the damage frequency. The sequences with the highest frequency are briefly described below.

Sequence LONP-4

This sequence is initiated by the loss of normal power and continues with the failure of the scram system under LOF conditions. The scram signal is successfully generated, but the control or safety rods are not inserted. The contributions to the scram failure are dominated by the common cause failures of the CP relays to open their contacts, the common cause failures of safety and control rods to drop, and the operator failure to push the 3MPB button. This sequence leads to potential experiment damage, because of the availability of constant power, which maintains excitation to the motor-generator clutches and does not result in the most severe coastdown of the primary pumps. This sequence accounts for about 28.9 % of the total damage frequency, and over 77 % of its damage class. Common cause failures are almost entirely responsible for the high frequency of this sequence.

Sequence OCSL-10

The initiator for this sequence is an overcooling caused by a major steam leak, and continues with a failure of the reactor safety system to produce a scram signal under TOP conditions. The dominant events in this sequence are human errors (failure to push scram button and a common cause miscalibration of the SOT sensors), common cause failures of the trip relays, and common mode failure of the power supply components for the detection system. The sequence, which leads to minor core damage and accounts for 20.9 % of the total damage frequency, is particularly sensitive to common cause and human error. This sequence frequency represents over 76 % of the minor core damage class frequency.

Sequence TSDC-2

The unrecoverable loss of a shutdown cooler initiates this sequence. The balance of plant is available for decay heat removal in the short term, but heat removal is insufficient in the long term, when either the remaining shutdown cooler or the shield and thimble cooling must be available. The failure of the long term heat removal is dominated by the combination of a single failure in the remaining shutdown cooler and a single failure in the shield or thimble cooling system that makes it inoperable. The single failures in the shutdown cooler are either a NaK leakage or a NaK blockage, while the most important failures in the shield/thimble cooling are the blockage of a shield cooling system filter, the human failure to recover it, the common cause blockage of the thimble cooling filters, and the unavailability of reactor water because of a water filter blockage. As a loss of decay heat removal this sequence contributes to the core/structural damage class, CSD, and is responsible for 27 % of the frequency in its class and about 8 % of the damage frequency.

Sequence LF2A-6

A double pump loss of flow sequence that proceeds in an identical manner to that of the LONP-4 sequence described above. The only difference is in the initiating event, which in this case is a loss of both primary pumps due to binding or to MG input faults instead of the loss of normal power. Since the LOF signal generation is successful, the minimal cutsets of the sequence are again dominated by the common cause failures of the CP relays, the common cause failures of the rods to drop and the failure of the operator to push the 3MPB scram button. Similarly, this sequence results in potential experiment damage, being responsible for about 16 % of the annual frequency of this damage class. The contribution of LF2A-6 to the damage frequency is of 6.1 %.

Sequence SPTR-3

The sequence is characterized by an initially small tube leak in a superheater module that gradually grows by self wastage and wastage of adjacent tubes. The leak becomes large enough to force backflow of water and reaction products to the IHX. The operator fails to mitigate the accident by dumping sodium and water in the secondary system and the steam generator before the leak grows to the critical size, four times the cross sectional area of a steam tube. The operator also fails to shutdown the reactor before the IHX fails, thus resulting in potential core/structural damage. This sequence accounts for approximately 18 % of the frequency in its damage class, and 5.3 % of the total damage frequency.

Sequence SPTR-13

The sequence is characterized by a rapid progression of a moderately sized tube leak in a superheater module to a leak size greater than four times the cross sectional area of a steam tube. This accident is not vigorous enough to cause failure of the superheater shell or secondary piping, which would lead to rapid pressure relief. The accident results in the transportation of water and reaction products to the IHX, causing its eventual failure. As in the previous sequence, the operator fails to shutdown the reactor before the IHX failure, so that the sequence is also a contributor to the CSD class, in which it is responsible for 17 % of the annual frequency. The frequency of SGTR-13 is about 5 % of the damage frequency in EBR-II.

Sequence RISB-6

A reactivity insertion between 20 and 65 ¢ initiates a TOP transient. The reactor safety system generates a scram signal under TOP conditions, but the scram fails. Although the scram signal has been generated, the rods fail to drop. Since the reasons for the rods not dropping are independent of the conditions under which the scram signal is generated, the minimal cutsets for this TOP sequence are dominated by the same

events as for the LOF sequences LONP-4 and LF2A-6, namely the common cause failure of the CP relays, the common cause failure of the rods to move, and the operator error to push the scram button. Contrary to the two LOF sequences, RISB-6 results in minor core damage (it accounts for about 8 % of the frequency for this damage class) and is responsible for 2.2 % of the annual damage frequency.

Sequence SHDL-3

This sequence is characterized by the short term failure of the decay heat removal when the annual long shutdown is initiated. The loss of the decay heat removal in the short term involves the failure of both shutdown coolers. The dominant minimal cutsets for the sequence consist of the simultaneous failure of the two coolers by NaK leakage or blockage and the failure of the electronic controller circuits combined with the human failure to recover the shutdown coolers. Like other loss of decay heat removal sequences, SHDL-3 leads to core/structural damage. It accounts for over 7 % of the annual frequency of the CSD class and for over 2 % of the damage frequency.

Sequence SPTR-23

The sequence is initiated by the simultaneous (within 0.2 s) double-ended guillotine failure of two or more steam tubes in a superheater module. The pressure wave generated by the accident causes direct failure of the IHX, resulting in a potential core/structural damage. The superheater shell or secondary piping do not fail immediately, allowing the pressure wave to travel to the IHX. This sequence is responsible for almost 7 % of the core/structural damage annual frequency and for 2 % of the total damage frequency.

Sequence RIFL-1

RIFL-1 is one of the direct damage sequences. Its frequency is therefore that of the initiator. The initiating event is a fast reactivity insertion caused by the rupture of

the primary tank. The rupture of the tank results in the reactor core dropping away from the control rods, with the consequent large and fast reactivity insertion. The major contributor to this mechanism is an overpressurization of the argon above the sodium level in the primary tank. The annual frequency of this accident sequence represents a contribution of about 32 % to the frequency of the core damage (CD) class, and 1.9 % to the total damage frequency.

All other sequences contribute less than 2 % to the sum of the frequencies of all the dominant damage sequences. Among those, only five have a contribution in excess of 1 %, for a combined frequency of 6.7 %.

11.3.2 Results by Plant Damage Class

The accident sequence results are shown in Table 11.4 grouped by damage class. This table also includes the sequences leading to major plant disruption with no fuel damage (ND sequences). Class P1 is made up of only one dominant sequence, the direct damage sequence RIFL-1.

Almost 60 % of the contribution to the total frequency of class P2 comes from two sequences described above, namely TSDC-2 and SHDL-3. One of these sequences involves the failure of the short term decay heat removal system, while the other involves the failure in the long term. Among the remaining sequences, 12 of them contribute more than 1 % each to the total for class P2, and they represent a mixture of failure of the decay heat removal in the short and in the long term.

The dominant sequences in class P3 are essentially no damage (ND) sequences, mainly from loss of heat sink, single pump loss of flow, and overcooling transients. Three of the sequences, TSEC-4, TBOP-4, and TAIR-4, are caused by the failure of manually scrambling the reactor under a LOHS transient. They account for more than 50 % of the total frequency for this class. The other top ND sequences contribute over

Table 11.4 Dominant Sequences Grouped by Transient Class

CLASS: P1			FREQUENCY: $2.91 \cdot 10^{-8} \text{ yr}^{-1}$	CONTRIBUTION: 0.1 %
ACCIDENT SEQUENCE	FREQUENCY yr^{-1}	PERCENTAGE %		
RIFL-1	$2.91 \cdot 10^{-8}$	100.0		
CLASS: P2			FREQUENCY: $2.67 \cdot 10^{-7} \text{ yr}^{-1}$	CONTRIBUTION: 0.9 %
ACCIDENT SEQUENCE	FREQUENCY yr^{-1}	PERCENTAGE %		
TSDC-2	$1.25 \cdot 10^{-7}$	46.9		
SHDL-3	$3.33 \cdot 10^{-8}$	12.5		
TSEC-3	$2.10 \cdot 10^{-8}$	7.8		
TSDC-5	$1.37 \cdot 10^{-8}$	5.1		
TBOP-3	$1.23 \cdot 10^{-8}$	4.6		
SHDL-2	$8.40 \cdot 10^{-9}$	3.1		
TAIR-3	$8.32 \cdot 10^{-9}$	3.1		
SCRM-2	$5.88 \cdot 10^{-9}$	2.2		
TSEC-2	$5.29 \cdot 10^{-9}$	2.0		
LF1A-2	$4.20 \cdot 10^{-9}$	1.6		
TSHC-2	$3.92 \cdot 10^{-9}$	1.5		
SHDS-3	$3.51 \cdot 10^{-9}$	1.3		
LONP-3	$3.33 \cdot 10^{-9}$	1.2		
TBOP-2	$3.11 \cdot 10^{-9}$	1.2		
OCPP-2	$2.52 \cdot 10^{-9}$	0.9		
TSDC-4	$2.20 \cdot 10^{-9}$	0.8		
LF1C-2	$2.10 \cdot 10^{-9}$	0.8		
TAIR-2	$2.10 \cdot 10^{-9}$	0.8		
LF1B-2	$1.26 \cdot 10^{-9}$	0.5		
LONP-2	$8.40 \cdot 10^{-10}$	0.3		
OCSL-2	$8.40 \cdot 10^{-10}$	0.3		
LOCP-2	$8.40 \cdot 10^{-10}$	0.3		
SCRM-5	$4.09 \cdot 10^{-10}$	0.2		
LF1A-5	$2.92 \cdot 10^{-10}$	0.1		
ANTC-3	$2.92 \cdot 10^{-10}$	0.1		
OCSP-2	$2.77 \cdot 10^{-10}$	0.1		
RISA-2	$2.52 \cdot 10^{-10}$	0.1		
LF1D-2	$2.27 \cdot 10^{-10}$	0.1		

Table 11.4 Dominant Sequences Grouped by Transient Class (Cont'd)

CLASS: P2 (cont'd)

ACCIDENT SEQUENCE	FREQUENCY yr^{-1}	PERCENTAGE %
LF2A-2	$1.76 \cdot 10^{-10}$	0.1
OCPP-5	$1.75 \cdot 10^{-10}$	0.1
LF1C-5	$1.46 \cdot 10^{-10}$	0.1
SCRM-4	$1.03 \cdot 10^{-10}$	0.0
LF1B-5	$8.76 \cdot 10^{-11}$	0.0
LF1A-4	$7.35 \cdot 10^{-11}$	0.0
TSHC-4	$6.87 \cdot 10^{-11}$	0.0
RISB-2	$6.47 \cdot 10^{-11}$	0.0
LOCP-5	$5.83 \cdot 10^{-11}$	0.0
OCSL-5	$5.83 \cdot 10^{-11}$	0.0
OCPP-4	$4.41 \cdot 10^{-11}$	0.0
LF2B-2	$3.95 \cdot 10^{-11}$	0.0
LF1C-4	$3.67 \cdot 10^{-11}$	0.0
LF1B-4	$2.20 \cdot 10^{-11}$	0.0
OCSP-5	$1.92 \cdot 10^{-11}$	0.0
TSHC-5	$1.92 \cdot 10^{-11}$	0.0
RISA-5	$1.75 \cdot 10^{-11}$	0.0
LF1D-5	$1.57 \cdot 10^{-11}$	0.0
LOCP-4	$1.47 \cdot 10^{-11}$	0.0
OCSL-4	$1.47 \cdot 10^{-11}$	0.0
LF2A-5	$1.22 \cdot 10^{-11}$	0.0

CLASS: P3 **FREQUENCY:** $1.36 \cdot 10^{-5} \text{ yr}^{-1}$ **CONTRIBUTION:** 44.5 %

ACCIDENT SEQUENCE	FREQUENCY yr^{-1}	PERCENTAGE %
TSEC-4	$3.48 \cdot 10^{-6}$	25.5
LF1A-6	$2.27 \cdot 10^{-6}$	16.7
TBOP-4	$2.04 \cdot 10^{-6}$	15.0
TAIR-4	$1.38 \cdot 10^{-6}$	10.1
LF1C-6	$1.14 \cdot 10^{-6}$	8.3
OCPP-10	$9.85 \cdot 10^{-7}$	7.2

Table 11.4 Dominant Sequences Grouped by Transient Class (Cont'd)

CLASS: P3 (cont'd)

ACCIDENT SEQUENCE	FREQUENCY yr^{-1}	PERCENTAGE %
LF1B-6	$6.82 \cdot 10^{-7}$	5.0
LONP-4	$4.55 \cdot 10^{-7}$	3.3
OCSL-10	$3.28 \cdot 10^{-7}$	2.4
RISA-6	$1.36 \cdot 10^{-7}$	1.0
LF1D-6	$1.23 \cdot 10^{-7}$	0.9
OCSP-10	$1.08 \cdot 10^{-7}$	0.8
RISA-8	$9.85 \cdot 10^{-8}$	0.7
LF2A-6	$9.55 \cdot 10^{-8}$	0.7
LF1A-8	$8.86 \cdot 10^{-8}$	0.7
LF1C-8	$4.43 \cdot 10^{-8}$	0.3
RISB-6	$3.50 \cdot 10^{-8}$	0.3
LF1B-8	$2.66 \cdot 10^{-8}$	0.2
RISB-8	$2.53 \cdot 10^{-8}$	0.2
LF2B-6	$2.14 \cdot 10^{-8}$	0.2
LOCP-6	$2.02 \cdot 10^{-8}$	0.1
LONP-5	$1.77 \cdot 10^{-8}$	0.1
LF1D-8	$4.79 \cdot 10^{-9}$	0.0
LF1A-7	$4.55 \cdot 10^{-9}$	0.0
RIFS-6	$4.55 \cdot 10^{-9}$	0.0
LF2A-7	$3.72 \cdot 10^{-9}$	0.0
RIFS-8	$3.28 \cdot 10^{-9}$	0.0
LF1C-7	$2.27 \cdot 10^{-9}$	0.0
LF2F-6	$1.64 \cdot 10^{-9}$	0.0
LF1B-7	$1.36 \cdot 10^{-9}$	0.0
OCPP-9	$1.36 \cdot 10^{-9}$	0.0
LF2C-6	$1.00 \cdot 10^{-9}$	0.0
LF2B-7	$8.33 \cdot 10^{-10}$	0.0
OCSL-9	$4.55 \cdot 10^{-10}$	0.0
RISA-7	$2.73 \cdot 10^{-10}$	0.0
LF1D-7	$2.45 \cdot 10^{-10}$	0.0
OCSP-9	$1.50 \cdot 10^{-10}$	0.0
RISB-7	$7.00 \cdot 10^{-11}$	0.0
LF2F-7	$6.38 \cdot 10^{-11}$	0.0
LF2C-7	$3.90 \cdot 10^{-11}$	0.0
LF2E-6	$3.18 \cdot 10^{-11}$	0.0

Table 11.4 Dominant Sequences Grouped by Transient Class (Cont'd)

CLASS: P4 **FREQUENCY: $1.36 \cdot 10^{-8} \text{ yr}^{-1}$** **CONTRIBUTION: 0.0 %**

ACCIDENT SEQUENCE	FREQUENCY yr^{-1}	PERCENTAGE %
LCDL-1	$1.10 \cdot 10^{-8}$	80.9
LF2D-1	$2.60 \cdot 10^{-9}$	19.1

CLASS: P5 **FREQUENCY: $1.67 \cdot 10^{-5} \text{ yr}^{-1}$** **CONTRIBUTION: 54.4 %**

ACCIDENT SEQUENCE	FREQUENCY yr^{-1}	PERCENTAGE %
SPTR-2	$8.40 \cdot 10^{-6}$	50.4
SPTR-12	$8.00 \cdot 10^{-6}$	48.0
SPTR-3	$8.40 \cdot 10^{-8}$	0.5
SPTR-13	$8.00 \cdot 10^{-8}$	0.5
EVTR-3	$3.96 \cdot 10^{-8}$	0.2
SPTR-5	$3.33 \cdot 10^{-8}$	0.2
SPTR-23	$3.20 \cdot 10^{-8}$	0.2
SPTR-15	$3.52 \cdot 10^{-9}$	0.0
SPTR-8	$2.94 \cdot 10^{-9}$	0.0
SPTR-18	$2.80 \cdot 10^{-9}$	0.0
EVTR-9	$7.04 \cdot 10^{-10}$	0.0
EVTR-4	$3.96 \cdot 10^{-10}$	0.0
SPTR-6	$3.33 \cdot 10^{-10}$	0.0
SPTR-16	$3.52 \cdot 10^{-11}$	0.0
SPTR-9	$2.94 \cdot 10^{-11}$	0.0
SPTR-19	$2.80 \cdot 10^{-11}$	0.0
EVTR-6	$1.54 \cdot 10^{-11}$	0.0
EVTR-13	$1.41 \cdot 10^{-11}$	0.0
SPTR-10	$1.29 \cdot 10^{-11}$	0.0

Table 11.4 Dominant Sequences Grouped by Transient Class (Cont'd)

CLASS: P6 **FREQUENCY: $4.21 \cdot 10^{-8} \text{ yr}^{-1}$** **CONTRIBUTION: 0.1 %**

ACCIDENT SEQUENCE	FREQUENCY yr^{-1}	PERCENTAGE %
SFPB-5	$1.26 \cdot 10^{-8}$	29.9
SFPB-6	$1.26 \cdot 10^{-8}$	29.9
MELF-5	$4.50 \cdot 10^{-9}$	10.7
OELF-4	$4.50 \cdot 10^{-9}$	10.7
OELF-5	$4.50 \cdot 10^{-9}$	10.7
SFPB-8	$1.26 \cdot 10^{-9}$	3.0
SFLE-5	$5.40 \cdot 10^{-10}$	1.3
SFLE-6	$5.40 \cdot 10^{-10}$	1.3
MELF-4	$4.50 \cdot 10^{-10}$	1.1
OELF-7	$4.50 \cdot 10^{-10}$	1.1
SFPB-7	$1.26 \cdot 10^{-10}$	0.3
MELF-7	$4.50 \cdot 10^{-11}$	0.1
SFPB-9	$1.26 \cdot 10^{-11}$	0.0

40 % of the frequency of P3. The first damage sequence is LONP-4, with a contribution of 3.3 %.

There are only two sequences in class P4, and they are both direct damage sequences due to a delayed loss of circulation (LCDL) and a double pump loss of flow (group D, LF2D) initiating events. The delayed loss of circulation frequency is close to one order of magnitude above that of the LF2D, and it accounts for almost 81 % of the annual frequency for this class.

The steam generator tube rupture class, P5, is dominated by two ND sequences, SPTR-2 and SPTR-12, which together represent over 98 % of the annual frequency for the class. In class P6, reserved for local faults, the main contributors are sequences of the subassembly partial blockage initiating event. The top two sequences have the same annual frequency and one contributes to core damage while the other is in the minor core damage class. Together, their frequency is about 60 % of the frequency of P6. Experiment local faults (both metal and oxide fuel) have a significant contribution to the P6 annual frequency (more than 30 %), while no sequences of the metal driver fuel local fault initiator appear among the dominant sequences.

A graphical representation of the results by transient class, showing the fractions of the total class frequency carried by the most dominant sequences, is shown in Figures 11.3 to 11.8. Figure 11.9 displays the contribution of each transient class to the total (damage plus ND) annual frequency in EBR-II.

11.3.3 Results by Damage Category

The results of the accident sequence quantification assembled by damage type are shown in Table 11.5, and graphically in Figures 11.10 to 11.14. Figure 11.15 shows the contribution by damage class to the annual damage frequency in EBR-II, excluding the no damage category, ND.

CLASS: P1

Fig. 11.3. Dominant Sequences in Plant Transient Class P1

CLASS: P2

Fig. 11.4. Dominant Sequences in Plant Transient Class P2

CLASS: P3

Fig. 11.5. Dominant Sequences in Plant Transient Class P3

CLASS: P4

Fig. 11.6. Dominant Sequences in Plant Transient Class P4

CLASS: P5

Fig. 11.7. Dominant Sequences in Plant Transient Class P5

CLASS: P6

Fig. 11.8. Dominant Sequences in Plant Transient Class P6

CONTRIBUTION BY CLASS

Fig. 11.9. Comparison of Plant Transient Classes by their Contribution to Total Frequency (damage + ND)

Table 11.5 Dominant Sequences Grouped by Damage Category

DAMAGE CLASS: CD

FREQUENCY: $9.01 \cdot 10^{-8} \text{ yr}^{-1}$

CONTRIBUTION TO DAMAGE: 5.7 %

CONTRIBUTION TO TOTAL: 0.3 %

ACCIDENT SEQUENCE	FREQUENCY yr^{-1}	PERCENTAGE %
RIFL-1	$2.91 \cdot 10^{-8}$	32.3
LF2B-6	$2.14 \cdot 10^{-8}$	23.7
LOCP-6	$2.02 \cdot 10^{-8}$	22.4
SFPB-5	$1.26 \cdot 10^{-8}$	14.0
OELF-4	$4.50 \cdot 10^{-9}$	5.0
LF2B-7	$8.33 \cdot 10^{-10}$	0.9
SFLE-5	$5.40 \cdot 10^{-10}$	0.6
MELF-4	$4.50 \cdot 10^{-10}$	0.5
RISA-7	$2.73 \cdot 10^{-10}$	0.3
SFPB-7	$1.26 \cdot 10^{-10}$	0.1
RISB-7	$7.00 \cdot 10^{-11}$	0.1
LF2E-6	$3.18 \cdot 10^{-11}$	0.0
SFPB-9	$1.26 \cdot 10^{-11}$	0.0

DAMAGE CLASS: MCD

FREQUENCY: $4.30 \cdot 10^{-7} \text{ yr}^{-1}$

CONTRIBUTION TO DAMAGE: 27.4 %

CONTRIBUTION TO TOTAL: 1.4 %

ACCIDENT SEQUENCE	FREQUENCY yr^{-1}	PERCENTAGE %
OCSL-10	$3.28 \cdot 10^{-7}$	76.3
RISB-6	$3.50 \cdot 10^{-8}$	8.1
RISB-8	$2.53 \cdot 10^{-8}$	5.9
SFPB-6	$1.26 \cdot 10^{-8}$	2.9
RIFS-6	$4.55 \cdot 10^{-9}$	1.1
MELF-5	$4.50 \cdot 10^{-9}$	1.0
OELF-5	$4.50 \cdot 10^{-9}$	1.0
RIFS-8	$3.28 \cdot 10^{-9}$	0.8
LF2D-1	$2.60 \cdot 10^{-9}$	0.6
LF1C-7	$2.27 \cdot 10^{-9}$	0.5
LF2F-6	$1.64 \cdot 10^{-9}$	0.4

Table 11.5 Dominant Sequences Grouped by Damage Category (Cont'd)

DAMAGE CLASS: MCD (cont'd)

ACCIDENT SEQUENCE	FREQUENCY yr^{-1}	PERCENTAGE %
LF1B-7	$1.36 \cdot 10^{-9}$	0.3
SFPB-8	$1.26 \cdot 10^{-9}$	0.3
LF2C-6	$1.00 \cdot 10^{-9}$	0.2
SFLE-6	$5.40 \cdot 10^{-10}$	0.1
OCSL-9	$4.55 \cdot 10^{-10}$	0.1
OELF-7	$4.50 \cdot 10^{-10}$	0.1
LF1D-7	$2.45 \cdot 10^{-10}$	0.1
LF2F-7	$6.38 \cdot 10^{-11}$	0.0
MELF-7	$4.50 \cdot 10^{-11}$	0.0
LF2C-7	$3.90 \cdot 10^{-11}$	0.0

DAMAGE CLASS: CSD

FREQUENCY: $4.64 \cdot 10^{-7} \text{ yr}^{-1}$
 CONTRIBUTION TO DAMAGE: 29.5 %
 CONTRIBUTION TO TOTAL: 1.5 %

ACCIDENT SEQUENCE	FREQUENCY yr^{-1}	PERCENTAGE %
TSDC-2	$1.25 \cdot 10^{-7}$	27.0
SPTR-3	$8.40 \cdot 10^{-8}$	18.1
SPTR-13	$8.00 \cdot 10^{-8}$	17.2
SHDL-3	$3.33 \cdot 10^{-8}$	7.2
SPTR-23	$3.20 \cdot 10^{-8}$	6.9
TSEC-3	$2.10 \cdot 10^{-8}$	4.5
TSDC-5	$1.37 \cdot 10^{-8}$	3.0
TBOP-3	$1.23 \cdot 10^{-8}$	2.7
SHDL-2	$8.40 \cdot 10^{-9}$	1.8
TAIR-3	$8.32 \cdot 10^{-9}$	1.8
SCRM-2	$5.88 \cdot 10^{-9}$	1.3
TSEC-2	$5.29 \cdot 10^{-9}$	1.1
LF1A-2	$4.20 \cdot 10^{-9}$	0.9
TSHC-2	$3.92 \cdot 10^{-9}$	0.8
SHDS-3	$3.51 \cdot 10^{-9}$	0.8
LONP-3	$3.33 \cdot 10^{-9}$	0.7

Table 11.5 Dominant Sequences Grouped by Damage Category (Cont'd)

DAMAGE CLASS: CSD (cont'd)

ACCIDENT SEQUENCE	FREQUENCY yr^{-1}	PERCENTAGE %
TBOP-2	$3.11 \cdot 10^{-09}$	0.7
OCPP-2	$2.52 \cdot 10^{-09}$	0.5
TSDC-4	$2.20 \cdot 10^{-09}$	0.5
LF1C-2	$2.10 \cdot 10^{-09}$	0.5
TAIR-2	$2.10 \cdot 10^{-09}$	0.5
LF1B-2	$1.26 \cdot 10^{-09}$	0.3
LONP-2	$8.40 \cdot 10^{-10}$	0.2
OCSL-2	$8.40 \cdot 10^{-10}$	0.2
LOCP-2	$8.40 \cdot 10^{-10}$	0.2
SCRM-5	$4.09 \cdot 10^{-10}$	0.1
EVTR-4	$3.96 \cdot 10^{-10}$	0.1
SPTR-6	$3.33 \cdot 10^{-10}$	0.1
LF1A-5	$2.92 \cdot 10^{-10}$	0.1
ANTC-3	$2.92 \cdot 10^{-10}$	0.1
OCSP-2	$2.77 \cdot 10^{-10}$	0.1
RISA-2	$2.52 \cdot 10^{-10}$	0.1
LF1D-2	$2.27 \cdot 10^{-10}$	0.0
LF2A-2	$1.76 \cdot 10^{-10}$	0.0
OCPP-5	$1.75 \cdot 10^{-10}$	0.0
LF1C-5	$1.46 \cdot 10^{-10}$	0.0
SCRM-4	$1.03 \cdot 10^{-10}$	0.0
LF1B-5	$8.76 \cdot 10^{-11}$	0.0
LF1A-4	$7.35 \cdot 10^{-11}$	0.0
TSHC-4	$6.87 \cdot 10^{-11}$	0.0
RISB-2	$6.47 \cdot 10^{-11}$	0.0
LOCP-5	$5.83 \cdot 10^{-11}$	0.0
OCSL-5	$5.83 \cdot 10^{-11}$	0.0
OCPP-4	$4.41 \cdot 10^{-11}$	0.0
LF2B-2	$3.95 \cdot 10^{-11}$	0.0
LF1C-4	$3.67 \cdot 10^{-11}$	0.0
SPTR-16	$3.52 \cdot 10^{-11}$	0.0
SPTR-9	$2.94 \cdot 10^{-11}$	0.0
SPTR-19	$2.80 \cdot 10^{-11}$	0.0
LF1B-4	$2.20 \cdot 10^{-11}$	0.0
OCSP-5	$1.92 \cdot 10^{-11}$	0.0

Table 11.5 Dominant Sequences Grouped by Damage Category (Cont'd)

DAMAGE CLASS: CSD (cont'd)

ACCIDENT SEQUENCE	FREQUENCY yr ⁻¹	PERCENTAGE %
TSHC-5	1.92 10 ⁻¹¹	0.0
RISA-5	1.75 10 ⁻¹¹	0.0
LF1D-5	1.57 10 ⁻¹¹	0.0
EVTR-6	1.54 10 ⁻¹¹	0.0
LOCP-4	1.47 10 ⁻¹¹	0.0
OCSL-4	1.47 10 ⁻¹¹	0.0
EVTR-13	1.41 10 ⁻¹¹	0.0
SPTR-10	1.29 10 ⁻¹¹	0.0
LF2A-5	1.22 10 ⁻¹¹	0.0

DAMAGE CLASS: PED

FREQUENCY: 5.87 10⁻⁰⁷ yr⁻¹
 CONTRIBUTION TO DAMAGE: 37.4 %
 CONTRIBUTION TO TOTAL: 1.9 %

ACCIDENT SEQUENCE	FREQUENCY yr ⁻¹	PERCENTAGE %
LONP-4	4.55 10 ⁻⁰⁷	77.4
LF2A-6	9.55 10 ⁻⁰⁸	16.3
LONP-5	1.77 10 ⁻⁰⁸	3.0
LCDL-1	1.10 10 ⁻⁰⁸	1.9
LF1A-7	4.55 10 ⁻⁰⁹	0.8
LF2A-7	3.72 10 ⁻⁰⁹	0.6

DAMAGE CLASS: ND

FREQUENCY: 2.91 10⁻⁰⁵ yr⁻¹
 CONTRIBUTION TO DAMAGE: -----
 CONTRIBUTION TO TOTAL: 94.9 %

ACCIDENT SEQUENCE	FREQUENCY yr ⁻¹	PERCENTAGE %
SPTR-2	8.40 10 ⁻⁰⁶	28.9
SPTR-12	8.00 10 ⁻⁰⁶	27.5
TSEC-4	3.48 10 ⁻⁰⁶	12.0

Table 11.5 Dominant Sequences Grouped by Damage Category (Cont'd)

DAMAGE CLASS: ND (cont'd)

ACCIDENT SEQUENCE	FREQUENCY yr^{-1}	PERCENTAGE %
LF1A-6	$2.27 \cdot 10^{-06}$	7.8
TBOP-4	$2.04 \cdot 10^{-06}$	7.0
TAIR-4	$1.38 \cdot 10^{-06}$	4.7
LF1C-6	$1.14 \cdot 10^{-06}$	3.9
OCPP-10	$9.85 \cdot 10^{-07}$	3.4
LF1B-6	$6.82 \cdot 10^{-07}$	2.3
RISA-6	$1.36 \cdot 10^{-07}$	0.5
LF1D-6	$1.23 \cdot 10^{-07}$	0.4
OCSP-10	$1.08 \cdot 10^{-07}$	0.4
RISA-8	$9.85 \cdot 10^{-08}$	0.3
LF1A-8	$8.86 \cdot 10^{-08}$	0.3
LF1C-8	$4.43 \cdot 10^{-08}$	0.2
EVTR-3	$3.96 \cdot 10^{-08}$	0.1
SPTR-5	$3.33 \cdot 10^{-08}$	0.1
LF1B-8	$2.66 \cdot 10^{-08}$	0.1
LF1D-8	$4.79 \cdot 10^{-09}$	0.0
SPTR-15	$3.52 \cdot 10^{-09}$	0.0
SPTR-8	$2.94 \cdot 10^{-09}$	0.0
SPTR-18	$2.80 \cdot 10^{-09}$	0.0
OCPP-9	$1.36 \cdot 10^{-09}$	0.0
EVTR-9	$7.04 \cdot 10^{-10}$	0.0
OCSP-9	$1.50 \cdot 10^{-10}$	0.0

FUEL DAMAGE: CD

Fig. 11.10. Dominant Sequences in the Core Damage Category

FUEL DAMAGE: MCD

Fig. 11.11. Dominant Sequences in the Minor Core Damage Category

FUEL DAMAGE: CSD

Fig. 11.12. Dominant Sequences in the Core/Structural Damage Category

FUEL DAMAGE: PED

Fig. 11.13. Dominant Sequences in the Potential Experiment Damage Category

FUEL DAMAGE: ND

Fig. 11.14. Dominant Sequences in the No Damage Category

CONTRIBUTION BY DAMAGE TYPE

Fig. 11.15. Comparison of Damage Categories by their Contribution to the Total Damage Frequency

The most severe damage class, core damage (CD), accounts for less than 6 % of the damage frequency. The most important sequence in this category is RIFL-1 (over 32 % of the frequency) followed by two sequences that involve a double clutch LOF with failure to scram (LF2B-6 and LOCP-6). Local fault sequences also carry a significant fraction of the annual frequency of CD.

More than 76 % of the total frequency in the minor core damage (MCD) category is due to the sequence OCSL-10, followed by other smaller contributions mostly from reactivity insertion and local faults initiators. Because the frequency of OCSL-10 is not only large within its class, but also relative to the entire damage frequency, the minor core damage class accounts for over 27 % of the annual damage frequency in EBR-II.

In the core/structural damage category (CSD) the dominant sequences involve the short term decay heat removal (those in the P2 class), and some of the superheater tube rupture sequences. The loss of a shutdown cooler has three dominant sequences in this class, in particular TSDC-2 (loss of a shutdown cooler and later failure of the long term decay heat removal), which accounts for 27 % of the CSD annual frequency. Three superheater tube rupture sequences carry more than 42 % of the frequency in this class. Other significant sequences in CSD are among the shutdown and loss of heat sink initiating events. The core/structural damage class is responsible for almost 30 % of the annual damage frequency.

The two most dominant sequences in the potential experiment damage category (PED) are initiated with a double pump loss of flow with similar coastdown characteristics (LONP-4 and LF2A-6). Over 77 % of the frequency of the PED class can be attributed to the loss of normal power sequence, with the LF2A sequence being a distant second with 16 % of the PED annual frequency. Another, minor, sequence in this class is delayed loss of circulation, LCDL-1, which is a direct damage sequence. Because of the comparatively high frequency of the LONP-4 sequence, the PED class represents more than 37 % of the annual damage frequency in EBR-II.

The no-damage (ND) class has the highest annual frequency, close to 3×10^{-5} . More than 56 % of this frequency is attributed to the superheater tube rupture initiator. The result is a potential severe damage and contamination of the IHX. A distant second group of sequences

in this class are formed by the three unprotected loss of heat sink sequences (TSEC-4, TBOP-4, TAIR-4), together accountable for almost 25 % of the ND annual frequency. Although the ND class does not contribute to fuel or experiment damage, it eclipses the damage classes when their frequencies are compared. As shown in Figure 11.16, the ND class is responsible for almost 95 % of the total (damage plus ND) annual frequency.

11.3.4 Results by Initiating Event

Table 11.6 presents the results of the accident sequence quantification associated by initiating event. The sequences leading to no fuel damage (ND) have been excluded from the table and from the calculation of the percentage contributions. The most important single initiator is the loss of normal power (LONP) with 30 % of the damage frequency. The next dominant initiators are the overcooling by major steam leak and the superheater tube rupture, which account for 21 % and 12 % of the damage frequency respectively. The unrecoverable loss of a shutdown cooler is another significant initiating event, responsible for 9 % of the damage frequency, and double pump loss of flow due to binding/MG input faults carries an additional 6.3 %. In summary, 32 % of the damage is due to loss of power initiators, 21 % due to overcooling transients, 13 % due to steam generator tube failures, 9 % due to loss of a decay heat removal system, 9 % due to the loss of flow initiators, 6 % is contributed by reactivity insertions, 7 % is due to shutdown and loss of heat sink initiators, 2 % is attributed to local faults, and less than 1 % can be imputed to the loss of circulation initiators. The relative contributions of the different initiators to the annual damage frequency in EBR-II is also shown in Fig. 11.17, while Fig. 11.18 depicts the contribution of the main initiators to the total frequency of all the dominant sequences (including the no damage, ND, sequences).

11.3.5 Results by Event Tree

The accident sequence quantification results are also listed by event tree in Figures 11.19 to 11.56. The value in the frequency column is listed for all sequences that were quantified. A blank is left for those sequences that were not analyzed because they did not lead to any damage state. The sequences that, after quantification, were found to have an annual frequency below the cutoff value of $1.0 \cdot 10^{-11} \text{ yr}^{-1}$ are noted by specifying that the frequency was below the cutoff.

CONTRIBUTION BY DAMAGE TYPE

Fig. 11.16. Comparison of Damage Categories by their Contribution to the Total (damage + ND) Frequency

Table 11.6 Damage Accident Sequence Quantification Results Grouped by
Initiating Event

TOTAL DAMAGE FREQUENCY:		1.57 10 ⁻⁶ yr ⁻¹		
INITIATING EVENT		NUMBER OF SEQUENCES	FREQUENCY yr ⁻¹	PERCENTAGE %
ANTC	Anticipatory Shutdown	1	2.92 10 ⁻¹⁰	0.0
EVTR	Evaporator Tube Rupture	4	4.33 10 ⁻¹⁰	0.0
LCDL	Loss of Circulation - Delayed	2	1.10 10 ⁻⁸	0.7
LF1A	Single Pump Loss of Flow - Group A	4	9.11 10 ⁻⁹	0.6
LF1B	Single Pump Loss of Flow - Group B	4	2.73 10 ⁻⁹	0.2
LF1C	Single Pump Loss of Flow - Group C	4	4.56 10 ⁻⁹	0.3
LF1D	Single Pump Loss of Flow - Group D	4	4.92 10 ⁻¹⁰	0.0
LF2A	Double Pump Loss of Flow - Group A	5	9.94 10 ⁻⁸	6.3
LF2B	Double Pump Loss of Flow - Group B	5	2.22 10 ⁻⁸	1.4
LF2C	Double Pump Loss of Flow - Group C	5	1.04 10 ⁻⁹	0.1
LF2D	Double Pump Loss of Flow - Group D	2	2.60 10 ⁻⁹	0.2
LF2E	Double Pump Loss of Flow - Group E	4	3.31 10 ⁻¹¹	0.0
LF2F	Double Pump Loss of Flow - Group F	5	1.70 10 ⁻⁹	0.1
LOCP	Loss of Constant Power	4	2.11 10 ⁻⁸	1.3
LONP	Loss of Normal Power	4	4.77 10 ⁻⁷	30.3
MDLF	Local Faults: Metal Driver Breach Initiator	5	1.48 10 ⁻¹²	0.0
MELF	Local Faults: Metal Experiment Breach Initiator	5	5.00 10 ⁻⁹	0.3
OCPP	Overcooling by Primary Pump Overspeed	4	2.74 10 ⁻⁹	0.2
OCSL	Overcooling by Major Steam Leak	6	3.30 10 ⁻⁷	21.0
OCSF	Overcooling by Secondary Pump Overspeed	3	3.01 10 ⁻¹⁰	0.0
OELF	Local Faults: Oxide Experiment Breach Initiator	5	9.45 10 ⁻⁹	0.6
RIFL	Fast Reactivity Insertion (> 1.00 \$)	1	2.91 10 ⁻⁸	1.9
RIFS	Fast Reactivity Insertion (< 0.35 \$)	6	7.85 10 ⁻⁹	0.5
RISA	Slow Reactivity Insertion - Group A (< 0.20 \$)	4	5.47 10 ⁻¹⁰	0.0
RISB	Slow Reactivity Insertion - Group B (0.20-0.65 \$)	6	6.04 10 ⁻⁸	3.8
RISC	Slow Reactivity Insertion - Group C (0.65-1.10 \$)	4	1.33 10 ⁻¹¹	0.0
RISD	Slow Reactivity Insertion - Group D (> 1.10 \$)	3	3.44 10 ⁻¹²	0.0

Table 11.6 Damage Accident Sequence Quantification Results Grouped by
Initiating Event (Cont'd)

INITIATING EVENT		NUMBER OF SEQUENCES	FREQUENCY yr^{-1}	PERCENTAGE %
SCRM	Scram	3	6.39×10^{-9}	0.4
SFLE	Subassembly Local Fault from Loading Error	5	1.09×10^{-9}	0.1
SFPB	Subassembly Local Fault from Partial Blockage	5	2.66×10^{-8}	1.7
SHDL	Long Shutdown	2	4.17×10^{-8}	2.7
SHDS	Short Shutdown	1	3.51×10^{-9}	0.2
SPTR	Superheater Tube Failure	9	1.96×10^{-7}	12.5
TAIR	Transient - Loss of Instrument Air	2	1.04×10^{-8}	0.7
TBOP	Loss of Steam Plant Heat Removal	2	1.54×10^{-8}	1.0
TSDC	Transient - Unrecoverable Loss of a Shutdown Cooler	3	1.41×10^{-7}	9.0
TSEC	Loss of Secondary System Heat Removal	2	2.63×10^{-8}	1.7
TSHC	Transient - Unrecoverable Loss of Shield Cooling	3	4.01×10^{-9}	0.3

DAMAGE SEQUENCES

Fig. 11.17. Comparison of Initiating Events by their Contribution to the Damage Frequency

ALL DOMINANT SEQUENCES

Fig. 11.18. Comparison of Initiating Events by their Contribution to the Total (damage + ND) Frequency

EVENT TREE FOR: Loss of Circulation - Delayed

31-MAY-91

Loss of Circulation - Delayed	Pumps keep running for 1 hour	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
LCDL	PKRU	DHRS	DHRL				
				LCDL-1	P4	PED	1.1 E-8
				LCDL-2	P2	CSD	<1.0 E-11
				LCDL-3	P2	CSD	<1.0 E-11
				LCDL-4	P1	CD	<1.0 E-11

Fig. 11.19. Event Tree for a Delayed Loss of Circulation with Sequence Frequency Results

EVENT TREE FOR: Single Pump Loss of Flow - Group A

31-MAY-91

Single Pump Loss of Flow - Group A	Scram signal during LOF event	Sufficient control or safety rods drop during LOF event	Primary pumps keep running	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
LF1A	FSIG	FROD	PRUN	BPHR	DHRS	DHRL	LF1A-1			
							LF1A-2	P2	CSD	4.2 E-9
							LF1A-3			
							LF1A-4	P2	CSD	7.4 E-11
							LF1A-5	P2	CSD	2.9 E-10
							LF1A-6	P3	ND	2.3 E-6
							LF1A-7	P3	PED	4.5 E-9
							LF1A-8	P3	ND	8.9 E-8

Fig. 11.20. Event Tree for a Single Pump Loss of Flow (Group A)
with Sequence Frequency Results

EVENT TREE FOR: Single Pump Loss of Flow - Group B

31-MAY-91

Fig. 11.21. Event Tree for a Single Pump Loss of Flow (Group B)
with Sequence Frequency Results

EVENT TREE FOR: Single Pump Loss of Flow - Group C

31-MAY-91

Single Pump Loss of Flow - Group C	Scram signal during LOF event	Sufficient control or safety rods drop during LOF event	Primary pumps keep running	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
LF1C	FSIG	FROD	PRUN	BPHR	DHRS	DHRL	LF1C-1			
							LF1C-2	P2	CSD	2.1 E-9
							LF1C-3			
							LF1C-4	P2	CSD	3.7 E-11
							LF1C-5	P2	CSD	1.5 E-10
							LF1C-6	P3	ND	1.1 E-6
							LF1C-7	P3	MCD	2.3 E-9
							LF1C-8	P3	ND	4.4 E-8

Fig. 11.22. Event Tree for Single Pump Loss of Flow (Group C)
with Sequence Frequency Results

EVENT TREE FOR: Single Pump Loss of Flow - Group D

31-MAY-91

Single Pump Loss of Flow - Group D	Scram signal during LOF event	Sufficient control or safety rods drop during LOF event	Primary pumps keep running	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
LF1D	FSIG	FROD	PRUN	BPHR	DHRS	DHRL	LF1D-1			
							LF1D-2	P2	CSD	2.3 E-10
							LF1D-3			
							LF1D-4	P2	CSD	< 1.0 E-11
							LF1D-5	P2	CSD	1.6 E-11
							LF1D-6	P3	ND	1.2 E-7
							LF1D-7	P3	MCD	2.5 E-10
							LF1D-8	P3	ND	4.8 E-9

Fig. 11.23. Event Tree for Single Pump Loss of Flow (Group D)
with Sequence Frequency Results

EVENT TREE FOR: Double Pump Loss of Flow - Group A

31-MAY-91

Double Pump Loss of Flow - Group A	Scram signal during LOF event	Sufficient control or safety rods drop during LOF event	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
LF2A	FSIG	FROD	BPHR	DHRS	DHRL				
						LF2A-1			
						LF2A-2	P2	CSD	1.8 E-10
						LF2A-3			
						LF2A-4	P2	CSD	<1.0 E-11
						LF2A-5	P2	CSD	1.2 E-11
						LF2A-6	P3	PED	9.5 E-8
						LF2A-7	P3	PED	3.7 E-9

Fig. 11.24. Event Tree for a Double Pump Loss of Flow (Group A) with Sequence Frequency Results

EVENT TREE FOR: Double Pump Loss of Flow - Group B

31-MAY-91

Double Pump Loss of Flow - Group B	Scram signal during LOF event	Sufficient control or safety rods drop during LOF event	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
LF2B	FSIG	FROD	BPHR	DHRS	DHRL				
						LF2B-1	P2	CSD	3.9 E-11
						LF2B-2			
						LF2B-3			
						LF2B-4	P2	CSD	< 1.0 E-11
						LF2B-5			
						LF2B-6	P3	CD	2.1 E-8
						LF2B-7			

Fig. 11.25. Event Tree for a Double Pump Loss of Flow (Group B)
with Sequence Frequency Results

EVENT TREE FOR: Double Pump Loss of Flow - Group C

31-MAY-91

Double Pump Loss of Flow - Group C	Scram signal during LOF event	Sufficient control or safety rods drop during LOF event	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
LF2C	FSIG	FROD	BPFR	DHRS	DHRL				
						LF2C-1	P2	CSD	< 1.0 E-11
						LF2C-2			
						LF2C-3			
						LF2C-4	P2	CSD	< 1.0 E-11
						LF2C-5	P2	CSD	< 1.0 E-11
						LF2C-6	P3	MCD	1.0 E-9
						LF2C-7	P3	MCD	3.9 E-11

Fig. 11.26. Event Tree for a Double Pump Loss of Flow (Group C)
with Sequence Frequency Results

EVENT TREE FOR: Double Pump Loss of Flow - Group D

31-MAY-91

Double Pump Loss of Flow - Group D	Scram signal during LOF event	Sufficient control or safety rods drop during LOF event		Sequence	Class	Fuel Damage	Frequency
		FSIG	FROD				
LF2D				LF2D-1	P4	MCD	2.6 E-9
				LF2D-2	P3	CD	< 1.0 E-11
				LF2D-3	P3	CD	< 1.0 E-11

Fig. 11.27. Event Tree for a Double Pump Loss of Flow (Group D)
with Sequence Frequency Results

EVENT TREE FOR: Double Pump Loss of Flow - Group E

31-MAY-91

Double Pump Loss of Flow - Group E	Scram signal during LOF event	Sufficient control or safety rods drop during LOF event	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
LF2E	FSIG	FROD	BPHR	DHRS	DHRL				
						LF2E-1			
						LF2E-2	P2	CSD	< 1.0 E-11
						LF2E-3			
						LF2E-4	P2	CSD	< 1.0 E-11
						LF2E-5	P2	CSD	< 1.0 E-11
						LF2E-6	P3	CD	3.2 E-11
						LF2E-7	P3	CD	< 1.0 E-11

Fig. 11.28. Event Tree for a Double Pump Loss of Flow (Group E)
with Sequence Frequency Results

EVENT TREE FOR: Double Pump Loss of Flow - Group F

31-MAY-91

Double Pump Loss of Flow - Group F	Scram signal during LOF event	Sufficient control or safety rods drop during LOF event	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
LF2F	FSIG	FROD	BPHR	DHRS	DHRL	LF2F-1			
						LF2F-2	P2	CSD	< 1.0 E-11
						LF2F-3			
						LF2F-4	P2	CSD	< 1.0 E-11
						LF2F-5	P2	CSD	< 1.0 E-11
						LF2F-6	P3	MCD	1.6 E-9
						LF2F-7	P3	MCD	6.4 E-11

Fig. 11.29. Event Tree for a Double Pump Loss of Flow (Group F)
with Sequence Frequency Results

EVENT TREE FOR: Loss of Normal Power

31-MAY-91

Loss of Normal Power	Scram signal during LOF event	Sufficient control or safety rods drop during LOF event	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
LONP	FSIG	FROD	DHRS	DHRL				
					LONP-1			
					LONP-2	P2	CSD	8.4 E-10
					LONP-3	P2	CSD	3.3 E-9
					LONP-4	P3	PED	4.5 E-7
					LONP-5	P3	PED	1.8 E-8

Fig. 11.30. Event Tree for the Loss of Normal Power with Sequence Frequency Results

Fig. 11.31. Event Tree for the Loss of Constant Power with Sequence Frequency Results

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EVENT TREE FOR: Overcooling by Primary Pump Overspeed

31-MAY-91

Overcooling by Primary Pump Overspeed	Scram signal during TOP event	Sufficient control or safety rods drop during TOP event	Secondary pump trips	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
OCPP	RSIG	RROD	SPTR	BPHR	DHRS	DHRL	OCPP-1			
							OCPP-2	P2	CSD	2.5 E-9
							OCPP-3			
							OCPP-4	P2	CSD	4.4 E-11
							OCPP-5	P2	CSD	1.8 E-10
							OCPP-6			
							OCPP-7	P2	CSD	< 1.0 E-11
							OCPP-8	P2	CSD	< 1.0 E-11
							OCPP-9	P3	ND	1.4 E-9
							OCPP-10	P3	ND	9.8 E-7

Fig. 11.33. Event Tree for an Overcooling by Primary Pump Overspeed
with Sequence Frequency Results

EVENT TREE FOR: Overcooling by Major Steam Leak

31-MAY-91

Overcooling by Major Steam Leak	Scram signal during TOP event	Sufficient control or safety rods drop during TOP event	Secondary pump trips	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
OCSL	RSIG	RROD	SPTR	BPHR	DHRS	DHRL	OCSL-1			
							OCSL-2	P2	CSD	8.4 E-10
							OCSL-3			
							OCSL-4	P2	CSD	1.5 E-11
							OCSL-5	P2	CSD	5.8 E-11
							OCSL-6			
							OCSL-7	P2	CSD	< 1.0 E-11
							OCSL-8	P2	CSD	< 1.0 E-11
							OCSL-9	P3	MCD	4.5 E-10
							OCSL-10	P3	MCD	3.3 E-7

Fig. 11.34. Event Tree for an Overcooling by Major Steam Leak
with Sequence Frequency Results

EVENT TREE FOR: Slow Reactivity Insertion - Group A ($< 0.20 \$$)

31-MAY-91

Fig. 11.35. Event Tree for a Slow Reactivity Insertion (Group A) with Sequence Frequency Results

EVENT TREE FOR: Slow Reactivity Insertion - Group B (0.20-0.65 \$)

31-MAY-91

Slow Reactivity Insertion - Group B (0.20-0.65 \$)	Scram signal during TOP event	Sufficient control or safety rods drop during TOP event	Primary pumps keep running	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
RISB	RSIG	RROD	PRUN	BPHR	DHRS	DHRL				
							RISB-1	P2	CSD	6.5 E-11
							RISB-2			
							RISB-3			
							RISB-4	P2	CSD	< 1.0 E-11
							RISB-5	P2	CSD	< 1.0 E-11
							RISB-6	P3	MCD	3.5 E-8
							RISB-7	P3	CD	7.0 E-11
							RISB-8	P3	MCD	2.5 E-8

Fig. 11.36. Event Tree for a Slow Reactivity Insertion (Group B) with Sequence Frequency Results

EVENT TREE FOR: Slow Reactivity Insertion - Group C (0.65-1.10 \$)

31-MAY-91

Slow Reactivity Insertion - Group C (0.65-1.10 \$)	Scram signal during TOP event	Sufficient control or safety rods drop during TOP event	Primary pumps keep running	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
RISC	RSIG	RROD	PRUN	BPHR	DHRS	DHRL	RISC-1			
							RISC-2	P2	CSD	< 1.0 E-11
							RISC-3			
							RISC-4	P2	CSD	< 1.0 E-11
							RISC-5	P2	CSD	< 1.0 E-11
							RISC-6	P3	CD	< 1.0 E-11
							RISC-7	P3	CD	< 1.0 E-11
							RISC-8	P3	CD	< 1.0 E-11

Fig. 11.37. Event Tree for a Slow Reactivity Insertion (Group C)
with Sequence Frequency Results

EVENT TREE FOR: Slow Reactivity Insertion - Group D (> 1.10 \$)

31-MAY-91

Slow Reactivity Insertion - Group D (> 1.10 \$)	Scram signal during TOP event	Sufficient control or safety rods drop during TOP event	Primary pumps keep running	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
RISD	RSIG	RROD	PRUN	BPHR	DHRS	DHRL				
							RISD-1	P2	CSD	< 1.0 E-11
							RISD-2			
							RISD-3			
							RISD-4			
							RISD-5			
							RISD-6	P3	CD	< 1.0 E-11
							RISD-7			
							RISD-8			

Fig. 11.38. Event Tree for a Slow Reactivity Insertion (Group D) with Sequence Frequency Results

EVENT TREE FOR: Fast Reactivity Insertion (> 1.00 \$)

03-JUN-91

Fast Reactivity Insertion (> 1.00 \$)	Sequence	Class	Fuel Damage	Frequency
RIFL				
	RIFL-1	P1	CD	2.9 E-8

Fig. 11.39. Event Tree for a Fast Reactivity Insertion (> 1.0 \$) with Sequence Frequency Results

Fig. 11.40. Event Tree for a Fast Reactivity Insertion (<0.35 \$) with Sequence Frequency Results

EVENT TREE FOR: Anticipatory Shutdown

31-MAY-91

Anticipatory Shutdown	Balance of plant decay heat removal	Decay heat removal short term	Sequence	Class	Fuel Damage	Frequency
ANTC	BPHR	DHRS				
			ANTC-1			
			ANTC-2			
			ANTC-3	P2	CSD	2.9 E-10

Fig. 11.41. Event Tree for an Anticipatory Shutdown with Sequence Frequency Results

EVENT TREE FOR: Scram

31-MAY-91

Scram	Balance of plant decay heat removal	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
SCRM	BPHR	DHRS	DHRL				
				SCRM-1			
				SCRM-2	P2	CSD	5.9 E-9
				SCRM-3			
				SCRM-4	P2	CSD	1.0 E-10
				SCRM-5	P2	CSD	4.1 E-10

Fig. 11.42. Event Tree for a Scram with Sequence Frequency Results

EVENT TREE FOR: Long Shutdown

31-MAY-91

Long Shutdown	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
SHDL	DHRS	DHRL				
			SHDL-1			
			SHDL-2	P2	CSD	8.4 E-9
			SHDL-3	P2	CSD	3.3 E-8

Fig. 11.43. Event Tree for a Long Shutdown with Sequence Frequency Results

EVENT TREE FOR: Short Shutdown

31-MAY-91

Short Shutdown	Balance of plant decay heat removal	Decay heat removal short term	Sequence	Class	Fuel Damage	Frequency
SHDS	BPHR	DHRS				
			SHDS-1			
			SHDS-2			
			SHDS-3	P2	CSD	3.5 E-9

Fig. 11.44. Event Tree for a Short Shutdown with Sequence Frequency Results

Fig. 11.46. Event Tree for the Loss of Steam Plant Heat Removal with Sequence Frequency Results

Fig. 11.45. Event Tree for the Loss of Instrument Air with Sequence Frequency Results

EVENT TREE FOR: Transient - Unrecoverable Loss of a Shutdown Cooler

31-MAY-91

Transient - Unrecoverable Loss of a Shutdown Cooler	Balance of plant decay heat removal	Decay heat removal short term (1 sd cooler)	Decay heat removal long term (1 sd cooler + shield cooling)	Sequence	Class	Fuel Damage	Frequency
TSDC	BPHR	DRSC	DRLA				
				TSDC-1			
				TSDC-2	P2	CSD	1.3 E-7
				TSDC-3			
				TSDC-4	P2	CSD	2.2 E-9
				TSDC-5	P2	CSD	1.4 E-8

Fig. 11.47. Event Tree for the Unrecoverable Loss of a Shutdown Cooler with Sequence Frequency Results

EVENT TREE FOR: Transient - Unrecoverable Loss of Shield Cooling

31-MAY-91

Transient - Unrecoverable Loss of Shield Cooling	Balance of plant decay heat removal	Decay heat removal short term (2 shutdown coolers)	Decay heat removal long term (2 shutdown coolers)	Sequence	Class	Fuel Damage	Frequency
TSHC	BPHR	DRSB	DRLB				
<div></div>				TSHC-1	P2	CSD	3.9 E-9
				TSHC-2			
				TSHC-3			
<div></div>				TSHC-4	P2	CSD	6.9 E-11
				TSHC-5			

Fig. 11.48. Event Tree for the Unrecoverable Loss of Shield Cooling
with Sequence Frequency Results

EVENT TREE FOR: Loss of Secondary System Heat Removal

05-JUN-91

Loss of Secondary System Heat Removal	Manual SCRAM Under LOHS Conditions	Decay heat removal short term	Decay heat removal long term	Sequence	Class	Fuel Damage	Frequency
TSEC	MRSS	DHRS	DHRL				
				TSEC-1			
				TSEC-2	P2	CSD	5.3 E-9
				TSEC-3	P2	CSD	2.1 E-8
				TSEC-4	P3	ND	3.5 E-6

Fig. 11.49. Event Tree for the Loss of Secondary System Heat Removal with Frequency Results

EVENT TREE FOR: Evaporator Tube Rupture

04-JUN-91

Evaporator Tube Rupture	Less than 2 DEG Failures within First 0.2 Seconds	Small or Medium Leak	Leak Detection by HMLDs	Pressure Relief System Activates	Water and Sodium Dump Before Leak Escalation	Pressure Relief by Secondary Boundary Failure	Reactor Shutdown Before IHX Failure	Sequence	Class	Fuel Damage	Frequency
EVTR	FSDG	SMLK	LKDT	SPRS	DUME	PRRE	RIHX	EVTR-1			
								EVTR-2			
								EVTR-3	P5	ND	4.0 E-8
								EVTR-4	P5	CSD	4.0 E-10
								EVTR-5			
								EVTR-6	P5	CSD	1.5 E-11
								EVTR-7			
								EVTR-8			
								EVTR-9	P5	ND	7.0 E-10
								EVTR-10	P5	CSD	< 1.0 E-11
								EVTR-11			
								EVTR-12			
								EVTR-13	P5	CSD	1.4 E-11

Fig. 11.50. Event Tree for an Evaporator Tube Rupture with Sequence Frequency Results

EVENT TREE FOR: Superheater Tube Failure

31-MAY-91

Superheater Tube Failure	Less than 2 DEG Failures within First 0.2 Seconds	Leak Escalates within First Minute	Leak Detection by HMLDs	Pressure Relief System Activates	Water and Sodium Dump Before Leak Escalation	Pressure Relief by Secondary Boundary Failure	Reactor Shutdown Before IHX Failure	Sequence	Class	Fuel Damage	Frequency
SPTR	FSDG	MODS	LKDT	SPRS	DUMS	PRRS	RIHX	SPTR-1	P5	ND	8.4 E-6
								SPTR-2	P5	CSD	8.4 E-8
								SPTR-3	P5	ND	3.3 E-8
								SPTR-4	P5	CSD	3.3 E-10
								SPTR-5	P5	ND	2.9 E-9
								SPTR-6	P5	CSD	2.9 E-11
								SPTR-7	P5	CSD	1.3 E-11
								SPTR-8	P5	ND	8.0 E-6
								SPTR-9	P5	CSD	8.0 E-8
								SPTR-10	P5	ND	3.5 E-9
								SPTR-11	P5	CSD	3.5 E-11
								SPTR-12	P5	ND	2.8 E-9
								SPTR-13	P5	CSD	2.8 E-11
								SPTR-14	P5	ND	<1.0 E-11
								SPTR-15	P5	CSD	3.2 E-8
								SPTR-16	P5	ND	
								SPTR-17	P5	CSD	
								SPTR-18	P5	ND	
								SPTR-19	P5	CSD	
								SPTR-20	P5	ND	
								SPTR-21	P5	CSD	
								SPTR-22	P5	ND	
								SPTR-23	P5	CSD	

Fig. 11.51. Event Tree for a Superheater Tube Failure with Sequence Frequency Results

EVENT TREE FOR: Local Faults: Metal Driver Breach Initiator

31-MAY-91

Fig. 11.52. Event Tree for Metal Driver Fuel Breach Local Fault
with Sequence Frequency Results

EVENT TREE FOR: Local Faults: Metal Experiment Breach Initiator

31-MAY-91

Local Faults: Metal Experiment Breach Initiato	No Pin-to-Pin Propagation	Self-Limiting Pin Propagation	Detect in Time to Shutdown and Prevent Propagation	Shutdown of Reactor	No Subassembly-to- Subassembly Propagation	Sequence	Class	Fuel Damage	Frequency
MELF	NPPE	SLPE	DETP	SHTS	NSSP				
<pre> graph LR MELF --> NPPE NPPE -- Success --> MELF1[MELF-1] NPPE -- Failure --> SLPE SLPE -- Success --> MELF2[MELF-2] SLPE -- Failure --> DETP DETP -- Success --> MELF3[MELF-3] DETP -- Failure --> SHTS SHTS -- Success --> MELF4[MELF-4] SHTS -- Failure --> NSSP NSSP -- Success --> MELF5[MELF-5] NSSP -- Failure --> MELF6[MELF-6] NSSP -- Success --> MELF7[MELF-7] NSSP -- Failure --> MELF8[MELF-8] </pre>						MELF-1			
						MELF-2			
						MELF-3			
						MELF-4	P6	CD	4.5 E-10
						MELF-5	P6	MCD	4.5 E-9
						MELF-6	P6	CD	< 1.0 E-11
						MELF-7	P6	MCD	4.5 E-11
						MELF-8	P6	CD	< 1.0 E-11

Fig. 11.53. Event Tree for a Metal Experiment Fuel Breach Local Fault
with Sequence Frequency Results

EVENT TREE FOR: Local Faults: Oxide Experiment Breach Initiator

31-MAY-91

Local Faults: Oxide Experiment Breach Initiato	No Pin-to-Pin Propagation	Self-Limiting Pin Propagation	Detect in Time to Shutdown and Prevent Propagation	Shutdown of Reactor	No Subassembly-to- Subassembly Propagation	Sequence	Class	Fuel Damage	Frequency
OELF	NPPO	SLPE	DETP	SHTP	NSSP				
						OELF-1			
						OELF-2			
						OELF-3			
						OELF-4	P6	CD	4.5 E-9
						OELF-5	P6	MCD	4.5 E-9
						OELF-6	P6	CD	<1.0 E-11
						OELF-7	P6	MCD	4.5 E-10
						OELF-8	P6	CD	<1.0 E-11

Fig. 11.54. Event Tree for a Oxide Experiment Fuel Breach Local Fault
with Sequence Frequency Results

EVENT TREE FOR: Subassembly Local Fault from Partial Blockage

31-MAY-91

Fig. 11.55. Event Tree for a Subassembly Local Fault from Partial Blockage
with Sequence Frequency Results

EVENT TREE FOR: Subassembly Local Fault from Loading Error

31-MAY-91

Fig. 11.56. Event Tree for a Subassembly Local Fault from Loading Error with Sequence Frequency Results

12.0 UNCERTAINTY, SENSITIVITY, AND IMPORTANCE ANALYSES

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12. UNCERTAINTY, SENSITIVITY, AND IMPORTANCE ANALYSES

12.1 Introduction

The previous section presented the results of the accident sequence frequencies in terms of a point mean value estimate. The certainty about these results, however, is an important issue. This section provides a summary of the analyses performed to assess the certainty about the point estimates. The analyses performed cover an analysis of the uncertainty associated with the reliability data, an analysis of the sensitivity of the results to variations in the data for some relevant basic events, and finally an importance evaluation of the basic events involved in the dominant sequences. With the exception of Table 12.3, the tables in this section provide a value labelled "MEAN" for sequence and damage bin frequencies. This mean value is that of the annual frequency distribution obtained with a Monte Carlo sampling method, and does not necessarily coincide, although it is close to, with the point mean value estimate presented in the previous section.

12.2 Uncertainty Analysis

In a PRA study, uncertainty can be divided into three principal types according to its source: data uncertainty, model uncertainty, and completeness uncertainty. Some of the uncertainties associated with a PRA can be quantified, but most of the contributions to uncertainty are not amenable to accurate quantitative estimation or calculation.

12.2.1 Data Uncertainties

The values of the input parameters are not exactly known, and probability distributions are required to describe the failure rates and unavailabilities of the systems components, the probabilities of human error and the probability of common cause failures.

Although plant-specific data has been used whenever possible in estimating the probability distributions for the failure rates and unavailabilities of the components, generic sources have been widely used.

Uncertainty is also associated with the estimated frequency of the initiating events. When these estimates have been obtained from plant records, statistical methods have been used to quantify the uncertainty, as explained in Section 5. For those initiating events whose frequency was estimated with the use of systems fault trees, the uncertainty of the basic events in the fault tree was propagated through the fault tree model using a Monte Carlo simulation.

Each basic event in the plant models has an uncertainty associated with the unreliability or unavailability that it represents. Throughout the analysis, a lognormal distribution has always been assumed for the probability of the basic events. The two parameters of choice to describe a lognormal distribution are usually a mean or a median value and the error factor, defined as the ratio of the 95th percentile of the distribution to the median. The error factors used in describing the lognormal distributions of the basic events have been listed in Section 7 along with the remaining basic event data, and are not repeated here.

The results of the uncertainty analysis presented below represent essentially the propagation of the data uncertainties through the fault tree and event tree models to quantify the uncertainty of the sequences, damage bins, and total plant damage estimates. The distributions have been estimated by using a Monte Carlo simulation of 5000 samples. In quantifying the sequence distributions, random values were drawn for all events appearing in the minimal cutsets of the sequence. Similarly, in estimating the damage bin distributions, the samples were drawn from the events distributions rather than from the sequence distributions previously estimated. Table 12.1 provides the uncertainty results by plant damage state. Table 12.2 list the uncertainty analysis results for the dominant accident sequences.

Table 12.1 Uncertainty Analysis Results by Plant Damage State and Transient Type

DAMAGE STATE		MEAN	5th	MEDIAN	95th
PLANT TRANSIENT:					
P1		$2.86 \cdot 10^{-8}$	$1.09 \cdot 10^{-9}$	$1.10 \cdot 10^{-8}$	$1.10 \cdot 10^{-7}$
P2		$2.70 \cdot 10^{-7}$	$2.31 \cdot 10^{-8}$	$1.19 \cdot 10^{-7}$	$9.28 \cdot 10^{-7}$
	P3-CD	$4.49 \cdot 10^{-8}$	$2.25 \cdot 10^{-9}$	$1.59 \cdot 10^{-8}$	$1.48 \cdot 10^{-7}$
	P3-MCD	$3.63 \cdot 10^{-7}$	$1.29 \cdot 10^{-8}$	$9.06 \cdot 10^{-8}$	$1.24 \cdot 10^{-6}$
	P3-PED	$5.68 \cdot 10^{-7}$	$1.80 \cdot 10^{-8}$	$1.77 \cdot 10^{-7}$	$2.10 \cdot 10^{-6}$
	P3-ND	$1.27 \cdot 10^{-5}$	$1.91 \cdot 10^{-6}$	$7.59 \cdot 10^{-6}$	$3.77 \cdot 10^{-5}$
P3		$1.41 \cdot 10^{-5}$	$2.11 \cdot 10^{-6}$	$8.06 \cdot 10^{-6}$	$4.18 \cdot 10^{-5}$
	P4-MCD	$2.77 \cdot 10^{-9}$	$2.23 \cdot 10^{-11}$	$4.51 \cdot 10^{-10}$	$1.02 \cdot 10^{-8}$
	P4-PED	$1.12 \cdot 10^{-8}$	$4.13 \cdot 10^{-10}$	$4.01 \cdot 10^{-9}$	$4.23 \cdot 10^{-8}$
P4		$1.40 \cdot 10^{-8}$	$8.36 \cdot 10^{-10}$	$5.96 \cdot 10^{-9}$	$5.05 \cdot 10^{-8}$
	P5-CSD	$2.04 \cdot 10^{-7}$	$1.15 \cdot 10^{-9}$	$2.77 \cdot 10^{-8}$	$7.02 \cdot 10^{-7}$
	P5-ND	$1.55 \cdot 10^{-5}$	$3.75 \cdot 10^{-7}$	$4.91 \cdot 10^{-6}$	$6.19 \cdot 10^{-5}$
P5		$1.83 \cdot 10^{-5}$	$3.96 \cdot 10^{-7}$	$5.24 \cdot 10^{-6}$	$6.44 \cdot 10^{-5}$
	P6-CD	$1.57 \cdot 10^{-8}$	$1.00 \cdot 10^{-10}$	$2.25 \cdot 10^{-9}$	$5.45 \cdot 10^{-8}$
	P6-MCD	$2.83 \cdot 10^{-8}$	$1.52 \cdot 10^{-10}$	$3.36 \cdot 10^{-9}$	$7.82 \cdot 10^{-8}$
P6		$4.80 \cdot 10^{-8}$	$3.77 \cdot 10^{-10}$	$7.30 \cdot 10^{-9}$	$1.52 \cdot 10^{-7}$
DAMAGE CLASS:					
CD		$9.36 \cdot 10^{-8}$	$1.07 \cdot 10^{-8}$	$4.84 \cdot 10^{-8}$	$2.92 \cdot 10^{-7}$
MCD		$4.45 \cdot 10^{-7}$	$1.86 \cdot 10^{-8}$	$1.14 \cdot 10^{-7}$	$1.24 \cdot 10^{-6}$
CSD		$4.44 \cdot 10^{-7}$	$3.98 \cdot 10^{-8}$	$2.00 \cdot 10^{-7}$	$1.48 \cdot 10^{-6}$
PED		$5.80 \cdot 10^{-7}$	$2.60 \cdot 10^{-8}$	$1.86 \cdot 10^{-7}$	$2.18 \cdot 10^{-6}$
DAMAGE		$1.53 \cdot 10^{-6}$	$2.46 \cdot 10^{-7}$	$8.48 \cdot 10^{-7}$	$4.74 \cdot 10^{-6}$
ND		$2.84 \cdot 10^{-5}$	$4.08 \cdot 10^{-6}$	$1.60 \cdot 10^{-5}$	$8.81 \cdot 10^{-5}$
TOTAL		$2.96 \cdot 10^{-5}$	$4.88 \cdot 10^{-6}$	$1.74 \cdot 10^{-5}$	$9.22 \cdot 10^{-5}$

Table 12.2 Uncertainty Analysis Results for the Dominant Sequences

ACCIDENT SEQUENCE	MEAN	5th	MEDIAN	95th
SPTR-2	9.02 10 ⁻⁶	1.09 10 ⁻⁷	1.92 10 ⁻⁶	3.28 10 ⁻⁵
SPTR-12	8.63 10 ⁻⁶	1.45 10 ⁻⁷	2.08 10 ⁻⁶	3.13 10 ⁻⁵
TSEC-4	3.40 10 ⁻⁶	3.99 10 ⁻⁷	1.95 10 ⁻⁶	1.04 10 ⁻⁵
LF1A-6	2.32 10 ⁻⁶	1.77 10 ⁻⁷	1.07 10 ⁻⁶	7.83 10 ⁻⁶
TBOP-4	2.15 10 ⁻⁶	2.37 10 ⁻⁷	1.20 10 ⁻⁶	6.77 10 ⁻⁶
TAIR-4	1.41 10 ⁻⁶	1.33 10 ⁻⁷	7.20 10 ⁻⁷	4.69 10 ⁻⁶
OCPP-10	1.21 10 ⁻⁶	2.09 10 ⁻⁹	6.82 10 ⁻⁸	2.95 10 ⁻⁶
LF1C-6	1.08 10 ⁻⁶	8.22 10 ⁻⁸	5.33 10 ⁻⁷	3.76 10 ⁻⁶
LF1B-6	6.83 10 ⁻⁷	3.85 10 ⁻⁸	2.96 10 ⁻⁷	2.35 10 ⁻⁶
LONP-4	4.26 10 ⁻⁷	7.68 10 ⁻⁹	1.11 10 ⁻⁷	1.70 10 ⁻⁶
OCSL-10	2.83 10 ⁻⁷	1.84 10 ⁻⁹	3.97 10 ⁻⁸	1.08 10 ⁻⁶
RISA-6	1.36 10 ⁻⁷	4.85 10 ⁻⁹	4.84 10 ⁻⁸	4.83 10 ⁻⁷
LF1D-6	1.32 10 ⁻⁷	4.42 10 ⁻⁹	4.11 10 ⁻⁸	4.71 10 ⁻⁷
TSDC-2	1.09 10 ⁻⁷	2.22 10 ⁻⁹	2.91 10 ⁻⁸	4.40 10 ⁻⁷
OCSP-10	1.03 10 ⁻⁷	2.33 10 ⁻¹⁰	7.73 10 ⁻⁹	3.30 10 ⁻⁷
SPTR-13	1.00 10 ⁻⁷	2.41 10 ⁻¹⁰	8.20 10 ⁻⁹	2.92 10 ⁻⁷
SPTR-3	9.56 10 ⁻⁸	1.94 10 ⁻¹⁰	7.30 10 ⁻⁹	3.00 10 ⁻⁷
LF2A-6	9.38 10 ⁻⁸	9.78 10 ⁻¹⁰	1.94 10 ⁻⁸	3.43 10 ⁻⁷
RISA-8	9.20 10 ⁻⁸	3.92 10 ⁻¹⁰	1.00 10 ⁻⁸	3.16 10 ⁻⁷
LF1A-8	8.20 10 ⁻⁸	9.76 10 ⁻¹⁰	1.61 10 ⁻⁸	3.33 10 ⁻⁷
LF1C-8	3.88 10 ⁻⁸	4.49 10 ⁻¹⁰	8.32 10 ⁻⁹	1.51 10 ⁻⁷
SPTR-5	3.83 10 ⁻⁸	1.87 10 ⁻¹⁰	4.71 10 ⁻⁹	1.25 10 ⁻⁷
EVTR-3	3.65 10 ⁻⁸	2.15 10 ⁻¹⁰	5.53 10 ⁻⁹	1.38 10 ⁻⁷
RISB-6	3.59 10 ⁻⁸	1.30 10 ⁻⁹	1.22 10 ⁻⁸	1.29 10 ⁻⁷
SHDL-3	3.31 10 ⁻⁸	1.65 10 ⁻⁰⁹	1.28 10 ⁻⁸	1.14 10 ⁻⁷
SPTR-23	3.21 10 ⁻⁸	2.79 10 ⁻¹¹	1.67 10 ⁻⁹	8.68 10 ⁻⁸
RIFL-1	2.92 10 ⁻⁸	1.09 10 ⁻⁹	1.08 10 ⁻⁸	1.07 10 ⁻⁷
LF1B-8	2.74 10 ⁻⁸	2.40 10 ⁻¹⁰	4.35 10 ⁻⁹	9.79 10 ⁻⁸
LF2B-6	2.18 10 ⁻⁸	6.91 10 ⁻¹⁰	7.29 10 ⁻⁹	8.06 10 ⁻⁸
RISB-8	2.14 10 ⁻⁸	9.54 10 ⁻¹¹	2.48 10 ⁻⁹	7.63 10 ⁻⁸
TSEC-3	2.10 10 ⁻⁸	9.63 10 ⁻¹⁰	7.88 10 ⁻⁹	7.32 10 ⁻⁸
LOCP-6	1.87 10 ⁻⁸	1.31 10 ⁻¹⁰	2.90 10 ⁻⁹	6.77 10 ⁻⁸
LONP-5	1.59 10 ⁻⁸	5.52 10 ⁻¹¹	1.65 10 ⁻⁹	5.54 10 ⁻⁸
TBOP-3	1.23 10 ⁻⁸	6.36 10 ⁻¹⁰	4.57 10 ⁻⁹	4.35 10 ⁻⁸
TSDC-5	1.17 10 ⁻⁸	8.63 10 ⁻¹¹	1.88 10 ⁻⁹	4.60 10 ⁻⁸
LCDL-1	1.12 10 ⁻⁸	4.25 10 ⁻¹⁰	4.29 10 ⁻⁹	4.26 10 ⁻⁸
SFPB-5	1.09 10 ⁻⁸	3.61 10 ⁻¹¹	1.28 10 ⁻⁹	4.40 10 ⁻⁸
SFPB-6	1.09 10 ⁻⁸	3.61 10 ⁻¹¹	1.28 10 ⁻⁹	4.40 10 ⁻⁸
TAIR-3	8.61 10 ⁻⁹	3.50 10 ⁻¹⁰	2.90 10 ⁻⁹	3.09 10 ⁻⁸
SHDL-2	8.23 10 ⁻⁹	1.74 10 ⁻¹⁰	1.91 10 ⁻⁹	3.03 10 ⁻⁸
SCRM-2	5.97 10 ⁻⁹	1.03 10 ⁻¹⁰	1.33 10 ⁻⁹	1.91 10 ⁻⁸

Table 12.2 Uncertainty Analysis Results for the Dominant Sequences (Cont'd)

ACCIDENT SEQUENCE	MEAN	5th	MEDIAN	95th
RIFS-6	5.29 10 ⁻⁹	5.84 10 ⁻¹¹	9.87 10 ⁻¹⁰	1.81 10 ⁻⁸
TSEC-2	5.05 10 ⁻⁹	1.00 10 ⁻¹⁰	1.25 10 ⁻⁹	1.86 10 ⁻⁸
LF1D-8	4.15 10 ⁻⁹	2.70 10 ⁻¹¹	6.32 10 ⁻¹⁰	1.60 10 ⁻⁸
SPTR-15	4.07 10 ⁻⁹	1.06 10 ⁻¹¹	3.64 10 ⁻¹⁰	1.27 10 ⁻⁸
LF1A-7	4.07 10 ⁻⁹	3.92 10 ⁻¹¹	7.84 10 ⁻¹⁰	1.58 10 ⁻⁸
OELF-4	3.82 10 ⁻⁹	3.19 10 ⁻¹²	1.77 10 ⁻¹⁰	1.11 10 ⁻⁸
OELF-5	3.82 10 ⁻⁹	3.19 10 ⁻¹²	1.77 10 ⁻¹⁰	1.11 10 ⁻⁸
TSHC-2	3.63 10 ⁻⁹	3.57 10 ⁻¹¹	6.22 10 ⁻¹⁰	1.37 10 ⁻⁸
LF1A-2	3.62 10 ⁻⁹	6.53 10 ⁻¹¹	9.12 10 ⁻¹⁰	1.39 10 ⁻⁸
RIFS-8	3.55 10 ⁻⁹	5.55 10 ⁻¹²	2.02 10 ⁻¹⁰	8.81 10 ⁻⁹
LONP-3	3.47 10 ⁻⁹	3.18 10 ⁻¹¹	5.77 10 ⁻¹⁰	1.18 10 ⁻⁸
MELF-5	3.45 10 ⁻⁹	3.19 10 ⁻¹²	1.87 10 ⁻¹⁰	1.23 10 ⁻⁸
LF2A-7	3.36 10 ⁻⁹	7.92 10 ⁻¹²	2.68 10 ⁻¹⁰	1.12 10 ⁻⁸
SPTR-8	3.28 10 ⁻⁹	6.91 10 ⁻¹²	2.59 10 ⁻¹⁰	9.76 10 ⁻⁹
SPTR-18	3.24 10 ⁻⁹	8.43 10 ⁻¹²	2.90 10 ⁻¹⁰	1.01 10 ⁻⁸
SHDS-3	3.14 10 ⁻⁹	2.96 10 ⁻¹¹	5.10 10 ⁻¹⁰	1.26 10 ⁻⁸
TBOP-2	2.75 10 ⁻⁹	6.43 10 ⁻¹¹	7.45 10 ⁻¹⁰	1.08 10 ⁻⁸
OCPP-2	2.48 10 ⁻⁹	1.17 10 ⁻¹¹	2.82 10 ⁻¹⁰	8.35 10 ⁻⁹
LF2D-1	2.42 10 ⁻⁹	2.10 10 ⁻¹¹	4.50 10 ⁻¹⁰	9.34 10 ⁻⁹
LF1C-7	2.30 10 ⁻⁹	2.12 10 ⁻¹¹	3.94 10 ⁻¹⁰	8.74 10 ⁻⁹
TSDC-4	2.13 10 ⁻⁹	8.32 10 ⁻¹²	2.14 10 ⁻¹⁰	7.76 10 ⁻⁹
LF1C-2	1.84 10 ⁻⁹	3.51 10 ⁻¹¹	4.60 10 ⁻¹⁰	6.95 10 ⁻⁹
TAIR-2	1.79 10 ⁻⁹	3.57 10 ⁻¹¹	4.59 10 ⁻¹⁰	6.85 10 ⁻⁹
LF2F-6	1.66 10 ⁻⁹	4.24 10 ⁻¹¹	5.13 10 ⁻¹⁰	6.07 10 ⁻⁹
LF1B-7	1.29 10 ⁻⁹	1.00 10 ⁻¹¹	2.21 10 ⁻¹⁰	5.22 10 ⁻⁹
OCPP-9	1.24 10 ⁻⁹	3.64 10 ⁻¹²	1.26 10 ⁻¹⁰	4.79 10 ⁻⁹
LF1B-2	1.19 10 ⁻⁹	1.67 10 ⁻¹¹	2.54 10 ⁻¹⁰	4.25 10 ⁻⁹
SFPB-8	1.09 10 ⁻⁹	3.61 10 ⁻¹²	1.28 10 ⁻¹⁰	4.40 10 ⁻⁹
LF2C-6	1.03 10 ⁻⁹	1.29 10 ⁻¹¹	2.22 10 ⁻¹⁰	3.54 10 ⁻⁹
LOCP-2	8.79 10 ⁻¹⁰	3.77 10 ⁻¹²	9.36 10 ⁻¹¹	2.88 10 ⁻⁹
LF2B-7	8.41 10 ⁻¹⁰	4.44 10 ⁻¹²	1.13 10 ⁻¹⁰	2.67 10 ⁻⁹
OCSL-2	8.26 10 ⁻¹⁰	1.11 10 ⁻¹¹	1.65 10 ⁻¹⁰	2.77 10 ⁻⁹
LONP-2	7.44 10 ⁻¹⁰	3.71 10 ⁻¹²	9.14 10 ⁻¹¹	2.75 10 ⁻⁹
EVTR-9	7.30 10 ⁻¹⁰	5.45 10 ⁻¹³	3.32 10 ⁻¹¹	2.06 10 ⁻⁹
SFLE-5	6.74 10 ⁻¹⁰	3.08 10 ⁻¹³	2.19 10 ⁻¹¹	1.38 10 ⁻⁹
SFLE-6	6.74 10 ⁻¹⁰	3.08 10 ⁻¹³	2.19 10 ⁻¹¹	1.38 10 ⁻⁹
EVTR-4	4.94 10 ⁻¹⁰	4.08 10 ⁻¹³	2.08 10 ⁻¹¹	1.08 10 ⁻⁹

Table 12.2 Uncertainty Analysis Results for the Dominant Sequences (Cont'd)

ACCIDENT SEQUENCE	MEAN	5th	MEDIAN	95th
OCSL-9	4.67 10^{-10}	3.26 10^{-12}	7.13 10^{-11}	1.69 10^{-9}
OELF-7	3.82 10^{-10}	3.19 10^{-13}	1.77 10^{-11}	1.11 10^{-9}
SCRM-5	3.71 10^{-10}	3.24 10^{-12}	6.35 10^{-11}	1.44 10^{-9}
SPTR-6	3.57 10^{-10}	3.03 10^{-13}	1.91 10^{-11}	9.40 10^{-10}
MELF-4	3.45 10^{-10}	3.19 10^{-13}	1.87 10^{-11}	1.23 10^{-9}
RISA-2	3.19 10^{-10}	2.09 10^{-12}	3.96 10^{-11}	8.06 10^{-10}
ANTC-3	3.07 10^{-10}	1.95 10^{-12}	3.97 10^{-11}	1.15 10^{-9}
OCSP-2	2.70 10^{-10}	1.34 10^{-12}	3.09 10^{-11}	9.18 10^{-10}
LF1A-5	2.62 10^{-10}	2.10 10^{-12}	4.12 10^{-11}	1.02 10^{-9}
RISA-7	2.58 10^{-10}	1.39 10^{-12}	3.76 10^{-11}	9.85 10^{-10}
LF1D-7	2.41 10^{-10}	1.27 10^{-12}	3.16 10^{-11}	8.72 10^{-10}
LF1D-2	1.97 10^{-10}	2.00 10^{-12}	3.65 10^{-11}	7.22 10^{-10}
OCPP-5	1.57 10^{-10}	3.86 10^{-13}	1.27 10^{-11}	4.65 10^{-10}
LF1C-5	1.53 10^{-10}	1.09 10^{-12}	1.96 10^{-11}	4.93 10^{-10}
LF2A-2	1.50 10^{-10}	4.95 10^{-13}	1.57 10^{-11}	5.72 10^{-10}
OCSP-9	1.48 10^{-10}	4.02 10^{-13}	1.32 10^{-11}	4.87 10^{-10}
SCRM-4	1.34 10^{-10}	3.90 10^{-13}	9.51 10^{-12}	3.25 10^{-10}
SFPB-7	1.27 10^{-10}	6.48 10^{-14}	4.63 10^{-12}	3.11 10^{-10}
LF1B-5	8.65 10^{-11}	5.70 10^{-13}	1.07 10^{-11}	3.10 10^{-10}
LF1A-4	8.36 10^{-11}	2.61 10^{-13}	6.20 10^{-12}	2.23 10^{-10}
TSHC-4	7.99 10^{-11}	1.26 10^{-13}	4.52 10^{-12}	2.05 10^{-10}
RISB-7	6.66 10^{-11}	3.49 10^{-13}	9.20 10^{-12}	2.31 10^{-10}
RISB-2	6.26 10^{-11}	5.58 10^{-13}	1.02 10^{-11}	2.12 10^{-10}
OCSL-5	5.44 10^{-11}	3.53 10^{-13}	7.37 10^{-12}	1.95 10^{-10}
LF2F-7	5.21 10^{-11}	2.63 10^{-13}	7.27 10^{-12}	1.97 10^{-10}
LOCP-5	5.17 10^{-11}	1.34 10^{-13}	4.57 10^{-12}	1.66 10^{-10}
LF2C-7	4.50 10^{-11}	9.73 10^{-14}	3.40 10^{-12}	1.40 10^{-10}
OCPP-4	4.37 10^{-11}	4.37 10^{-14}	2.00 10^{-12}	1.11 10^{-10}
LF1C-4	4.25 10^{-11}	1.18 10^{-13}	3.19 10^{-12}	1.01 10^{-10}
SPTR-16	3.90 10^{-11}	1.80 10^{-14}	1.36 10^{-12}	8.77 10^{-11}
LF2B-2	3.55 10^{-11}	3.32 10^{-13}	6.07 10^{-12}	1.28 10^{-10}
MELF-7	3.45 10^{-11}	3.19 10^{-14}	1.87 10^{-12}	1.23 10^{-10}
LF2E-6	3.18 10^{-11}	2.15 10^{-13}	4.80 10^{-12}	1.16 10^{-10}
SPTR-19	3.11 10^{-11}	1.43 10^{-14}	1.08 10^{-12}	6.98 10^{-11}
SPTR-9	2.98 10^{-11}	1.23 10^{-14}	9.80 10^{-13}	7.16 10^{-11}
OCSP-5	2.10 10^{-11}	4.13 10^{-14}	1.32 10^{-12}	5.49 10^{-11}
TSHC-5	1.90 10^{-11}	3.98 10^{-14}	1.44 10^{-12}	5.76 10^{-11}
EVTR-13	1.66 10^{-11}	2.70 10^{-15}	2.82 10^{-13}	2.85 10^{-11}

Table 12.2 Uncertainty Analysis Results for the Dominant Sequences (Cont'd)

ACCIDENT SEQUENCE	MEAN	5th	MEDIAN	95th
LF1D-5	$1.65 \cdot 10^{-11}$	$6.68 \cdot 10^{-14}$	$1.62 \cdot 10^{-12}$	$5.43 \cdot 10^{-11}$
LF1B-4	$1.61 \cdot 10^{-11}$	$5.92 \cdot 10^{-14}$	$1.69 \cdot 10^{-12}$	$5.84 \cdot 10^{-11}$
OCSL-4	$1.54 \cdot 10^{-11}$	$3.99 \cdot 10^{-14}$	$1.22 \cdot 10^{-12}$	$4.17 \cdot 10^{-11}$
RISA-5	$1.47 \cdot 10^{-11}$	$7.12 \cdot 10^{-14}$	$1.78 \cdot 10^{-12}$	$5.34 \cdot 10^{-11}$
SPTR-10	$1.43 \cdot 10^{-11}$	$1.29 \cdot 10^{-14}$	$6.80 \cdot 10^{-13}$	$3.82 \cdot 10^{-11}$
EVTR-6	$1.33 \cdot 10^{-11}$	$1.50 \cdot 10^{-14}$	$8.20 \cdot 10^{-13}$	$4.13 \cdot 10^{-11}$
SFPB-9	$1.27 \cdot 10^{-11}$	$6.48 \cdot 10^{-15}$	$4.63 \cdot 10^{-13}$	$3.11 \cdot 10^{-11}$
LOCP-4	$1.24 \cdot 10^{-11}$	$1.40 \cdot 10^{-14}$	$6.55 \cdot 10^{-13}$	$3.73 \cdot 10^{-11}$
LF2A-5	$1.06 \cdot 10^{-11}$	$1.76 \cdot 10^{-14}$	$7.07 \cdot 10^{-13}$	$3.81 \cdot 10^{-11}$
RIFS-7	$1.06 \cdot 10^{-11}$	$2.12 \cdot 10^{-14}$	$7.50 \cdot 10^{-13}$	$2.74 \cdot 10^{-11}$

12.2.2 Model Uncertainties

Model uncertainties appear with regard to how well the models developed represent the actual plant and its behavior under accident conditions. Uncertainty has been introduced during model development in the form of simplifying, conservative assumptions and approximations, properly explained in the event tree and system description sections. The models have been thoroughly reviewed by plant personnel. The use of the rare event approximation to sum large number of cutsets results in a conservative approach. Its use is also appropriate given the absence of events with large probability of occurrence. Other model uncertainties such as human errors and common mode failure are specifically addressed in the sensitivity analysis that has been performed. The results are listed in Section 12.3 below.

12.2.3 Completeness Uncertainty

This source of uncertainty deals with the possibility of using incomplete models. Commercial PRA experience could only be used to a very limited extent because of the uniqueness of the EBR-II plant. An extensive search of plant records was conducted to identify all possible initiating events, and comparisons were made with commercial nuclear plants and existing PRAs for other fast reactors. The system models were developed to the level of detail needed to identify the appropriate relationships between the modelled system and the support systems.

Exclusion of significant numbers of low-probability accident sequences or minimal cutsets has been avoided by defining very low truncation setpoints in both the accident sequence analysis and the damage bin calculations. Because of the steps taken to ensure completeness in modelling task, the PRA is deemed to be complete in addressing the scope of the project.

12.3 Sensitivity Analysis

The sensitivity of the results to several important issues has been analyzed. In particular, the cases that have been studied address the modeling uncertainties related to human error

quantification and common cause failures, the effect of the direct damage initiators, the effect of the special transient classes P5 (steam generator tube rupture) and P6 (local faults), and combinations of the different issues. The following cases have been analyzed;

- 1 Human error probability increased: the probability of the basic events representing human failures has been increased by a factor on 10.
- 2 Human error probability decreased: the probability of the basic events representing human failures has been decreased by a factor of 10.
- 3 Common cause probability increased: the probability of the basic events representing common cause failures has been increased by a factor of 3.
- 4 Common cause probability decreased: the probability of the basic events representing common cause failures has been decreased by a factor of 3.
- 5 Direct damage initiator frequency increased: the frequency of the initiating events leading to direct damage has been increased by a factor of 10.
- 6 Direct damage initiator frequency decreased: the frequency of the initiating events leading to direct damage has been decreased by a factor of 10.
- 7 Local Faults initiator frequency increased: the frequency of the local faults initiating events has been increased by a factor of 10.
- 8 Local Faults initiator frequency decreased: the frequency of the local faults initiating events has been decreased by a factor of 10.
- 9 Steam generator tube rupture initiator frequency increased: the frequency of the evaporator and superheater tube rupture initiating events has been increased by a factor of 10.
- 10 Steam generator tube rupture initiator frequency decreased: the frequency of the evaporator and superheater tube rupture initiating events has been decreased by a factor of 10.
- 11 Human error and common cause failure probabilities increased: combination of cases 1 and 3; human error probabilities increased by a factor of 10 and common cause failures by a factor of 3.
- 12 Human error and common cause failure probabilities decreased: combination of cases 2 and 4; human error probabilities decreased by a factor of 10 and common cause failure by a factor of 3.

- 13 Direct damage, local faults, and steam generator tube rupture initiator frequencies increased: combination of cases 5, 7, and 9; all of these frequencies increased by a factor of 10.
- 14 Direct damage, local faults, and steam generator tube rupture initiator frequencies decreased: combination of cases 6, 8, and 10; all of these frequencies decreased by a factor of 10.
- 15 Human error and common cause failure probabilities and direct damage, local faults, and steam generator tube rupture initiator frequency increased: combination of cases 11 and 13; common cause failure probabilities increased by a factor of 3, and the other items by a factor of 10.
- 16 Human error and common cause failure probabilities and direct damage, local faults, and steam generator tube rupture initiator frequency decreased: combination of cases 12 and 14; common cause failure probabilities decreased by a factor of 3, and the other items by a factor of 10.

A summary of the results, compared with the baseline case, is presented in Table 12.3. Only the point mean values of the damage frequency are compared in this table, while the distributions are compared in Table 12.4. Note that the mean value of the distribution does not necessarily coincide with the point mean value estimate. It must be pointed out that only the damage sequences have been included in the sensitivity calculations; the ND category has been left out.

Human Error

There are 44 different human error events in the dominant minimal cutsets of the dominant sequences, mostly from the reactor safety system and recovery actions in the decay heat removal systems. An increase by a factor of 10 in the human error probabilities results in an increase by a factor of almost 22 in the annual damage frequency. The effect is so important because several of the human failures already have a large probability in the baseline case, placing them in the most dominant cutsets. Furthermore, some of the top cutsets include two human errors, namely the failure to push the scram button and the miscalibration of the SOT

Table 12.3 Results of the Sensitivity Analysis; Comparison of Point Mean Values of the Annual Damage Frequency

CASE		FREQUENCY	BASELINE FREQUENCY	RATIO
1	Human error increased by 10	$3.43 \cdot 10^{-5}$	$1.57 \cdot 10^{-6}$	21.80
2	Human error decreased by 10	$1.12 \cdot 10^{-6}$	$1.57 \cdot 10^{-6}$	0.71
3	Common cause increased by 3	$3.50 \cdot 10^{-6}$	$1.57 \cdot 10^{-6}$	2.23
4	Common cause decreased by 3	$9.29 \cdot 10^{-7}$	$1.57 \cdot 10^{-6}$	0.59
5	Direct damage increased by 10	$1.96 \cdot 10^{-6}$	$1.57 \cdot 10^{-6}$	1.25
6	Direct damage decreased by 10	$1.53 \cdot 10^{-6}$	$1.57 \cdot 10^{-6}$	0.97
7	Local faults increased by 10	$1.95 \cdot 10^{-6}$	$1.57 \cdot 10^{-6}$	1.24
8	Local faults decreased by 10	$1.53 \cdot 10^{-6}$	$1.57 \cdot 10^{-6}$	0.97
9	Steam generator tube rupture increased by 10	$3.34 \cdot 10^{-6}$	$1.57 \cdot 10^{-6}$	2.13
10	Steam generator tube rupture decreased by 10	$1.39 \cdot 10^{-6}$	$1.57 \cdot 10^{-6}$	0.89
11	Human error and common cause increased (cases 1 and 3)	$8.59 \cdot 10^{-5}$	$1.57 \cdot 10^{-6}$	54.71
12	Human error and common cause decreased (cases 2 and 4)	$6.79 \cdot 10^{-7}$	$1.57 \cdot 10^{-6}$	0.43
13	Direct damage, local faults and steam generator tube rupture increased by 10 (cases 5, 7, and 9)	$4.11 \cdot 10^{-6}$	$1.57 \cdot 10^{-6}$	2.62
14	Direct damage, local faults and steam generator tube rupture decreased by 10 (cases 6, 8, and 10)	$1.32 \cdot 10^{-6}$	$1.57 \cdot 10^{-6}$	0.84
15	Direct damage, local faults, steam generator tube rupture, human error and common cause increased (cases 11 and 13)	$8.85 \cdot 10^{-5}$	$1.57 \cdot 10^{-6}$	56.37
16	Direct damage, local faults, steam generator tube rupture, human error and common cause decreased (cases 12 and 14)	$4.26 \cdot 10^{-7}$	$1.57 \cdot 10^{-6}$	0.27

Table 12.4 Distributions of the Sensitivity Analysis Cases

SEQUENCE	MEAN	5th	MEDIAN	95th
BASELINE	$1.53 \cdot 10^{-6}$	$2.46 \cdot 10^{-7}$	$8.48 \cdot 10^{-7}$	$4.74 \cdot 10^{-6}$
1	$2.82 \cdot 10^{-5}$	$1.81 \cdot 10^{-6}$	$1.00 \cdot 10^{-5}$	$1.07 \cdot 10^{-4}$
2	$1.12 \cdot 10^{-6}$	$1.67 \cdot 10^{-7}$	$5.77 \cdot 10^{-7}$	$3.24 \cdot 10^{-6}$
3	$3.55 \cdot 10^{-6}$	$4.60 \cdot 10^{-7}$	$1.70 \cdot 10^{-6}$	$1.03 \cdot 10^{-5}$
4	$9.13 \cdot 10^{-7}$	$1.68 \cdot 10^{-7}$	$5.30 \cdot 10^{-7}$	$2.64 \cdot 10^{-6}$
5	$1.94 \cdot 10^{-6}$	$3.89 \cdot 10^{-7}$	$1.21 \cdot 10^{-6}$	$5.47 \cdot 10^{-6}$
6	$1.51 \cdot 10^{-6}$	$2.23 \cdot 10^{-7}$	$8.10 \cdot 10^{-7}$	$4.72 \cdot 10^{-6}$
7	$1.92 \cdot 10^{-6}$	$3.10 \cdot 10^{-7}$	$1.08 \cdot 10^{-6}$	$6.02 \cdot 10^{-6}$
8	$1.51 \cdot 10^{-6}$	$2.37 \cdot 10^{-7}$	$8.13 \cdot 10^{-7}$	$4.71 \cdot 10^{-6}$
9	$3.12 \cdot 10^{-6}$	$3.48 \cdot 10^{-7}$	$1.41 \cdot 10^{-6}$	$9.67 \cdot 10^{-6}$
10	$1.39 \cdot 10^{-6}$	$2.19 \cdot 10^{-7}$	$7.45 \cdot 10^{-7}$	$4.30 \cdot 10^{-6}$
11	$6.90 \cdot 10^{-5}$	$3.46 \cdot 10^{-6}$	$2.20 \cdot 10^{-5}$	$2.62 \cdot 10^{-4}$
12	$6.68 \cdot 10^{-7}$	$1.13 \cdot 10^{-7}$	$3.77 \cdot 10^{-7}$	$2.02 \cdot 10^{-6}$
13	$4.28 \cdot 10^{-6}$	$5.94 \cdot 10^{-7}$	$2.03 \cdot 10^{-6}$	$1.26 \cdot 10^{-5}$
14	$1.28 \cdot 10^{-6}$	$1.77 \cdot 10^{-7}$	$6.56 \cdot 10^{-7}$	$3.86 \cdot 10^{-6}$
15	$7.21 \cdot 10^{-5}$	$4.58 \cdot 10^{-6}$	$2.42 \cdot 10^{-5}$	$2.58 \cdot 10^{-4}$
16	$4.14 \cdot 10^{-7}$	$6.03 \cdot 10^{-8}$	$2.30 \cdot 10^{-7}$	$1.24 \cdot 10^{-6}$

sensors. Decreasing the human failure probabilities by a factor of 10, in contrast, only reduces the damage frequency to 71% of the baseline.

Common Cause

There are 58 common cause failures that appear in the dominant cutsets of the damage sequences. For the common cause failures, the analysis is equivalent to a sensitivity study to the beta factors used in defining the probabilities of the common cause failures. The probability has been increased (case 3) and decreased (case 4) by a factor of 3 instead of 10 as in the other sensitivity cases. The factor of three has been chosen to keep the common cause failure probability of a component below the probability of independent failure (typical beta factors used were between 0.1 and 0.3). As in the case of the human failures, the results are more sensitive to increases than decreases in the probabilities of the common cause failures. An increase by a factor of 2.23 occurs when the common causes were increased by a factor of 3, and a 40 % reduction in the damage frequency is observed when the common cause is decreased by the same factor. An increase in the common cause events probabilities is fairly important because several of the common cause failures of the reactor safety system appear in the top minimal cutsets. Several top minimal cutsets are single term, common cause events cutsets. Several other dominant cutsets, the ones containing two human errors, contain also a common cause event. A disproportionately large effect in the damage frequency is observed when raising the human error probabilities because of the minimal cutsets with the two human error terms. On the other hand, lowering the common cause probability (by a factor of 3) has a larger impact on the frequency than lowering the human error probability (by a factor of 10) because the common cause affect both the single term, common cause dominant cutsets and the human error-common cause dominant cutsets.

Direct Damage Initiators

Increasing the frequency of the three direct damage initiators (RIFL, LCDL, and

LF2D) by a factor of 10 results in an increase by 25 % of the damage frequency, mostly due to the RIFL initiator, which has the largest frequency of the three. The results of case 6, however, indicate that reducing their frequency by a factor of 10 only reduces the damage frequency by about 3 %. The small effect observed by decreasing their frequency is due to the fact that, in the baseline case, the first direct damage minimal cutset (RIFL) appears with a frequency almost an order of magnitude below that of the most dominant cutsets.

Local Faults Initiators

A similar effect to that observed in the direct damage initiators is seen in the case of the local faults. Local faults sequences are about an order of magnitude in frequency lower than the dominant cutsets in the baseline case. Therefore, an increase by a factor of 10 is significant (25 % increase in damage frequency) and a decrease is essentially insignificant (3 % difference).

Steam Generator Tube Rupture Initiators

Three of the minimal cutsets for superheater tube rupture sequences (those of SPTR-3, SPTR-13, and SPTR-23) are among the dominant cutsets in the baseline case. This results in a high sensitivity of the damage frequency to changes in the frequency of the steam generator tube rupture initiators. Increasing their frequency by a factor of 10 raises the damage frequency by a factor of 2.13, while decreasing their frequency by the same amount lowers the damage frequency by a factor of 0.89. The superheater tube rupture initiator is almost entirely responsible for the changes.

Human Error and Common Cause

When combining these two types of failure the results change more considerably, especially when increasing the probability of both types of failure. The annual frequency of damage increases by a factor of almost 55 in response to increasing the human error probabilities by 10 and the common cause failures by 3, while a 60 % reduction in

damage frequency is observed when the probabilities are decreased by the same amounts. As it was noted above, both human error and common cause failures appear in the dominant cutsets of the damage sequences. Some of the top minimal cutsets combine both types of failures, and there is one particular event, miscalibration of the SOT sensors, which is a common cause human error. Consequently, the damage frequency is very sensitive to changes in the probabilities of the two types of failures.

Direct Damage, Local Faults, and Steam Generator Tube Rupture Initiators

The combination of changes in this three types of initiating event frequencies produces considerable changes in the annual frequency of damage. When increasing the initiator frequency by a factor of 10, an increase by a factor of 2.62 appears in the damage frequency. Decreasing the initiator frequency by 10, lowers the damage frequency 16 %. The superheater tube rupture initiating event dominates the changes in the damage frequency, and the sensitivity results for this combined case are close to those obtained in cases 9 and 10, where the steam generator tube rupture initiators were considered alone.

Human Error, Common Cause, and Direct Damage, Local Faults and Steam Generator Tube Rupture Initiators

This combination does not differ much from the combination cases of human error and common cause failure, only in this case the changes in damage frequency are somewhat more drastic. The human error and common cause events (but specially the human errors) are the prevalent factor in increasing the damage frequency in case 15 (by a factor of over 56). The initiator frequency changes, most notably that of the superheater tube rupture initiating event, play a more important role in case 16, when probabilities and frequencies are lowered, resulting in a damage frequency that is only 27 % of the original. This effect occurs because most of the dominant cutsets include at least one of the initiators or basic events whose frequency or probability is being decreased. The direct damage and specially the local faults initiators have a moderate effect overall.

12.4 Importance Calculations

An importance analysis has been carried out for the basic events that appear in the minimal cutsets of the dominant sequences. The importance calculations provide a ranking of the basic events ordered by their contribution to the damage frequency. The summary of the importance results appears in Table 12.5. Three measures of importance have been calculated for each basic event: the Fussell-Vesely importance (labeled F-V IMPORTANCE in the table), the risk achievement worth (RAW), and the risk reduction worth (RRW).

The Fussell-Vesely importance represents the fraction of the total damage frequency that involves the basic event. It is calculated as the difference between the damage frequency with the basic event probability at its nominal value and the frequency with the event probability set to 0.0, divided by the nominal damage frequency. The risk achievement worth importance measure represents the increase in damage frequency expected if the basic event were a true event. It is calculated as the ratio of the damage frequency with the basic event probability set to 1.0 to the frequency with the nominal event probability. The risk reduction worth importance measure indicates the reduction in damage frequency if the basic event in question were a false event. It is computed as the ratio of the nominal frequency to the damage frequency calculated with the basic event probability set to 0.0. Only the most important (Fussell-Vesely, F-V, criterion) basic events are listed in Table 12.5. Among the basic events with the highest F-V importance are the common cause and human error events, as could be expected from the sensitivity analyses. Although some human errors are more important than the common cause failures under the F-V measure, it is noticeable that the 6 events with the largest RAW measure importance are all common cause failures (all in the reactor safety system). The increase that would result in the damage frequency in EBR-II if any of this common cause events were a true event would be of the order between 10^4 and 10^5 . This very large RAW measure for these events occurs because they appear in single term minimal cutsets in several accident sequences.

Table 12.5 Results of the Importance Calculations

EVENT NAME	MEAN PROBABILITY	F-V IMPORTANCE	RAW	RRW
RHU002CL	1.000 10 ⁻¹	0.2270	3.0433	1.2937
RCZC00QU	2.400 10 ⁻⁶	0.2093	8.72 10 ⁴	1.2647
RHU302QL	2.500 10 ⁻²	0.1963	8.6572	1.2443
RREC00QU	1.920 10 ⁻⁶	0.1674	8.72 10 ⁴	1.2011
VHERIXWF	1.000 10 ⁻²	0.1049	11.3849	1.1172
DCR001OF	2.230 10 ⁻⁴	0.0762	342.6879	1.0825
RND2Z0RU	3.600 10 ⁻⁴	0.0625	174.6201	1.0667
RDC2Z1QU	3.600 10 ⁻⁴	0.0625	174.6201	1.0667
VHEMO2WF	8.400 10 ⁻¹	0.0537	1.0102	1.0567
VHEDM1WF	1.000 10 ⁻¹	0.0535	1.4813	1.0565
VHEPF2WF	5.000 10 ⁻¹	0.0510	1.0510	1.0537
VHEMO1WF	1.600 10 ⁻¹	0.0509	1.2675	1.0537
WFL095YE	2.210 10 ⁻²	0.0323	2.4310	1.0334
HFL717SE	2.210 10 ⁻²	0.0323	2.4310	1.0334
HHU01BSZ	2.000 10 ⁻²	0.0293	2.4341	1.0301
9CR001OF	1.010 10 ⁻⁴	0.0282	280.2044	1.0290
FPP112RF	1.000 10 ⁻³	0.0268	27.7885	1.0276
DCR001OE	7.440 10 ⁻⁵	0.0254	341.7316	1.0260
9HU000OL	4.000 10 ⁻⁶	0.0242	6044.3315	1.0248
9CR002OF	1.010 10 ⁻⁴	0.0217	215.4129	1.0221
RAM2Z1QU	1.200 10 ⁻⁴	0.0208	174.6252	1.0213
VHEFSDWF	3.200 10 ⁻³	0.0204	7.3446	1.0208
VHEPF3WF	1.000 10 ⁻¹	0.0204	1.1833	1.0208
FPP991RF	9.000 10 ⁻¹	0.0176	1.0020	1.0179
FPP003RF	1.000 10 ⁻¹	0.0169	1.1523	1.0172
RCZ2Z0QU	2.400 10 ⁻⁶	0.0166	6918.5205	1.0169
DCR002OF	2.230 10 ⁻⁴	0.0154	69.8630	1.0156
CSR30IRU	1.000 10 ⁻⁷	0.0151	1.51 10 ⁵	1.0153
CSR10GRU	1.000 10 ⁻⁷	0.0151	1.51 10 ⁵	1.0153
RHU001CL	5.000 10 ⁻²	0.0142	1.2707	1.0145
RRE2Z0QU	1.920 10 ⁻⁶	0.0133	6919.4282	1.0135
VOOOOOUW	1.710 10 ⁻²	0.0133	1.7618	1.0134
RBS004QP	1.000 10 ⁻³	0.0113	12.3030	1.0114
RDC2A1QW	3.600 10 ⁻³	0.0111	4.0730	1.0112
RDC2B1QW	3.600 10 ⁻³	0.0111	4.0730	1.0112

Table 12.5 Results of the Importance Calculations (Cont'd)

EVENT NAME	MEAN PROBABILITY	F-V IMPORTANCE	RAW	RRW
RDC2C1QW	3.600 10 ⁻³	0.0111	4.0730	1.0112
RND2BORW	3.600 10 ⁻³	0.0111	4.0730	1.0112
RND2CORW	3.600 10 ⁻³	0.0111	4.0730	1.0112
RND2AORW	3.600 10 ⁻³	0.0111	4.0730	1.0112
RCZ1Z0QU	2.400 10 ⁻⁶	0.0096	4019.7942	1.0097
9CR001OE	3.360 10 ⁻⁵	0.0094	280.2633	1.0095
FPP902RF	1.000 10 ⁻³	0.0084	9.3606	1.0084
RCA2Z0RU	4.800 10 ⁻⁵	0.0083	174.6974	1.0084
9CR002OE	3.360 10 ⁻⁵	0.0072	215.4595	1.0073
DHU000OL	4.000 10 ⁻⁶	0.0068	1689.7104	1.0068
FPP002RF	1.000 10 ⁻²	0.0067	1.6645	1.0068
9AM4ALAW	1.010 10 ⁻³	0.0064	7.3662	1.0065
9AM4BLQW	1.010 10 ⁻³	0.0064	7.3662	1.0065
9ARX82QW	1.010 10 ⁻³	0.0064	7.3662	1.0065
BOO99ZUW	2.000 10 ⁻³	0.0056	3.7888	1.0056
DCR002OE	7.440 10 ⁻⁵	0.0051	69.0796	1.0051
RAM2C1QW	1.200 10 ⁻³	0.0037	4.0770	1.0037
RAM2B1QW	1.200 10 ⁻³	0.0037	4.0770	1.0037
RAM2A1QW	1.200 10 ⁻³	0.0037	4.0770	1.0037
TFLJINFU	2.230 10 ⁻³	0.0033	2.4590	1.0033
FPP004RF	1.000 10 ⁻³	0.0032	4.1787	1.0032
FPP901RF	1.000 10 ⁻⁴	0.0032	32.5007	1.0032
RBS007QP	1.000 10 ⁻⁴	0.0029	29.6660	1.0029
RVI1Z0QU	6.720 10 ⁻⁷	0.0027	4026.6624	1.0027
KIN003UW	1.710 10 ⁻³	0.0025	2.4596	1.0025
RHU00ZCL	7.500 10 ⁻³	0.0022	1.2857	1.0022
DAM4ALAW	2.230 10 ⁻³	0.0020	1.9108	1.0020
DARX82QW	2.230 10 ⁻³	0.0020	1.9108	1.0020
DAM4BLQW	2.230 10 ⁻³	0.0020	1.9108	1.0020
9ARX82QP	3.000 10 ⁻⁴	0.0019	7.3698	1.0019
9RE53ACP	3.000 10 ⁻⁴	0.0019	7.3698	1.0019
RCA2AORW	4.800 10 ⁻⁴	0.0015	4.0700	1.0015
RCA2BORW	4.800 10 ⁻⁴	0.0015	4.0700	1.0015
RCA2CORW	4.800 10 ⁻⁴	0.0015	4.0700	1.0015
RBI1Z0QU	3.360 10 ⁻⁷	0.0014	4026.2332	1.0014
CCL00PRU	1.600 10 ⁻⁶	0.0012	730.6188	1.0012
EGE1EFGW	5.680 10 ⁻¹	0.0011	1.0008	1.0011

13.0 INTERPRETATION OF RESULTS

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13.0 INTERPRETATION OF RESULTS

13.1 Internal Events

The most obvious result of the EBR-II PRA is that the frequency of plant transients is low, $\sim 3.1 \cdot 10^{-5}$ per year, and less than 7% of these sequences leads to any form of damage. If EBR-II were operated only as a power reactor (i.e., without radiation experiments), then the PED damage category could be excluded and the estimated damage frequency would be further reduced.

13.1.1 Plant Transients

The total frequency of sequences arising from internal initiating events which are classified as plant transient states, i.e., P1 through P6, has been calculated to be $3.07 \cdot 10^{-5} \text{ yr}^{-1}$. However the frequency of sequences leading to all forms of damage is only $1.57 \cdot 10^{-6}$. The difference is a result of the fact that there are many sequences that lead to plant transients which can be shown not to lead to any fuel or experiment damage but are important in that they could lead to administrative shutdown of EBR-II. For example, the P3 plant transient category, which includes all of the sequences with failure to scram, has a frequency of $1.36 \cdot 10^{-5} \text{ yr}^{-1}$ and 93% of these have been identified as leading to no damage because of the inherent response of the system. Figure 13.1 shows the relative contribution of initiating events to those plant transients which lead to no damage. Almost 24% of the frequency in the no damage class comes from loss of heat sink events (TSEC, TBOP, TAIR). A total of 18% of the initiating events are faults on a single pump, (LF1A, LF1B, LF1C and OCPP), while a further 56% arise from a steam generator tube rupture. Two sequences which result from superheater tube rupture, SPTR-2 and SPTR-12, which model a propagating sodium-water reaction which does not fail the IHX due to overpressure, are included as plant transients, P5, with frequencies of $8.4 \cdot 10^{-6}$ and $8.0 \cdot 10^{-6}$ respectively, but assigned to the no damage category.

FUEL DAMAGE: ND

Fig. 13.1. Relative Contribution of Initiating Events to the No Damage Category

There is a threat to the IHX in these sequences but the threat is stress corrosion cracking due to unreacted water, sodium hydroxide etc. remaining in the IHX for a few hours. The system is shutdown and no route to core damage can be identified.

13.1.2 Damage Sequences

Figure 13.2 gives the relative contribution of initiating events to all forms of damage. The largest contributor is the loss of normal power, LONP, followed by overcooling from a major steam leak (OCSL) and the rupture of a superheater module tube. The direct damage events do not appear to be significant. The only event which does appear, representing 1.9% of the risk, is RIFL, the fast reactivity insertion which occurs only because of the contribution of the argon overpressurization event. The loss of normal power appears by virtue of inclusion of the Potential Experiment Damage Category and would not lead to damage in a power reactor.

Figures 13.3 to 13.6 give the relative contribution of initiating events to each damage category. Figure 13.3 shows that the loss of a shutdown cooler and the superheater tube rupture are the dominant contributors to the CSD category. The long shutdown is also important, because the secondary sodium is drained and unavailable. Figure 13.4 gives the contribution of initiating events to the Potential Experiment Damage Category. Damage to driver fuel is precluded and there is at most damage to one assembly occurring in these sequences. The major contributors to this category are the loss of normal power, ~80%, and a double pump failure (LF2A) ~17%. Events in this class would not lead to radionuclide release of any consequence but would be undesirable from a plant viewpoint and would certainly lead to a lengthy administrative shutdown.

DAMAGE SEQUENCES

Fig. 13.2. Relative Contribution of Initiating Events to All Damage Sequences

FUEL DAMAGE: CSD

Fig. 13.3. Relative Contribution of Initiating Events to the CSD Category

FUEL DAMAGE: PED

Fig. 13.4. Relative Contribution of Initiating Events to the PED Category

Minor Core Damage, Fig. 13.5, is dominated by an overcooling event, OCSL, $\sim 77\%$, and a reactivity insertion sequence. The overcooling sequence has been modeled without recovery being applied. This accident is slowly developing, where uncontrolled release of steam extracts sufficient energy to cool the primary tank which causes the reactor power to rise until some damage cannot be ruled out. Simply deenergizing the secondary pump is sufficient to prevent damage.

Core Damage, Fig. 13.6, is not dominated by any single initiating event: two double clutch failures (LOCP and LF2B), the RIFL direct damage sequence, and local faults (SFPB and OELF) contribute similar fractions. However the frequency of the class is $9 \cdot 10^{-8} \text{ yr}^{-1}$, which is very low.

13.1.3 General Insights

Generally the reactivity insertion events have been found to be fairly unimportant, with the plant transient categories being dominated by pump faults and decay heat removal faults. Direct damage events and local faults are not significant, however the other special plant transient category, evaporator and superheater tube rupture, has significant implications. Furthermore all three of these families of sequences are quantified in essentially a subjective manner relying on expert judgement and simple calculations for end state assignation and are therefore more uncertain.

The important systems from a mitigative point of view are the scram system and shutdown coolers and therefore the postulated common cause and human error events are most important in these systems.

FUEL DAMAGE: MCD

Fig. 13.5. Relative Contribution of Initiating Events to the MCD Category

FUEL DAMAGE: CD

Fig. 13.6. Relative Contribution of Initiating Events to the CD Category

13.1.4 Sensitivity

Since the overall frequency of damage is sufficiently small and there is a lack of dominant event, the sensitivity analysis assumes a complexity that is important in interpreting the results.

Fortunately, sensitivity studies show a very weak dependence on initiating event frequencies for those direct damage events which are difficult to quantify. There is a large sensitivity to human error events which arises mainly through manual scram actions. Common cause is extensively modeled in the scram system which affects the ATWS sequences and thus tends to be very important. The numerical sensitivity studies undertaken and the importance calculations show that results are not very sensitive to local faults, while the superheater tube rupture event has a more significant impact. In general, the sensitivity to those events whose initiating event frequencies were subjectively evaluated is not significant.

Significant human errors (of omission) have been identified, most notably the failure to manually scram, miscalibration of the SOT set points, and recovery failures in the heat removal systems. The SOT setpoint miscalibration can be dealt with by an improved procedure. Common cause effects are important in the scram system especially through the relays. These are examined in the sensitivity studies of Section 12. The results of that study indicate a high sensitivity to the combination of human error and common cause because of the existence of cutsets containing two human errors and a common cause.

13.1.5 Comparison to Other PRAs

13.1.5.1 Clinch River Breeder Reactor (CRBR)

The CRBR PRA [13.1] found only two classes of sequence above a 10^{-8} cutoff for the annual frequency, the protected and unprotected Loss of Heat Sink transients at $1.4 \cdot 10^{-6}$ and $2.5 \cdot 10^{-7} \text{ yr}^{-1}$ respectively. The EBR-II PRA has a detailed operating history to draw upon for fault characterization whereas CRBR was never built. For EBR-II the

scram system has estimated probabilities of failure to scram of $\sim 5 \cdot 10^{-6}$ per demand. In contrast, for the CRBR, diverse and redundant primary and secondary scram systems were used with the result that the total failure to scram was estimated to be $< 10^{-7}$ per demand and therefore only frequent initiating events, i.e. heat sink faults, gave any contribution above the cutoff of 10^{-8} yr^{-1} when unprotected. The contribution from protected heat sink faults is comparable in CRBR and EBR-II. CRBR did not consider local faults except as a means of reactivity insertion, and the whole category of Potential Experimental Damage does not apply to CRBR because it was to be a power reactor. With these facts in mind the results of the two PRAs are broadly consistent.

13.1.5.2 NUREG-1150

Core damage frequencies for the EBR-II are compared with relevant results from NUREG-1150 [13.2] in Figs. 13.7 and 13.8. It can readily be seen that the core damage frequency is in the same range as the two BWR plants, Peach Bottom and Grand Gulf, and significantly below the other plants. This comparison may be misleading because of the different definitions of damage applied and the existence of conservatism in the EBR-II results. Further detailed analysis can be expected to reduce the EBR-II value whereas the NUREG-1150 results have been extensively reviewed and are considered to be final.

13.1.5.3 Other USDOE Reactors

PRAs have been issued for HFIR, ATR, N-Reactor, SRP K-Reactor, and HFBR, results have been presented in [13.3, 13.4, 13.5, 13.6, 13.7, and 13.8]. The results for these plants give the frequency of core damage from internal events in the range of $4 \cdot 10^{-4}$ to $7 \cdot 10^{-5} \text{ yr}^{-1}$. In many of the Category A reactors direct damage events, through failure of pressure vessels, has been found to be important. These types of events have been found to be unimportant in EBR-II and therefore the lower frequencies described here should not be surprising.

Comparison of EBR-II Damage Frequency
with Core Damage at Commercial LWRs
(LWR data from NUREG-1150)

Fig. 13.7 Comparison of Internal Core Damage Frequency Distributions

COMPARISON OF EBR-II WITH NUREG-1150 LWRs

Fig. 13.8. Comparison of Internal Core Damage Frequency Ranges (5th and 95th percentiles) from NUREG-1150 and EBR-II

13.1.6 Conclusions

The results presented above describe a very low risk for operation of EBR-II when internal events alone are considered. This is not particularly surprising since EBR-II has a mission which is directly safety related and much effort has been expended to ensure that EBR-II is demonstrably safe. In fact EBR-II is ANL's prototype for the Integral Fast Reactor which bases part of its appeal on its passive safety characteristics.

13.2 Seismic Event

Not included at this time.

13.3 External Events (non-seismic)

A screening analysis of external (non-seismic) events has identified three sequences of note. Two sequences arise from a rupture in a NaK-filled shutdown cooler under the deck plates, SDFR-3 and SDFR-6. In SDFR-6 a rupture, rather than a small leak, is presumed to lead to a large liquid metal fire and the operators dump the secondary sodium after shutting down the reactor. It is assumed that the fire renders the remaining shutdown cooler less reliable. The result is a sequence where the 1000°F structural limit is exceeded. Clearly a large fire is required. The sequence has an estimated frequency of $2.1 \cdot 10^{-6} \text{ yr}^{-1}$. SDFR-3 is an unsuppressed leak. The operators dump secondary sodium in this case also and cooling failure is due to failure of the remaining cooler. This sequence frequency is $5.8 \cdot 10^{-9} \text{ yr}^{-1}$. A similar sodium fire sequence arises from a rupture of secondary sodium piping within containment. The fire is assumed to disable one shutdown cooler and render the other cooler less reliable. The frequency of this sequence, SSFR-6, is calculated to be $1.5 \cdot 10^{-6} \text{ yr}^{-1}$. The external events analysis was done as a screening analysis to identify the important sequences using generic numbers. Detailed analysis is required to refine the quantification and to better establish the relative importance of these sequences. It should be noted that the external events sequences identified here dominate the internal events contributions to damage frequency.

3.4 Summary

Table 13.1 compares published results from PRAs on USDOE Category A reactors. Where comparisons can be made it can be seen that EBR-II compares favorably with the other facilities. Comparisons of this sort should be viewed with caution as the widely varying reactor types and definitions of damage could lead to a bias in the results. Nevertheless, the favorable comparison is not surprising as the EBR-II mission has been centered upon demonstrating the safety of LMRs for commercial application and EBR-II is viewed as a prototype of a particular design concept for LMRs, the Integral Fast Reactor or IFR.

Table 13.1 Mean Core Damage Frequencies for USDOE Facilities

Reference		Mean Core Damage Frequencies (Yr. ⁻¹)					
		Total Internal	Seismic	Fire	Wind/ Tornado	Total External	Total
	EBR-II	1.6 10 ⁻⁶		3.6 10 ⁻⁶	-		
[13.4]	ATR	1.8 10 ⁻⁴	-	-	-	-	-
[13.5]	HFBR	3.5 10 ⁻⁴	-	-	-	-	
[13.3]	HFIR	3.1 10 ⁻⁴	1.2 10 ⁻⁴	1.9 10 ⁻⁵	2.9 10 ⁻⁴	4.3 10 ⁻⁴	7.4 10 ⁻⁴
[13.6]	N-Reactor	6.7 10 ⁻⁵	1.7 10 ⁻⁴	1.7 10 ⁻⁵	-	1.9 10 ⁻⁴	2.5 10 ⁻⁴
[13.7] [13.8]	SRP K-Reactor	2.1 10 ⁻⁴	1.2 10 ⁻⁴	1.4 10 ⁻⁷	-	2.2 10 ⁻⁴	4.3 10 ⁻⁴

13.0 References

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14.0 SUMMARY AND CONCLUSIONS

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14.0 SUMMARY AND CONCLUSIONS

14.1 Summary of Key Results

The most obvious result of the EBR-II PRA is that the risk arising from the operation of EBR-II is very low. This result holds when both internal and external events are considered. While this reflects a great deal of credit upon those responsible for the design and operation of EBR-II, the results of the PRA can be used to determine where improvements might be made that could improve the risk profile for the plant. Potential improvements can then be evaluated for cost, amount of risk reduction and for any reduction in plant availability that might arise. Furthermore the insight provided will assist in applying the lessons learned from this study to future IFR design studies.

This discussion identifies those characteristics of EBR-II that result in the low estimate of risk found in this study.

- (1) Initiating Event Frequency - The initiating event frequency is low. Roughly four initiating events per year require a scram, manual or automatic, and the scram reliability has been estimated to be between 5×10^{-6} and 10^{-5} per demand dependent on the initiating event.
- (2) Reactivity feedbacks for those accident sequences in which failure to scram occurs - The results clearly demonstrate the effects of the reactivity feedbacks in limiting the temperature excursion in most of the more likely ATWS events - 93% do not lead to any form of damage.
- (3) Decay Heat Removal - EBR-II possesses multiple passive decay heat removal paths and a large heat capacity. Therefore, in contrast to more conventional reactor systems, the loss of decay heat removal systems is only a small factor in the EBR-II risk profile.

- (4) Support Systems - The loss of support systems is relatively unimportant. Loss of Normal or Constant Power systems do challenge the reactor shutdown system to some degree. There is no dependence on electrical power for short term decay heat removal. For long term decay heat removal (> 14 days) mitigating systems do require electrical power.
- (5) Human Factors - The sensitivity to human error is low because there are relatively few actions required of the operator. Latent errors identified involve maintenance and the improper setting of trip points, however the risk can be reduced with straightforward procedural modifications.
- (6) External Events - The sensitivity to a wide range of external events is low because of the wide physical separation between the means of decay heat removal and their relative robustness in the face of external hazard. No external events have been identified which can lead to failure to scram.

In general terms, the risk of operation of EBR-II has important contributions from the relatively rare events of large consequence whose estimated frequency is commensurate with failure-to-scram probabilities. These events, such as primary tank failure or steam generator tube ruptures, have frequencies whose estimation is at best subjective and open to debate. Therefore the engineering approach to these events should not be solely probabilistically based but more in the traditional defense-in-depth manner, including inservice inspection, detectability (ideally on precursor events) and automatic protection (scram, sodium and water dumps, etc.).

One class of events which show up as contributors to risk are the so-called Local Faults due to performing experimental irradiations in EBR-II. However, it is not that these experiments are particularly risky in themselves, it is more that the overall level of risk for EBR-II is low and the uncertainty on the physical phenomena involved in these experiments leads to some level of risk. Two points need to be made here. Firstly, EBR-II is a research reactor and these experiments are being performed to investigate many of the uncertain phenomena. Secondly, the results are uncertainty driven, and should be compared to the other class of risk whose

numerical estimation is uncertainty driven, the seismic risk. When viewed in that context, the risk from these experiments is very low.

14.2 Conclusions

The conclusions reached by the Level 1 study are:

- The frequency of damage at EBR-II is generally lower than that reported for commercial LWRs and the USDOE Category A reactors.
- There is no single dominant source or cause of the various accident sequences.
- The passive decay heat removal function and the ability of reactivity feedbacks to limit the temperature excursion in a transient lead to low risk.
- The passive decay heat removal function and ability of reactivity feedbacks to limit temperature excursion in a transient also lead to a sparse set of requirements for active safety systems or operator intervention.
- EBR-II is relatively insensitive to human error and only a few important human actions were identified.
- EBR-II has little or no requirement for electrical power to prevent fuel damage.

Appendix A
AN ASSESSMENT OF RISK
DURING CORE REFUELING
AT EBR-II

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An Assessment of Risk during Core Refueling at EBR-II

1.0 Introduction

The Experimental Breeder Reactor II (EBR-II) is a liquid-metal cooled, fast neutron nuclear reactor located on the Idaho National Engineering Laboratory [A.1, A.2]. The driver fuel for EBR-II is a uranium alloy encapsulated in stainless steel. However, because the reactor is an experimental facility, it routinely contains a complement of both fueled and structural experiments of widely varying compositions. The heterogenous nature of the core results in individual subassembly reactivity worths that vary from slightly negative for some structural experiments to approximately 1.8\$ positive for core drivers being loaded in central positions. A typical core is about 6-1/2 rows across and contains 100 subassemblies.

During the core refueling process, the reactor is taken subcritical by fully withdrawing and unlatching the fueled control rods. Typically, the reactor is five to eight dollars subcritical during refueling. Two safety rods, having a combined worth of $\sim 1\$$, are fully inserted in the core during the refueling phase and are automatically tripped out by a two-out-of-three neutron detector system should the refueling process cause the system state to approach too close to critical. Forced convective cooling of the reactor by the primary pumps is not allowable during refueling so achieving criticality under these conditions is very undesirable.

The purpose of this document is to present an assessment of the risks associated with the core refueling process in EBR-II. A simplified event tree and its corresponding fault trees using the methods of reference [A.3] are developed and analyzed to yield an estimate of the overall probability of sustaining an inadvertent criticality, and therefore possible core damage, during a reactor operating year. The systems that support core refueling will be described to the extent necessary to impart a basic understanding of the process.

2.0 The Core Refueling Process

2.1 General

The fuel handling system includes the various mechanisms required for the remote manipulation of reactor subassemblies in the reactor vessel, together with the control equipment necessary for the safe operation of the mechanisms. The fuel-handling operation involving the movement of the subassemblies between the reactor and the storage basket is termed unrestricted fuel-handling. Unrestricted fuel-handling requires that the reactor be shut down. The operations are performed within the primary tank with the subassemblies completely submerged in sodium.

The principal components used for core refueling are the storage basket, the transfer arm, and the core subassembly gripper and holddown mechanisms, which are positioned by two rotating plugs. Figure A.1 displays the global arrangement of these components on a cross section of the primary system. Brief descriptions of the major components follow.

2.1.1 Gripper Mechanism

The gripper mechanism is the electro-mechanical component used to insert or to extract subassemblies from the reactor vessel grid. It is mounted on the small rotating shield plug and extends through a nozzle in the small rotating plug into the sodium in the primary tank, and through a sleeve in the reactor vessel top cover into the reactor vessel. The gripper mechanism grasps the upper adaptor of the subassembly and moves the subassembly vertically out of or into the reactor vessel. The gripper cannot physically engage more than one subassembly at a time.

Figure A.1 - Fuel-Handling Mechanisms In Primary Tank

2.1.2 Holddown Mechanism

The holddown mechanism is an electro-mechanical device which operates in conjunction with the gripper mechanism. It is also mounted on the small rotating shield plug, and extends through a nozzle into the reactor vessel. During the insertion or removal of a subassembly from the reactor vessel, the holddown mechanism functions as a guide funnel and prevents inadvertent extraction or movement of adjacent subassemblies. The holddown mechanism also functions as a guide sleeve to steady the subassembly while it is being moved through the primary tank bulk sodium to the transfer position by the rotating plugs.

2.1.3 Large and Small Rotating Shield Plugs

Two rotating shield plugs, the large plug and the small plug, are located in the top part of the primary tank cover. The large plug is mounted in the primary tank cover and supported by the T-1 central ring with its vertical axis coinciding with the vertical axis of the reactor core. The small plug is positioned eccentrically within the large plug with the small plug vertical axis offset 15.2 in. from the large plug vertical axis. Both plugs rotate about their respective vertical axes and serve two basic purposes:

- (1) Provide a means of positioning the gripper mechanism over any desired point on the reactor when extracting or inserting any of the 637 subassemblies in the reactor vessel,¹ and to position the gripper to the operating position or the transfer position.
- (2) To position the control rod drives over their respective subassemblies for reactor operation.

An important function of the two plugs is maintaining a positioning accuracy of 1/64 in. by either rotating plug with respect to any of the 637 grid locations.

¹ Included are stainless reflector assemblies and breeder blanket assemblies.

2.1.4 Transfer Arm

The transfer arm is an "L" shaped structure that is manually rotated to transfer subassemblies between the core gripper and the storage basket. Transfer arm motion is both vertical and horizontal. The arm rotates through an arc of approximately 180° and a vertical motion of 12 in. to allow the fuel handling operator to make manual contacts between the transfer arm and associated equipment. The fuel handling area traversed by the transfer arm carrier block (outside the reactor vessel wall) is provided with a subassembly catch basin.

2.1.5 Storage Basket

The storage basket is a cylindrical structure with two basic purposes: (1) provide storage for irradiated subassemblies allowing radioactive decay and decay heat reduction prior to transfer from the primary tank, and (2) provide storage of nonirradiated subassemblies prior to installation in the reactor grid [A.2].

The basket contains 75 storage tubes arranged in three concentric rows. The outer two rows each contain 30 tubes and the inner row contains 15.

The basket has both rotary and vertical drive systems. The rotary drive has 30 positions, one for each storage location (only even location numbers are used for the inner row). The vertical drive is provided with numerous stop elevations to facilitate receiving or transferring subassemblies.

The basket is wholly contained within the primary tank, and the subassemblies are cooled by conduction and natural convection to the pool sodium.

2.1.6 Description of a Transfer

To transfer a subassembly from the core to the storage basket, the gripper and holddown are positioned over the proper core location. The holddown is lowered around the adapter of the subassembly being removed. The gripper is then lowered onto the upper adapter of the subassembly in the holddown funnel and the gripper jaws are closed. The holddown, as its name implies, prevents any adjacent subassemblies from being lifted with the one being raised by the gripper. When the gripper engagement is properly effected, the sense rod within the gripper shaft is lifted approximately 3/4 in. This motion is indicated on the fuel handling console. The gripper jaws are then closed by vertical shaft motion. This information is displayed to the fuel handling operator by an indicator on the fuel handling console.

The gripper is raised to its full-up position, approximately 10 ft. above the core, with the subassembly removed from the reactor vessel, but still completely immersed in sodium. The holddown mechanism is then raised to its up position and serves as a guide for the lower portion of the subassembly during plug rotation. The shield plugs are rotated to the "transfer" position where the subassembly is picked up by the transfer arm. The transfer arm is manually rotated to engage the lower (rectangular) portion of the subassembly upper adapter and then locked to the subassembly. A "wiggle test" is performed at this point to assure that a proper transfer has been made. When the transfer verification is complete, the gripper jaws are opened and the transfer arm is lowered to remove the subassembly adapter from the gripper jaws.

To complete the transfer of the subassembly to its programmed storage basket position requires rotation of the transfer arm to one of three storage basket rows. The basket is rotated to the coordinates programmed in the Numerically Controlled Positioning System, and then raised until the subassembly's lower adapter almost enters the storage hole. The subassembly is started into the storage hole manually by lowering the transfer arm, and then the basket is raised until the subassembly is seated. Another wiggle test verifies that the transfer is complete, and the transfer arm is unlocked and moved clear of the subassembly. This process is reversed to remove a fresh subassembly from the storage basket and transfer it to the core. Under optimum conditions, one complete interchange can be completed in about 45 minutes.

Certain special operations are also performed with the fuel handling equipment. These operations include removal and replacement of safety rods, safety and control-rod thimbles, and neutron source rods. These operations involve manual rotation of the core gripper to lock and unlock bayonet fittings in the reactor grid.

Two important facts should be emphasized at this point. The gripper mechanism is the only device capable of engaging and removing a subassembly from, or inserting one into, the reactor grid. Its design prevents it from engaging more than one subassembly. In fact, the capacity of all of the EBR-II equipment that transfers subassemblies is limited to a single subassembly.

2.2 Fuel Handling Control System

The fuel handling control system consists of the fuel handling control console, the interlock and repetitive motion controls, and the numerically controlled positioning system. Also part of the fuel handling control system are the limit switches and relays that are integrated with the various fuel handling mechanisms, the motors to operate the mechanisms, and the motor controllers and contactors that control operation of the mechanism's motor.

The purpose of the fuel handling control system is to coordinate the operation of all associated fuel handling mechanisms in a sequentially prescribed manner when removing or replacing reactor subassemblies. The system functions to automatically direct and limit the mechanical motions initiated by an operator at the fuel handling control center to prevent a potentially dangerous or damaging condition from developing.

The EBR-II fuel handling operation is essentially an automatically sequenced series of repetitive operations. The basic control philosophy consists of a permissive system. The operator must initiate each step in the process from the fuel handling control console, or refueling machine control panel, but the control system will permit the step to proceed only if all prerequisite conditions have been satisfied.

In the event an abnormal condition develops, the causative action is stopped automatically and in many cases an identifying alarm light is illuminated. Overall supervision of the fuel handling operations is effected by a qualified fuel handling operator at the fuel handling console.

Nuclear instrumentation is provided for automatically scrambling the safety rods during fuel handling operations if the neutron flux level exceeds a preset maximum, or if the reactor period is less than a preset minimum. A trip from the neutron monitors also arrests the vertical motion of the core gripper, thereby stopping any fuel movement. The neutron flux alarms if the signal on any one of the three neutron channels exceeds the trip setpoint. The alarm provides an additional alerting device that signals a low subcriticality margin. The neutron flux level is also visually monitored during the fuel handling operation by means of the log count rate recorder located on the fuel handling control console. Safety rod scram can be manually initiated if necessary by a pushbutton on the fuel handling console or on the main control room console. Additionally, after each subassembly insertion in the reactor grid, an operator in the reactor control room takes three-minute counts on all three nuclear channels and records, plots, and interprets the results. He will notify the Shift Supervisor if any unusual condition is observed. The fuel handling operator is provided with a specific count rate threshold at which he is instructed to terminate fuel handling.

A large part of the fuel handling control system is concerned with control of the angular position of the large rotating plug, small rotating plug, and gripper; the angular positioning and elevation of the storage basket; and the manual positioning of the transfer arm over the storage basket or under the core gripper. This part of the system is known as the Numerically Controlled Positioning System (NCPS). It positions and supervises the positioning of the plugs, gripper, and basket in accordance with coordinate information available in a reference library.

2.2.1 Numerically Controlled Positioning

The major components of the NCPS are: a microprocessor computer, plug and gripper position synchro-transmitters, synchro-to-digital converters, digital-to-analog converters, drive motor controllers, and drive motor units.

Rotating plug and gripper (plus storage basket, as appropriate) command coordinates are transmitted to the microprocessor computer by the FHANDLE program on the Data Acquisition System Computer (DAS). Analog output (digital-to-analog) units provide signal voltages to the motor controllers for the plug and gripper drives to position the core gripper over the prescribed grid position.

The FHANDLE program on the DAS provides input (type of transfer, sequence number, coordinates, basket position etc.) to the NCPS computer. This program also provides a check of each fuel handling sequence to verify it can be performed, maintains a record of the current reactor and basket loadings, and provides a record of each transfer (including time and operator name). The computer operations consist primarily of three basic functions: (1) reading input data from the DAS FHANDLE program, (2) measuring and controlling the rotation (and elevation) of the storage basket, and (3) controlling the angular rotation of the rotating plugs and the core gripper.

2.2.2 Control & Verification

In fuel handling Sequence B, the operator tells the DAS to transmit the proper coordinates to the NCPS computer. These data are then displayed on the fuel handling console. The plugs and gripper are then rotated to the destination angles. When the plugs and gripper reach the destination angles, a green completion light is received. The operator then visually verifies that the location and destination displays are correct. The actual angles of the plugs and gripper are then manually checked and compared to the displayed angles on the fuel handling console. If these agree, the fuel handling console operator then enters the angles into the terminal and these data are sent to DAS. A coordcheck subroutine in the

FHANDLE program verifies that the coordinates are correct. An interlock prevents further fuel handling until this verification sequence is complete.

The same routine is used to place a new subassembly in an open core position during Sequence G of fuel handling.

2.3 Preparation for Fuel Handling

Preparation for unrestricted fuel handling involves two major efforts. The first is to shut the reactor down and place the reactor and the primary system in the proper configuration for fuel handling. The second is the preparation, verification, and approval of the fuel handling loading plan. The following two subsections describe these processes.

2.3.1 System Preparation

The reactor is shut down by lowering all of the fueled control and safety rods to their least reactive (down) positions. During this process, the Operations Physicist establishes the degree of subcriticality of the reactor.

Immediately after shut down, the reactor is prepared for core refueling. A fuel handling checksheet is completed and fuel handling interlocks are checked. To prepare the reactor vessel for fuel handling, the control rods are released from their drive mechanisms. All 12 drives are then raised 3 in. to preclude their interference with the subassembly upper adapters when the shield plugs are rotated.

The reactor vessel cover is unlocked and raised until three torque pins mounted to the top of the cover engage sockets on the underside of the small rotating plug. Each rotating plug employs a eutectic alloy freeze seal that must be fully molten before plug rotation can be attempted. To fulfill the required interlocks, the safety rods are driven into the core to their maximum reactivity position ($\sim 1\%$). When the above conditions have been met and the gripper and holddown operation have been checked, fuel handling may be started.

2.3.2 Loading Plan

Conducting refueling requires the preparation of a carefully considered loading plan, which results in the creation of loading instructions. A basic tenet of core refueling is that subassemblies are always replaced on a one-for-one exchange basis except during replacement of control and safety rod thimbles, when up to six fueled subassemblies may be removed.

A flow chart for creation of loading instructions [A.4] is shown in Figure A.2. Reactivity predictions used in developing a loading plan are made by computing changes from the known conditions of the previous loading. This minimizes the reliance placed on predictive capabilities and thereby reduces the chances of a serious reactivity prediction error. The Chief Physicist has the responsibility for developing the loading plan. In the course of this work he predicts the reactivity at each step of the loading and the Operations Physicist assures that Technical Specification restrictions on reactivity are not violated. The Fuel Management Supervisor prepares the unrestricted-loading instructions which are reviewed by the Chief Physicist, the Operations Physicist, and the Reactor Shift Supervisor, and approved by the Reactor Neutronics Manager and the Plant Operations Manager. Restricted-loading instructions are reviewed by the Reactor Shift Supervisor and approved by the Plant Operations Manager.

A flow chart for implementation of the loading instructions [A.4] is shown in Figure A.3. The steps shown there apply to both restricted and unrestricted refueling. Several operations personnel are involved in the process to provide built-in cross checks. The procedure requires a number of sequential steps and intermediate checks, e.g., the preprogramming of instructions before the actual loading is done. These features preclude a single operation from causing a series of errors. The process of transferring the actual coordinate locations via a hard-wired computer eliminates operator input errors.

Figure A.2 - Creation of Loading Instructions

Figure A.3 - Execution of Loading Instructions

2.4 Summary Observations on Refueling

This section summarizes the basic principles, facts, and crosschecks attendant to the EBR-II core refueling process.

- (1) Only one subassembly can be handled at a time.
- (2) Subassemblies are replaced on a one-for-one exchange basis, i.e., multiple consecutive removals are not allowed. Even if an insertion positioning error is made, it is not possible to insert the subassembly, (with the exception of control and safety rods) because the position already contains an assembly.
- (3) Neutron count rate surveillance is formally required of the reactor console operator after each subassembly insertion. The operator is required to notify the fuel handling console operator and Shift Supervisor to terminate fuel handling if the count rate ratios fall to a prescribed standard. The fuel handling console operator also checks neutron count rates.
- (4) The loading plan is prepared by the Chief Physicist and checked by the Operations Physicist.
- (5) After the actual loading instructions are prepared by the Fuel Management Supervisor, they are checked by the Chief Physicist, the Operations Physicist, and a Reactor Shift Supervisor.
- (6) The loading instruction is approved by the Reactor Neutronics Manager and the Plant Operations Manager.
- (7) The loading instruction is computer-checked by a standing program - FHAN - to ensure that physically improper placements are not undetected.
- (8) After these checks, the approved loading instruction is, by the process, already stored in the DAS computer.
- (9) Actual core destination coordinates are withdrawn from a standard reference library and transmitted down to the NCPS by a hard-wired computer connection from the DAS.

- (10) After each subassembly transfer, multiple cross checks of the core position, both by the fuel handling computer and by the operator, must be satisfied before further fuel handling steps are allowed.
- (11) An automatic system is in place to prevent an inadvertent criticality. Based on a coincidence signal from two-out-of-three neutron detector channels, the safety rods are tripped if the neutron count rate or the reactor period approaches trip set points.

3.0 The Event Tree

The sequence of events during which errors may occur leading to an unplanned criticality during refueling has been described in the previous section. These events can be simplified and summarized.

- Initiating events leading towards a critical configuration which may originate either from loading plan errors or system-induced mispositioning errors.
- Failure to complete the count rate surveillance activity.
- Either of:
 - Failure of two-out-of-three neutron channels to respond either to high neutron count rate levels or short reactor period.
 - Failure of safety rods to remove fuel from the reactor given a trip signal from the neutron detector channels.

Figure A.4 is the event diagram for this accident sequence. It has been simplified in the following ways. No credit is taken for the period trips for two reasons. First, they derive their signals from the same neutron detectors as the level trips, so there is commonality. Second, for normal insertion or removal speeds, the reactor period is not sufficiently short to reach the period trip points. Another item that is not included in the tree is interruption of the gripper vertical motion by the neutron channels when they trip (conservative assumption). Additionally, the secondary accident of dropping a fueled subassembly into a just-subcritical core is omitted from the tree. No subassembly has been dropped into the core in 20,700 subassembly insertions and removals, so its probability of occurrence is of the order of 5×10^{-5} less than any sequence leading to a critical core.² The probability of this secondary event sequence is discussed in Section 5.5 below.

² The EBR-II fuel transfer history includes two subassemblies that were dropped in the primary tank. These were dropped by the transfer arm; it is impossible for subassemblies released from the transfer arm to interact with the reactor core. The two cases that occurred were completely benign.

Figure A.4 - Core Refueling Accident Event Tree

4.0 The Initiating Event Sequence

Prior to initiating the refueling sequence, the reactor is taken to a subcritical condition by fully withdrawing the control and safety rods. During this process, the degree of subcriticality is established; subsequently the safety rods are cocked (inserted) to their full worth of 1.0\$. The EBR-II Technical Specifications require that at least 4.0\$ of reactivity be vested in the safety and control rods during the refueling process. This means that the absolute minimum subcritical position allowed during refueling is -3.0\$. This relationship is shown in Fig. A.5.

In practice, refueling must be conducted at a level more subcritical than -3.0\$. Analysis of 304 sub-runs gave an average shut-down reactivity margin of approximately 6.1\$ at the conclusion of fuel handling. Normally, the worth of subassemblies range from -0.1\$ to +1.8\$. From the 3.0\$ position, ignoring the trippable reactivity vested in the safety rods, it would take two errors averaging 1.5\$ each to just reach critical, or three of 1.0\$ each or five of 0.6\$, etc. However, considering that reloading occurs at a subcriticality margin substantially greater than -3.0\$ and that the safety rods provide a response of -1.0\$, in practice it would take four or more reloading errors of 1.5\$ each, five to seven errors of 1.0\$ each, and a large number of positive insertion errors for subassemblies having worths less than 1.0\$ to approach critical.

Considering that, in general, the refueling process involves replacing one subassembly with another that has only marginally greater reactivity worth, the relative probability of producing a sequence of errors (within any loading plan) having sufficient additional reactivity to reach critical is very low when compared to the bounding cases described above. In the case of control rod or safety rod thimble replacement, the fueled subassemblies (up to six) temporarily removed and replaced represent a small increase in reactivity.

Figure A.5 - State of Subcriticality During Core Refueling

5.0 Assessment of Event Probabilities

5.1 The Fault Trees

The event tree shown in Fig. A.4 indicates four fault trees, one for initiating events, one to evaluate the protective surveillance activity, one for neutron detector trip signal response, and one for safety rod trip.

Figures A.6, A.7, A.8, and A.9 show the four fault trees. The human error probabilities (HEPs) were prepared by a consultant to ANL for the PRA effort [A.5]. A few remarks are appropriate. The effect of common mode, that is, planning errors on the initiating event sequence is directly accounted for in the initiating event fault tree by a human factors analysis. The process of count rate surveillance is required after each subassembly insertion in the core. However, credit is taken in the protective surveillance tree only once -- when the count rate response threshold is reached -- for this activity. Calibration errors which could impact surveillance effectiveness are specifically accounted for in the protective surveillance tree.

However, a review of the protective system and the potential effects of calibration errors make it clear that calibration errors will not invalidate the generation of trip signals. The trip level may be altered by errors in the calibration process, but as the reactor approaches or reaches critical, the large increase in neutron population ensures that trip levels will be reached. Note also that interlock checks require both low-level and high-level trip tests on the neutron channels prior to conducting core refueling.

Figure A.6 - Initiating Event Fault Tree

PROTECTIVE SURVEILLANCE FAULT TREE	
CORE REFUELING	
09-MAY-90	PAGE 1 OF 1

Figure A.7 - Protective Surveillance Fault Tree

TRIP SIGNAL FAULT TREE	
CORE REFUELING	
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Figure A.8 - Trip Signal Fault Tree

CORE REFUELING FAULT TREE	
FAILURE OF SAFETY RODS TO TRIP	
31-MAY-90	PAGE 1 OF 1

Figure A.9 - Safety Rod Trip Fault Tree

5.2 Analysis of The Fault Trees

5.2.1 The Initiating Event Tree

The details of human error and fault tree probability analyses are presented in appendices A.1 and A.2. For the fuel reloading planning process, process errors were evaluated using ASEP techniques as shown on Fig. A.6. The value estimated by this technique (4×10^{-5}) was sufficiently low that it was not refined by further analysis. No direct estimates were found for computer or data transmission errors, so these and hardware misposition errors were conservatively estimated from plant data.

Regarding the computer and hardware systems; the current software has been operating for the past ten (10) loading sequences (subruns) and the hardware system has been in place since the reactor was constructed. No failures have been observed. Assume that a loading error occurs on the next use. Then, the probability of an error is $1/11$ or 0.0909 . This number only represents the probability of a random error, without regard to reactivity magnitude. With the reactor 3.0% or more subcritical, two loading errors having a net positive reactivity change of 1.5% each must occur for criticality to be approached. This means that two 1.5% assemblies must replace, by mistake, two null-worth assemblies. The net probability for one mistake of this magnitude is $0.0909 \times \text{frequency of } 1.5\% \text{ insertions} \times \text{freq. of null-worth removals}$ or $0.0909 \times 0.161 \times 0.091 = 1.34 \times 10^{-3}$. A minimum of two errors is required to overcome the 3.0% margin. If one loading error has been made, it is assumed that a companion error is highly probable, so the net probability of a second 1.5% error is taken as $1 \times 0.161 \times 0.091$ or 1.47×10^{-2} . The maximum probability of system-induced positioning errors just leading to critical is thus $(1.34 \times 10^{-3}) \times (1.47 \times 10^{-2})$ or 2.0×10^{-5} per subrun.

These data were input to the MODULE code [A.6], which analyzes the fault tree. The mean probability (per reactor subrun) of incurring the initiating event sequence is 5.6×10^{-5} . (These values are from the modularized analytical method in MODULE).

5.2.2 The Protective Surveillance Tree

The effect of neutron detector aging is nominally indicated on the fault tree for this activity (Fig. A.7). However, the detector failure rate is not developed because of the manner in which the detectors are managed. Each detector is inserted years apart, providing a gradient in age, so that multiple simultaneous failures are extremely unlikely. Further, there are a number of self-check devices, including low count rate trips which stop the fuel handling process if any one channel trips on low count rate.

The calibration error and operator surveillance failure events were assigned probabilities of 3.0×10^{-5} and 7.33×10^{-4} per opportunity, e.g., per subrun (see HEP analysis in Appendix A.1). The MODULE code combines these effects to yield a mean probability of failure of the protective surveillance activity of 7.4×10^{-4} per subrun.

5.2.3 The Trip Signal Fault Tree

This fault tree is composed of two basic events: failure of two-out-of-three of the neutron detector channels and improper setting of the trip points. For the trip settings, the HEP analysis took credit for interlock checks prior to refueling (as recovery actions). For this event, the probability was computed as 5×10^{-6} per subrun. Published data exist for the neutron detector channels [A.7]. A conservative fuel loading time span of 3 days was assumed.

For the three detector channels, the failure probability in a three day refueling sequence is given from

$$\begin{aligned} P_f &= 3(\lambda\tau)^2 - 2(\lambda\tau)^3 \\ &= 3(5.3 \times 10^{-6}/\text{hr} \times 72 \text{ hr})^2 - 2(5.3 \times 10^{-6} \times 72)^3 \\ &= 4.4 \times 10^{-7} \text{ per loading sequence.} \end{aligned}$$

The failure rate data are given in reference [A.7].

With these inputs, the predicted rate of failure to generate a trip signal is 5.3×10^{-6} per sub run. Detector channel failure represents, again, only a small fraction of this rate.

5.2.4 Safety Rod Trip

The probability of the safety rods failing to scram is calculated as $\frac{1}{2} \times (1/475)$ or 1.1×10^{-3} per demand (zero failure in 475 demands). Noting that the demand rate is about one use per subrun, the failure rate is 1.1×10^{-3} per subrun.

5.3 Analysis of the Event Tree

Referring back to the event tree (Fig. A.4), there are two paths leading to an unprotected criticality during core refueling, I-PS-S and I-PS-T. These were computer-analyzed with Monte-Carlo techniques [A.8]. The probabilities per subrun for these two branches are:

I-PS-S: 2.6×10^{-13} with 5%/95% Confidence Limits of $2.9 \times 10^{-15}/8.1 \times 10^{-13}$

I-PS-T: 1.0×10^{-10} with 5%/95% Confidence Limits of $1.6 \times 10^{-12}/3.55 \times 10^{-10}$

Because of the high reliability of the three-channel detector system, the probability of the I-PS-S event is sufficiently low compared to the I-PS-T event that it can be dismissed. Considering that there are 7.5 subruns per year, the mean probability of incurring an unprotected criticality is $1.0 \times 10^{-10} \times 7.5 = 7.5 \times 10^{-10}$ per reactor year.

5.4 Discussion of the Event Probability

Some of the factors that could modify, or affect, the above conclusion are reviewed herein. When a trip signal is generated, it interrupts core gripper motion, thereby stopping any further reactivity insertion. Clearly, this action reduces those branches in the event tree for which trip signal success is indicated; e.g., the I-PS-T event would create lessened consequences because fuel insertion is stopped.

Another factor of conservatism is that count rate surveillance is credited only once in the structure of the fault and event trees. In practice, count rate surveillance is required after every insertion. The operators at both the fuel handling console and the reactor console are changed every two hours; since an average round trip (core to basket to core) takes 45 minutes, only two loading changes typically occur for each pair of operators. Even if surveillance were omitted by one set of operators, it is highly likely that the next set would record, and respond to, system neutron count rates.

An experience rated value of $\sim 1 \cdot 10^{-3}$ per demand for safety rod trip failure is used in the I-PS-T event branch. Another member of the EBR-II PRA team has judged that this value is less than $1 \cdot 10^{-4}$ [A.9]. Note also that the loading plan error rate input to the initiating event tree was derived by the ASEP method. It appears that a more detailed analysis of the loading plan preparation process would produce a lower failure rate estimate for loading plan errors. In the same tree, since core refueling is conducted from a subcritical position considerably greater than the required minimum of -3% , it would take more mechanical or computer loading errors than is assumed to actually challenge the protective system.

Finally, no attempt was made to take credit for manual safety rod trip by the operators. Note that if a single detector channel reaches its trip point, an alarm is triggered in the control room. This would provide the basis for a manual trip even before the automatic trip which is based on two-out-of-three coincidence.

Arrayed against the above factors of conservatism are uncertainties in some of the estimates; for example, more recent, detector channel failure rates and more definitive estimates of computer and mechanical failures. Note, however, that a search of the EBR-II incident and UOR records did not produce any evidence of a mechanical (NCPS) error in over 20,000 core fuel movements.

This discussion is concluded by saying that there is an array of conservative factors that will offset any uncertainties or significant underestimates in the numerical

probabilities. It is concluded that the event estimates are reasonable, but conservative, representations of the process and that the probability of incurring plant damage during core refueling is very low.

5.5 Secondary Events

Secondary events of two other types related to core refueling can be postulated. One is the rapid insertion of a fueled assembly into the core (failure of speed interlocks or a dropped subassembly). The second is an unplanned startup of the primary pumps with core refueling in progress.

Regarding the first of these events, the event has safety significance only if it occurs with the core near critical.

At lower subcritical levels, a dropped assembly will simply serve to replace the reactivity of the previously removed assembly without further effect (note, however, that it will set off the period trips which will initiate protective action). Since this event has significance only if the initiating event sequence and protective surveillance activities have failed, it is a very low probability scenario. The probability of dropping an assembly is less than 5×10^{-5} per demand (< 1 in 20,690 transfers). Inputting this value to the event tree analysis code while taking no credit for automatic protective response,³ the mean probability per subrun is: 2.7×10^{-11} with 5%/95% confidence limits of $3.8 \times 10^{-13}/1.0 \times 10^{-10}$. The dropped subassembly is considered to be by far the worst of the two rapid insertion scenarios, as the other requires multiple interlock failures, and the safety system can respond to it because the insertion occurs at lower speeds. The mean probability for this case over a reactor year is thus $7.5 \times 2.7 \times 10^{-11}$ or 2.0×10^{-10} .

³ For a dropped assembly, system automatic trip is not sufficiently fast to prevent core criticality, assuming that the core is just subcritical [A.10].

Startup of the primary pumps would be a significant event if there were any possibility that hydraulic forces could lift the fueled control rods into the core. An enumeration of factors related to this event serves to demonstrate that this event is not possible.

First, the pumps are interlocked out during core refueling and the pump motor power is off. But more directly, the control rods, within their thimbles, are specifically designed to achieve positive hydraulic holddown. Technical analysis demonstrates that they have positive holddown over the entire range of flow provided by the primary pumps. Therefore, control rod lift due to pump startup is precluded.

The contribution of these two secondary scenarios is thus either zero (pump induced lift of the control rods) or very low ($2 \cdot 10^{-10}$ for a subassembly dropped into a just-subcritical core). The event tree analyses are attached as Appendix A.3.

6.0 Summary

The events leading to an unprotected criticality during core refueling have been separated into four fault trees.

Evaluation of the components contributing to each tree leads to the final conclusion that the probability of incurring such an event is $8 \cdot 10^{-10}$ per reactor year or less. Although there are always uncertainties associated with the genesis of such a number, these uncertainties tend to be offset by conservative factors inherent in the analysis.

Two secondary events potentially leading to an unplanned criticality were also reviewed. One of these is considered to be impossible - that is, prevented by physical design - and the other has a predicted probability of $2 \cdot 10^{-10}$ per reactor year.

It is concluded that the risks presented by the core refueling process at EBR-II are substantially below any threshold that requires further analysis.

Appendix A References

- A.1. L. J. Koch, et al., "Hazard Summary Report, Experimental Breeder Reactor II," ANL-5719 (May 1957).
- A.2. L. J. Koch, W. B. Loewenstein, and H. O. Monson, "Addendum to Hazard Summary Report, Experimental Breeder Reactor II (EBR-II)," ANL-5719 (Addendum) (June 1962).
- A.3. W. E. Vesely, et. al., Fault Tree Handbook, NUREG-0492, USNRC, January 1981.
- A.4. R. W. Schaefer, private communication (1990).
- A.5. D. Schurman, private communication (1990).
- A.6. S. H. Han, T. W. Kim, and K. J. Yoo, "Description and Input Manual of an Integrated Fault Tree Analysis Computer Code, MODULE," Reactor Safety Department, Korea Advanced Energy Research Institute, Daeduk-Danji, Choongnam, Korea.
- A.7. J. I. Sackett and N. L. Gale, "FSAA to Hazards Summary Report, EBR-II: Upgrading of Plant Protective System," Vol. I, Section 3, ANL-76-34, August 1977.
- A.8. J. Roglans - Ribas, private communication (1990).
- A.9. S. A. Kamal, private communication (1990).
- A.10. H. A. Larson and J. I. Sackett, "Safety Considerations for Trips Related to Reactivity Change in the EBR-II PPS (In The Fuel Handling Mode), ANL-77-32, Feb. 1983.

APPENDIX A.1

HUMAN RELIABILITY TREES (Human Factors)

- Detector Reliability
- Countrate Surveillance
- Trip Setting

HRA TREE DIAGRAM

ID: *FHCR Calibrate Log Count Rate on channels A, B, & C (fuel handling)*
 Sequence: *Unknown*
 Date modified: *02-APR-90*
 Analyst: *Schuman*
 NOTES: *Tech. calibrates instruments for log count rate on Wide Range Nuclear Channels A, B, and C. Checked by the Interlock check prior to Fuel Handling.*

FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES

Calibrate Log Count Rate on channels A,B, & C (fuel handling)

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.0005 x 0.005	0.000002		
2 .aCD	0.0005 x 0.005	0.000002		
3 .acEF	0.0015 x 0.005	0.000007		
4 .aceGHI	0.001 x 0.005	0.000005		
5 .acegIJ	0.001 x 0.005	0.000005		
6 .acegiKL	0.0015 x 0.005	0.000007		
A.1-2			0.00003	0.99997
			EF = 5	

HRA TREE DIAGRAM

ID: *FHCU Operator fails to follow fuel handling Count rate Procedure.*
 Sequence: *Unknown*
 Date modified: *04-MAY-90*
 Analyst: *Schurman*
 NOTES: *When notified by Fuel-Handling Console Operator - Control Room Operator fails to use and follow Count rate Surveillance Procedure.*

FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES

Operator fails to follow fuel handling Countrate Procedure.

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.000001 x 0.001	*		
2 .aCDEF	0.0005 x 0.143285 x 0.143285 x 0.185714	0.000001		
3 .acGHIJ	0.003 x 0.5015 x 0.5015 x 0.525	0.000396		
4 .acgKLM	0.000333 x 0.500166 x 0.500166	0.000083		
5 .acgkNOPQ	0.05 x 0.185714 x 0.185714 x 0.0557	0.000096		
6 .acgknRS	0.003 x 0.0519	0.000155		
A.1-4			0.000733	0.999266
			EF = 5	

FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES

Operator fails to follow fuel handling Countrate Procedure.

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.000001 x 0.001	*		
2 .aCDEF	0.0005 x 0.143285 x 0.143285 x 0.185714	0.000001		
3 .acGHIJ	0.003 x 0.5015 x 0.5015 x 0.525	0.000396		
4 .acgKLM	0.000333 x 0.500166 x 0.500166	0.000083		
5 .acgkNOPQ	0.05 x 0.185714 x 0.185714 x 0.0557	0.000096		
6 .acgknRS	0.003 x 0.0519	0.000155		
			0.000733	0.999266

HEPs for Operator fails to follow fuel handling Countrate Procedure.

Human Action/Error	Basic IEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal IEP (Error Factor)
A Operator fails to initiate countrate surveillance when requested.	0.000001 (3)	Table 20-5(2)	SBS	1		ZD	0.000001 (3)
B FHCO fails to notify CRO to initiate surveillance.	0.001 (5)	Table 20-6(2)	SBS	1		ZD	0.001 (5)
C CRO misreads Channel A countrate LOW.	0.0005 (5)	.5 x Table 20-10(2)	SBS	1		ZD	0.0005 (5)
D CRO misreads Channel B countrate LOW.	0.0005 (5)	.5 x Table 20-10(2)	SBS	1		MD	0.143285 (5)
E CRO misreads Channel C countrate LOW.	0.0005 (5)	.5 x Table 20-10(2)	SBS	1		MD	0.143285 (5)
F CRO does not catch reading errors when recording data from each channel.	0.05 (10)	Page 11-13/An. Jdgm	SBS	1		MD	0.185714 (10)
G CRO fails to calculate CO/CN ratio for Channel A.	0.003 (3)	Table 20-5(1)	SBS	1		ZD	0.003 (3)
H CRO fails to calculate CO/CN ratio for Channel B.	0.003 (3)	Table 20-5(1)	SBS	1		HD	0.5015 (3)
I CRO fails to calculate CO/CN ratio for Channel C.	0.003 (3)	Table 20-5(1)	SBS	1		HD	0.5015 (3)
J CRO fails to write down CO/CN ratios.	0.05 (3)	Table 20-10(10)	SBS	1		HD	0.525 (3)
K CRO fails chart CO/CN ratio for Channel A. (LCB because of written prior step)	0.000333 (10)	Table 20-7(LCB)	SBS	1		ZD	0.000333 (10)
L CRO fails chart CO/CN ratio for Channel B. (LCB because of written prior step)	0.000333 (10)	Table 20-7(LCB)	SBS	1		HD	0.500166 (10)

M	CRO fails chart CO/CN ratio for Channel C. (LCB because of written prior step)	0.000333 (10)	Table 20-7(LCB)	SBS	1	HD	0.500166 (10)
N	CRO fails to note CO/CN ratio for Channel A is trending out of range.	0.05 (5)	Table 20-10(11)	SBS	1	ZD	0.05 (5)
O	CRO fails to note CO/CN ratio for Channel B is trending out of range.	0.05 (5)	Table 20-10(11)	SBS	1	MD	0.185714 (5)
P	CRO fails to note CO/CN ratio for Channel C is trending out of range.	0.05 (5)	Table 20-10(11)	SBS	1	MD	0.185714 (5)
Q	FHCO fails to warn CRO that counterates are trending out of range.	0.006 (5)	Table 20-10(3)	SBS	1	LD	0.0557 (5)
R	CRO fails to terminate fuel handling prior to Safety Rod SCRAM.	0.003 (3)	Table 20-7(2)	SBS	1	ZD	0.003 (3)
S	CRO fails to attend to high count rate alarm.	0.001 (3)	Table 20-23(1)	SBS	2	LD	0.0519 (3)

HRA TREE DIAGRAM

ID: FHSP Tech. sets bistable trip points for LCR A,B, & C. (fuel hand)
Sequence: Unknown
Date modified: 03-APR-90
Analyst: Schurman
NOTES: Technician sets LCR bistable trip points for log count rate of channels A, B, and C. These settings are checked by the Interlock checks before start up of the reactor. (Proc. STM-MC-77 Rev. 8: 2/24/87)

HIEPs for Tech. sets bistable trip points for LCR A,B, & C. (fuel handling)

Human Action/Error	Basic HEP (Error Factor)	THERP Source Table (Item)	Step-by- Step (SBS) Dynamic (DYN)	Modifier for Stress, Experience, SBS or DYN	Modifier Source THERP Table (Item)	Dependency	Nominal HEP (Error Factor)
A Tech. fails to set adjust power supply properly.(Step 9.1)	0.0005 (3)	.5 x Table 20-7 (1)	SBS	1		ZD	0.0005 (3)
B Interlock check fails to find misadjustment.(STM-MC-14, Log I-E (11.))	0.005 (5)	Table 20-22 (4, 10)	SBS	1		ZD	0.005 (5)
C Tech fails to adjust discriminator properly. (Step 9.2)	0.0005 (3)	.5 x Table 20-7 (1)	SBS	1		ZD	0.0005 (3)
D Interlock check fails to discover misadjustment.	0.005 (5)	Table 20-22 (4, 10)	SBS	1		ZD	0.005 (5)

FAILURE PATHS and TOTAL FAILURE/SUCCESS PROBABILITIES

Tech. set bistable trip points for LCR A,B, & C. (fuel handling)

Failure Path	Calculations	Resulting Failure Probability	Total Failure Probability	Total Probability of Success
1 .AB	0.0005 x 0.005	0.000002		
2 .aCD	0.0005 x 0.005	0.000002		
			0.000005	0.999995
			EF = 5	

APPENDIX A.2

FAULT TREE ANALYSES (Module)

- Initiating Events
- Protective (Counter) Surveillance
- Trip Setting
- Safety Rod Trip
- Dropped Subassembly

5/9/90 2:56 PM

HD-40:PRA Codes:INI EV

Page 1

\$TREE

TOP +
"

CFA005RL CCG005RG

CORE RELOADING ERRORS

CFA005RL C

ERRORS IN PLANNING

CCG005RG C

COMPUTER OR MECHANICAL FAILURE

\$END

\$DATA

4.000E-05 16.0

CFA005RL

2.000E-05 10.0

CCG005RG

\$END

5/9/90 3:06 PM

HD-40:PRA Codes:IE.trd

Page 1

TREDRA FILE CREATED BY MODULE

* OPTION

PRINTPLO 6 7 F F

END

* TREE

TOP OR 0 2 CFA005RL CCG005RG

END

* TITLE

INITIATING EVENT FAULT TREE

CORE REFUELING

09-MAY-90

END

* INFORM

8.50 11.00NORMAL F F 11.00 1.00

END

* EVENTS

TOP	CORE	RELOADING	ERRORS
CFA005RL 2	ERRORS IN	PLANNING	
CCG005RG 2	COMPUTER OR	MECHANICAL	FAILURE

END

FINISH

STOP

Output from module

Wed, May 9, 1990

PROGRAM MODULE ENTERED: 09-MAY-90

ENTER THE INPUT FILE NAME :

INI EV

ENTER THE OUTPUT FILE NAME :

IE OUT

READ DATA : 43.0 SEC REQUIRED

ENTER TOP EVENT NAME :

TOP

WRITE TREDRA INPUT : 8.0 SEC REQUIRED

MODULARIZATION : 0.0 SEC REQUIRED

ENTER CUT-OFF PROBA. :

1.0E-07

ENTER MAX. TERM NUM. :

4

MCS : 1 TERMS GENERATED

DO YOU WANT TO ANALYZE IMPORTANCE ?

Y

IMPORTANCE : 3.0 SEC REQUIRED

DO YOU WANT TO ANALYZE UNCERTAINTY ?

Y

SAMPLING : 8.0 SEC REQUIRED

SORTING : 1.0 SEC REQUIRED

UNCERTAINTY : 1.0 SEC REQUIRED

BOTTOM-UP : 33.0 SEC REQUIRED

DO YOU WANT TO EXPAND MODULARIZED MCS ?:

Y

MCS : 2 TERMS GENERATED

CUTSET : 4.0 SEC REQUIRED

DO YOU WANT TO ANALYZE IMPORTANCE ?:

Y

IMPORTANCE : 3.0 SEC REQUIRED

DO YOU WANT TO ANALYZE UNCERTAINTY ?:

Y

SAMPLING : 13.0 SEC REQUIRED

SORTING : 0.0 SEC REQUIRED

UNCERTAINTY : 1.0 SEC REQUIRED

EXPAND : 21.0 SEC REQUIRED

MODULE : 105.0 SEC REQUIRED

END OF MODULE RUN

STOP

5/9/90 3:09 PM

HD-40:PRA Codes:IE.mcs

Page 1

MODULE VERSION 2.0 RUN AT ANL ON: 09-MAY-90

CUT SETS FOR THE TOP EVENT: TOP

PROBABILITY = 6.000E-05

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
CFA005RL	6.000E-05	1.0000	16666.6680	INF

NO. OF MINIMAL CUT SETS = 1

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	6.000E-05	1.0000	CFA005RL

PROBABILITY = 6.000E-05

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
CFA005RL	4.000E-05	0.6667	16667.0000	3.0000
CCG005RG	2.000E-05	0.3333	16667.3340	1.5000

NO. OF MINIMAL CUT SETS = 2

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	4.000E-05	0.6667	CFA005RL
2	2.000E-05	0.3333	CCG005RG

MODULE VERSION 2.0 RUN AT ANL ON: 09-MAY-90

FAULT TREE INPUT

TOP + CFA005RL CCG005RG
" CORE RELOADING ERRORS
CFA005RL C ERRORS IN PLANNING
CCG005RG C COMPUTER OR MECHANICAL FAILURE
1

THE FOLLOWING ARE TOP EVENTS : 1

TOP

THE FOLLOWING ARE GATES : 1

TOP

THE FOLLOWING ARE EVENTS : 2

CFA005RL CCG005RG

RELIABILITY INPUT DATA

4.000E-05 16.0 CFA005RL
2.000E-05 10.0 CCG005RG
1

MODULARIZED FAULT TREE

TOP + CFA005RL

MODULES INFORMATION

CFA005RL + CFA005RL CCG005RG

CUT SETS FOR THE TOP EVENT: TOP

NO. OF EVENTS IN FAULT TREE = 1

CFA005RL 6.000E-05 11.345

CUTSETS

:

20.0 SEC REQUIRED

A.2-5

PROBABILITY = 6.000E-05

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
CFA005RL	6.000E-05	1.0000	16666.6680	INF

1

NO. OF MINIMAL CUT SETS = 1

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	6.000E-05	1.0000	CFA005RL

1

R E S U L T S S U M M A R I Z E D

DISTRIBUTION VALUES

NUMBER OF SAMPLE	=	1200
ACCURACY ON 95% CONFIDENCE INTERVAL	=	1.03496E+00
POINT MEAN VALUE	=	6.00000E-05
MEDIAN VALUE	=	1.97560E-05
ERROR FACTOR	=	1.21363E+01
MEAN VALUE	=	5.60293E-05
STANDARD DEVIATION	=	1.41182E-04
5% BOUND	=	1.62785E-06
95% BOUND	=	2.15119E-04

0

DISTRIBUTION CONFIDENCE LIMITS

CONFIDENCE (PERCENT)	FUNCTION VALUE
0.5	3.89317E-07
1.0	5.50815E-07
2.5	1.01425E-06
5.0	1.62785E-06
10.0	2.96571E-06
20.0	5.86514E-06
25.0	7.59584E-06
30.0	9.41912E-06
40.0	1.40493E-05
50.0	1.97560E-05
60.0	2.86400E-05
70.0	4.41414E-05
75.0	5.39243E-05
80.0	6.52844E-05
90.0	1.24124E-04
95.0	2.15119E-04
97.5	3.28589E-04
99.0	5.68533E-04
99.5	7.87373E-04

1

PROBABILITY = 6.000E-05

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
CFA005RL	4.000E-05	0.6667	16667.0000	3.0000
CCG005RG	2.000E-05	0.3333	16667.3340	1.5000

1

NO. OF MINIMAL CUT SETS = 2

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	4.000E-05	0.6667	CFA005RL
2	2.000E-05	0.3333	CCG005RG

1

R E S U L T S S U M M A R I Z E D

DISTRIBUTION VALUES

NUMBER OF SAMPLE	=	1200
ACCURACY ON 95% CONFIDENCE INTERVAL	=	1.03496E+00
POINT MEAN VALUE	=	6.00000E-05
MEDIAN VALUE	=	2.42780E-05
ERROR FACTOR	=	9.37351E+00
MEAN VALUE	=	6.38393E-05
STANDARD DEVIATION	=	1.75463E-04
5% BOUND	=	3.62710E-06
95% BOUND	=	2.27570E-04

0

DISTRIBUTION CONFIDENCE LIMITS

CONFIDENCE (PERRCENT)	FUNCTION VALUE
0.5	1.24301E-06
1.0	1.73254E-06
2.5	2.63171E-06
5.0	3.62710E-06
10.0	5.44619E-06
20.0	9.16734E-06
25.0	1.10777E-05
30.0	1.32602E-05
40.0	1.82556E-05
50.0	2.42780E-05
60.0	3.31054E-05
70.0	4.89109E-05
75.0	5.78617E-05
80.0	7.35172E-05
90.0	1.34821E-04
95.0	2.27570E-04
97.5	3.17953E-04
99.0	5.16274E-04

5/9/90 3:07 PM

HD-40:PRA Codes:IE.mod

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99.5 1.10218E-03

MODULE : 105.0 SEC EXECUTION TIME

END OF MODULE RUN

INITIATING EVENT FAULT TREE	
CORE REFUELING	
09-MAY-90	PAGE 1 OF 1

5/9/90 3:19 PM

HD-40:PRA Codes:PR SURV

Page 1

```
$TREE
TOP      +      G0112      FHU001RZ
          "      PROTECTIVE  SURVEILLANCE  FAILURE
G0112    +      CND001RM  CND002RW
          "      LOW        NEUTRON      COUNTS
CND001RM C      ERROR IN   CALIBRATION
CND002RW C      DETECTOR   AGING      (UNDEVELOPED)
FHU001RZ C      INADEQUATE  COUNTRATE  SURVEILLANCE
$END
$DATA
  3.000E-05  5.0  CND001RM
  1.000E-10  3.0  CND002RW
  7.330E-04  5.0  FHU001RZ
$END
```

5/9/90 3:15 PM

HD-40:PRA Codes:PS.trd

Page 1

TREDRA FILE CREATED BY MODULE

* OPTION

PRINTPLO 6 7 F F

END

* TREE

TOP OR 1 1 G0112 FHU001RZ

G0112 OR 0 2 CND001RM CND002RW

END

* TITLE

PROTECTIVE SURVEILLANCE FAULT TREE
CORE REFUELING

09-MAY-90

END

* INFORM

8.50 11.00NORMAL F F 11.00 1.00

END

* EVENTS

TOP	PROTECTIVE	SURVEILLANCE	FAILURE
G0112	LOW	NEUTRON	COUNTS
FHU001RZ 2	INADEQUATE	COUNTRATE	SURVEILLANCE
CND001RM 2	ERROR IN	CALIBRATION	
CND002RW 2	DETECTOR	AGING	(UNDEVELOPED)

END

FINISH

STOP

Output from module

Wed, May 9,

```

                                PROGRAM MODULE ENTERED: 09-MAY-90
ENTER THE INPUT FILE NAME :
PR SURV
ENTER THE OUTPUT FILE NAME :
PS.OUT
                                READ DATA          :      34.0 SEC REQUIRED
ENTER TOP EVENT NAME :
TOP
                                WRITE TREDRA INPUT :      4.0 SEC REQUIRED
                                MODULARIZATION    :      0.0 SEC REQUIRED
ENTER CUT-OFF PROBA. :
1.0E-08
ENTER MAX. TERM NUM. :
5
                                MCS :      1 TERMS GENERATED
DO YOU WANT TO ANALYZE IMPORTANCE ?
Y
                                IMPORTANCE :      2.0 SEC REQUIRED
DO YOU WANT TO ANALYZE UNCERTAINTY ?
Y
                                SAMPLING :      7.0 SEC REQUIRED
                                SORTING :      1.0 SEC REQUIRED
                                UNCERTAINTY :      0.0 SEC REQUIRED
                                BOTTOM-UP :      53.0 SEC REQUIRED
DO YOU WANT TO EXPAND MODULARIZED MCS ?
Y
                                MCS :      2 TERMS GENERATED
                                CUTSET :      4.0 SEC REQUIRED
DO YOU WANT TO ANALYZE IMPORTANCE ?
Y
                                IMPORTANCE :      4.0 SEC REQUIRED
DO YOU WANT TO ANALYZE UNCERTAINTY ?
Y
                                SAMPLING :      13.0 SEC REQUIRED
                                SORTING :      1.0 SEC REQUIRED
                                UNCERTAINTY :      0.0 SEC REQUIRED
                                EXPAND :      22.0 SEC REQUIRED
                                MODULE :      113.0 SEC REQUIRED

END OF MODULE RUN

STOP
```

MODULE VERSION 2.0 RUN AT ANL ON: 09-MAY-90

CUT SETS FOR THE TOP EVENT: TOP

PROBABILITY = 7.630E-04

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
FHU001RZ	7.630E-04	1.0000	1310.6158	INF

1

NO. OF MINIMAL CUT SETS = 1

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	7.630E-04	1.0000	FHU001RZ

PROBABILITY = 7.630E-04

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
FHU001RZ	7.330E-04	0.9607	1310.6553	25.4333
CND001RM	3.000E-05	0.0393	1311.5765	1.0409

1

NO. OF MINIMAL CUT SETS = 2

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	7.330E-04	0.9607	FHU001RZ
2	3.000E-05	0.0393	CND001RM

MODULE VERSION 2.0 RUN AT ANL ON: 09-MAY-90

FAULT TREE INPUT

TOP	+	G0112	FHU001RZ		
	"	PROTECTIVE	SURVEILLANCE	FAILURE	
G0112	+	CND001RM	CND002RW		
	"	LOW	NEUTRON	COUNTS	
CND001RM	C	ERROR IN	CALIBRATION		
CND002RW	C	DETECTOR	AGING	(UNDEVELOPED)	
FHU001RZ	C	INADEQUATE	COUNTRATE	SURVEILLANCE	

1

THE FOLLOWING ARE TOP EVENTS : 1

TOP

THE FOLLOWING ARE GATES : 2

TOP G0112

THE FOLLOWING ARE EVENTS : 3

FHU001RZ CND001RM CND002RW

RELIABILITY INPUT DATA

3.000E-05	5.0	CND001RM
1.000E-10	3.0	CND002RW
7.330E-04	5.0	FHU001RZ

1

MODULARIZED FAULT TREE

TOP + FHU001RZ

MODULES INFORMATION

G0112	+	CND001RM	CND002RW
FHU001RZ	+	G0112	FHU001RZ

CUT SETS FOR THE TOP EVENT: TOP

NO. OF EVENTS IN FAULT TREE = 1

FHU001RZ 7.630E-04 4.801

CUTSETS : 43.0 SEC REQUIRED

1

PROBABILITY = 7.630E-04

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
FHU001RZ	7.630E-04	1.0000	1310.6158	INF

1

NO. OF MINIMAL CUT SETS = 1

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	7.630E-04	1.0000	FHU001RZ

1

R E S U L T S S U M M A R I Z E D

DISTRIBUTION VALUES

NUMBER OF SAMPLE	=	1200
ACCURACY ON 95% CONFIDENCE INTERVAL	=	1.03496E+00
POINT MEAN VALUE	=	7.63000E-04
MEDIAN VALUE	=	4.77689E-04
ERROR FACTOR	=	5.01491E+00
MEAN VALUE	=	7.40154E-04
STANDARD DEVIATION	=	8.61753E-04
5% BOUND	=	9.52537E-05
95% BOUND	=	2.23346E-03

0

DISTRIBUTION CONFIDENCE LIMITS

CONFIDENCE (PERCENT)	FUNCTION VALUE
0.5	3.78049E-05
1.0	4.73035E-05
2.5	7.01712E-05
5.0	9.52537E-05
10.0	1.40333E-04
20.0	2.18001E-04
25.0	2.57630E-04
30.0	2.96040E-04
40.0	3.83280E-04
50.0	4.77689E-04
60.0	6.07181E-04
70.0	8.02925E-04
75.0	9.13761E-04
80.0	1.03386E-03
90.0	1.56571E-03

95.0 2.23346E-03
 97.5 2.93641E-03
 99.0 4.18426E-03
 99.5 5.16382E-03

1

PROBABILITY = 7.630E-04

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
FHU001RZ	7.330E-04	0.9607	1310.6553	25.4333
CND001RM	3.000E-05	0.0393	1311.5765	1.0409

1

NO. OF MINIMAL CUT SETS = 2

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	7.330E-04	0.9607	FHU001RZ
2	3.000E-05	0.0393	CND001RM

1

R E S U L T S S U M M A R I Z E D

DISTRIBUTION VALUES

NUMBER OF SAMPLE = 1200
 ACCURACY ON 95% CONFIDENCE INTERVAL = 1.03496E+00
 POINT MEAN VALUE = 7.63000E-04
 MEDIAN VALUE = 4.79150E-04
 ERROR FACTOR = 5.32981E+00
 MEAN VALUE = 7.77385E-04
 STANDARD DEVIATION = 1.02385E-03
 5% BOUND = 1.11516E-04
 95% BOUND = 2.55378E-03

0

DISTRIBUTION CONFIDENCE LIMITS

CONFIDENCE (PERCENT)	FUNCTION VALUE
0.5	4.89684E-05
1.0	6.13423E-05
2.5	7.99520E-05
5.0	1.11516E-04
10.0	1.52970E-04
20.0	2.18788E-04
25.0	2.52932E-04
30.0	2.83635E-04
40.0	3.65183E-04
50.0	4.79150E-04
60.0	6.10710E-04
70.0	7.74163E-04
75.0	8.79451E-04

5/9/90 3:16 PM

HD-40:PRA Codes:PS.mod

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80.0	1.06862E-03
90.0	1.67896E-03
95.0	2.55378E-03
97.5	3.18339E-03
99.0	4.53526E-03
99.5	7.09643E-03

MODULE : 113.0 SEC EXECUTION TIME

END OF MODULE RUN

PROTECTIVE SURVEILLANCE FAULT TREE	
CORE REFUELING	
09-MAY-90	PAGE 1 OF 1

5/9/90 3:27 PM

HD-40:PRA Codes:TR SIG

Page 1

\$TREE

TOP +

"

FHU002RZ CND003RH

NO TRIP

SIGNAL

FHU002RZ C

IMPROPER TRIP

SETTINGS

CND003RH C

TWO-OF-THREE DETECTOR

CHANNEL

FAILURES

\$END

\$DATA

5.000E-06 5.0

FHU002RZ

4.400E-07 3.0

CND003RH

\$END

5/9/90 3:25 PM

HD-40:PRA Codes:TS.trd

Page 1

TREDRA FILE CREATED BY MODULE

* OPTION

PRINTPLO 6 7 F F

END

* TREE

TOP OR 0 2 FHU002RZ CND003RH

END

* TITLE

TRIP SIGNAL FAULT TREE

CORE REFUELING

09-MAY-90

END

* INFORM

8.50 11.00NORMAL F F 11.00 1.00

END

* EVENTS

TOP

FHU002RZ 2

NO

IMPROPER

TRIP

TRIP

SIGNAL

SETTINGS

CND003RH 2

TWO-OF-THREE

DETECTOR

CHANNEL

FAILURES

END

FINISH

STOP

Output from module

Wed, May 9, 1990

```

                                PROGRAM MODULE ENTERED: 09-MAY-90
ENTER THE INPUT FILE NAME :
TR SIG
ENTER THE OUTPUT FILE NAME :
TS.OUT
                                READ DATA      :      19.0 SEC REQUIRED
ENTER TOP EVENT NAME :
TOP
                                WRITE TREDRA INPUT :      4.0 SEC REQUIRED
                                MODULARIZATION    :      0.0 SEC REQUIRED
ENTER CUT-OFF PROBA. :
1.0E-08
ENTER MAX. TERM NUM. :
4
                                MCS :      1 TERMS GENERATED
DO YOU WANT TO ANALYZE IMPORTANCE ?
Y
                                IMPORTANCE :      3.0 SEC REQUIRED
DO YOU WANT TO ANALYZE UNCERTAINTY ?
Y
                                SAMPLING :      7.0 SEC REQUIRED
                                SORTING :      1.0 SEC REQUIRED
                                UNCERTAINTY :      0.0 SEC REQUIRED
                                BOTTOM-UP :      29.0 SEC REQUIRED
DO YOU WANT TO EXPAND MODULARIZED MCS ?
Y
                                MCS :      2 TERMS GENERATED
                                CUTSET :      6.0 SEC REQUIRED
DO YOU WANT TO ANALYZE IMPORTANCE ?
Y
                                IMPORTANCE :      4.0 SEC REQUIRED
DO YOU WANT TO ANALYZE UNCERTAINTY ?
Y
                                SAMPLING :      13.0 SEC REQUIRED
                                SORTING :      0.0 SEC REQUIRED
                                UNCERTAINTY :      1.0 SEC REQUIRED
                                EXPAND :      24.0 SEC REQUIRED
                                MODULE :      76.0 SEC REQUIRED

END OF MODULE RUN

STOP
```

MODULE VERSION 2.0 RUN AT ANL ON: 09-MAY-90

CUT SETS FOR THE TOP EVENT: TOP

PROBABILITY = 5.440E-06

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
FHU002RZ	5.440E-06	1.0000	*****	INF

NO. OF MINIMAL CUT SETS = 1

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	5.440E-06	1.0000	FHU002RZ

PROBABILITY = 5.440E-06

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
FHU002RZ	5.000E-06	0.9191	*****	12.3636
CND003RH	4.400E-07	0.0809	*****	1.0880

NO. OF MINIMAL CUT SETS = 2

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	5.000E-06	0.9191	FHU002RZ
2	4.400E-07	0.0809	CND003RH

MODULE VERSION 2.0 RUN AT ANL ON: 09-MAY-90

FAULT TREE INPUT

1

TOP	+	FHU002RZ	CND003RH		
	"	NO	TRIP	SIGNAL	
FHU002RZ	C	IMPROPER	TRIP	SETTINGS	
CND003RH	C	TWO-OF-THREE	DETECTOR	CHANNEL	FAILURES

THE FOLLOWING ARE TOP EVENTS : 1

TOP

THE FOLLOWING ARE GATES : 1

TOP

THE FOLLOWING ARE EVENTS : 2

FHU002RZ CND003RH

RELIABILITY INPUT DATA

1

5.000E-06	5.0	FHU002RZ
4.400E-07	3.0	CND003RH

MODULARIZED FAULT TREE

TOP + FHU002RZ

MODULES INFORMATION

FHU002RZ + FHU002RZ CND003RH

CUT SETS FOR THE TOP EVENT: TOP

NO. OF EVENTS IN FAULT TREE = 1

FHU002RZ 5.440E-06 4.590

CUTSETS : 18.0 SEC REQUIRED

PROBABILITY = 5.440E-06

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
FHU002RZ	5.440E-06	1.0000	*****	INF

NO. OF MINIMAL CUT SETS = 1

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	5.440E-06	1.0000	FHU002RZ

R E S U L T S S U M M A R I Z E D

DISTRIBUTION VALUES

NUMBER OF SAMPLE	=	1200
ACCURACY ON 95% CONFIDENCE INTERVAL	=	1.03496E+00
POINT MEAN VALUE	=	5.44000E-06
MEDIAN VALUE	=	3.49563E-06
ERROR FACTOR	=	4.78886E+00
MEAN VALUE	=	5.28507E-06
STANDARD DEVIATION	=	5.89882E-06
5% BOUND	=	7.29951E-07
95% BOUND	=	1.56386E-05

DISTRIBUTION CONFIDENCE LIMITS

CONFIDENCE (PERCENT)	FUNCTION VALUE
0.5	2.97468E-07
1.0	3.69829E-07
2.5	5.42459E-07
5.0	7.29951E-07
10.0	1.06355E-06
20.0	1.63149E-06
25.0	1.91888E-06
30.0	2.19621E-06
40.0	2.82249E-06
50.0	3.49563E-06
60.0	4.41285E-06
70.0	5.78900E-06
75.0	6.56379E-06
80.0	7.40032E-06
90.0	1.10750E-05
95.0	1.56386E-05
97.5	2.04003E-05
99.0	2.87765E-05
99.5	3.53003E-05

PROBABILITY = 5.440E-06

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
FHU002RZ	5.000E-06	0.9191	*****	12.3636
CND003RH	4.400E-07	0.0809	*****	1.0880

NO. OF MINIMAL CUT SETS = 2

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	5.000E-06	0.9191	FHU002RZ
2	4.400E-07	0.0809	CND003RH

R E S U L T S S U M M A R I Z E D

DISTRIBUTION VALUES

NUMBER OF SAMPLE	=	1200
ACCURACY ON 95% CONFIDENCE INTERVAL	=	1.03496E+00
POINT MEAN VALUE	=	5.44000E-06
MEDIAN VALUE	=	3.51559E-06
ERROR FACTOR	=	5.04508E+00
MEAN VALUE	=	5.53629E-06
STANDARD DEVIATION	=	6.98674E-06
5% BOUND	=	9.50156E-07
95% BOUND	=	1.77364E-05

DISTRIBUTION CONFIDENCE LIMITS

CONFIDENCE (PERCENT)	FUNCTION VALUE
0.5	4.83732E-07
1.0	5.94763E-07
2.5	7.57991E-07
5.0	9.50156E-07
10.0	1.26117E-06
20.0	1.70644E-06
25.0	1.97069E-06
30.0	2.19928E-06
40.0	2.75084E-06
50.0	3.51559E-06
60.0	4.39291E-06
70.0	5.49891E-06
75.0	6.24136E-06
80.0	7.57784E-06
90.0	1.16374E-05
95.0	1.77364E-05
97.5	2.20581E-05
99.0	3.12171E-05

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HD-40:PRA Codes:TS.mod

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99.5 4.86015E-05

MODULE : 76.0 SEC EXECUTION TIME

END OF MODULE RUN

TRIP SIGNAL FAULT TREE	
CORE REFUELING	
09-MAY-90	PAGE 1 OF 1

9/19/90 13:39

HD-40:PRA Codes:ROD TRIP

Page 1

\$TREE

TOP +
"

CSF001RW

SAFETY RODS

FAIL TO

TRIP

CSF001RW C

SAFETY RODS

DO NOT TRIP

ON DEMAND

\$END

\$DATA

2.100E-03 3.0

CSF001RW

\$END

5/31/90 9:09 AM

HD-40:PRA Codes:RT.trd

Page 1

TREDRA FILE CREATED BY MODULE

* OPTION

PRINTPLO 6 7 F F

END

* TREE

TOP OR 0 1 CSF001RW

END

* TITLE

CORE REFUELING FAULT TREE

FAILURE OF SAFETY RODS TO TRIP

31-MAY-90

END

* INFORM

8.50 11.00NORMAL F F 11.00 1.00

END

* EVENTS

TOP		SAFETY RODS	FAIL TO	TRIP
CSF001RW 2		SAFETY RODS	DO NOT TRIP	ON DEMAND

END

FINISH

TOP

Output from module

Thu, May 31, 196

```

      PROGRAM MODULE ENTERED:  31-MAY-90
ENTER THE INPUT FILE NAME  :
ROD TRIP
ENTER THE OUTPUT FILE NAME :
RT.OUT
      READ DATA           :      45.0 SEC REQUIRED
ENTER TOP EVENT NAME  :
TOP
      WRITE TREDRA INPUT :      4.0 SEC REQUIRED
      MODULARIZATION    :      0.0 SEC REQUIRED
ENTER CUT-OFF PROBA. :
1.0E-06
ENTER MAX. TERM NUM. :
5
      MCS :      1 TERMS GENERATED
DO YOU WANT TO ANALYZE IMPORTANCE ?
Y
      IMPORTANCE :      2.0 SEC REQUIRED
DO YOU WANT TO ANALYZE UNCERTAINTY ?
Y
      SAMPLING :      7.0 SEC REQUIRED
      SORTING :      1.0 SEC REQUIRED
      UNCERTAINTY :      0.0 SEC REQUIRED
      BOTTOM-UP :      29.0 SEC REQUIRED
DO YOU WANT TO EXPAND MODULARIZED MCS ?
Y
      MCS :      1 TERMS GENERATED
      CUTSET :      7.0 SEC REQUIRED
DO YOU WANT TO ANALYZE IMPORTANCE ?
Y
      IMPORTANCE :      3.0 SEC REQUIRED
DO YOU WANT TO ANALYZE UNCERTAINTY ?
Y
      SAMPLING :      8.0 SEC REQUIRED
      SORTING :      1.0 SEC REQUIRED
      UNCERTAINTY :      1.0 SEC REQUIRED
      EXPAND :      20.0 SEC REQUIRED
      MODULE :      98.0 SEC REQUIRED

END OF MODULE RUN

STOP
```

MODULE VERSION 2.0 RUN AT ANL ON: 31-MAY-90

CUT SETS FOR THE TOP EVENT: TOP

PROBABILITY = 2.100E-03

EVENT DATA INFORMATION

	NAME	MEAN	IMPORTANCE	RAW	RRW
1	CSF001RW	2.100E-03	1.0000	476.1905	INF

NO. OF MINIMAL CUT SETS = 1

MINIMAL CUT SETS INFORMATION

	NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
	1	2.100E-03	1.0000	CSF001RW

PROBABILITY = 2.100E-03

EVENT DATA INFORMATION

	NAME	MEAN	IMPORTANCE	RAW	RRW
1	CSF001RW	2.100E-03	1.0000	476.1905	INF

NO. OF MINIMAL CUT SETS = 1

MINIMAL CUT SETS INFORMATION

	NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
	1	2.100E-03	1.0000	CSF001RW

MODULE VERSION 2.0 RUN AT ANL ON: 31-MAY-90

FAULT TREE INPUT

TOP + CSF001RW
" SAFETY RODS FAIL TO TRIP
CSF001RW C SAFETY RODS DO NOT TRIP ON DEMAND
1

THE FOLLOWING ARE TOP EVENTS : 1

TOP

THE FOLLOWING ARE GATES : 1

TOP

THE FOLLOWING ARE EVENTS : 1

CSF001RW

RELIABILITY INPUT DATA

2.100E-03 3.0 CSF001RW
1

MODULARIZED FAULT TREE

TOP + CSF001RW

CUT SETS FOR THE TOP EVENT: TOP

NO. OF EVENTS IN FAULT TREE = 1

CSF001RW 2.100E-03 3.000
1 CUTSETS : 19.0 SEC REQUIRED

PROBABILITY = 2.100E-03

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
CSF001RW	2.100E-03	1.0000	476.1905	INF

1

NO. OF MINIMAL CUT SETS = 1

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	2.100E-03	1.0000	CSF001RW

1

R E S U L T S S U M M A R I Z E D

DISTRIBUTION VALUES

NUMBER OF SAMPLE	=	1200
ACCURACY ON 95% CONFIDENCE INTERVAL	=	1.03496E+00
POINT MEAN VALUE	=	2.10000E-03
MEDIAN VALUE	=	1.66440E-03
ERROR FACTOR	=	3.09291E+00
MEAN VALUE	=	2.06506E-03
STANDARD DEVIATION	=	1.49934E-03
5% BOUND	=	5.38134E-04
95% BOUND	=	4.90135E-03

0

DISTRIBUTION CONFIDENCE LIMITS

CONFIDENCE (PERCENT)	FUNCTION VALUE
0.5	2.81743E-04
1.0	3.29625E-04
2.5	4.34460E-04
5.0	5.38134E-04
10.0	7.05878E-04
20.0	9.60922E-04
25.0	1.08015E-03
30.0	1.19055E-03
40.0	1.42657E-03
50.0	1.66440E-03
60.0	1.96882E-03
70.0	2.39435E-03
75.0	2.62127E-03
80.0	2.85803E-03
90.0	3.82198E-03
95.0	4.90135E-03
97.5	5.93653E-03
99.0	7.60737E-03
99.5	8.81465E-03

1

PROBABILITY = 2.100E-03

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
CSF001RW	2.100E-03	1.0000	476.1905	INF

A.2-33

1

NO. OF MINIMAL CUT SETS = 1

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	2.100E-03	1.0000	CSF001RW

1

R E S U L T S S U M M A R I Z E D

DISTRIBUTION VALUES

NUMBER OF SAMPLE	=	1200
ACCURACY ON 95% CONFIDENCE INTERVAL	=	1.03496E+00
POINT MEAN VALUE	=	2.10000E-03
MEDIAN VALUE	=	1.68540E-03
ERROR FACTOR	=	3.02220E+00
MEAN VALUE	=	2.08947E-03
STANDARD DEVIATION	=	1.60316E-03
5% BOUND	=	5.57672E-04
95% BOUND	=	5.05613E-03

0

DISTRIBUTION CONFIDENCE LIMITS

CONFIDENCE (PERCENT)	FUNCTION VALUE
0.5	3.08592E-04
1.0	3.52549E-04
2.5	4.48240E-04
5.0	5.57672E-04
10.0	6.90646E-04
20.0	9.28801E-04
25.0	1.03674E-03
30.0	1.14591E-03
40.0	1.38548E-03
50.0	1.68540E-03
60.0	1.97055E-03
70.0	2.31367E-03
75.0	2.55980E-03
80.0	3.00483E-03
90.0	3.96435E-03
95.0	5.05613E-03
97.5	6.36085E-03
99.0	8.57124E-03
99.5	9.97013E-03

MODULE : 98.0 SEC EXECUTION TIME

END OF MODULE RUN

A.2-35

CORE REFUELING FAULT TREE	
FAILURE OF SAFETY RODS TO TRIP	
31-MAY-90	PAGE 1 OF 1

9/19/90 13:42

HD-40:PRA Codes:DROP.SA

Page 1

```
$TREE
TOP      +      CCG001RV
          "      SUBASSEMBLY      DROPPED      INTO CORE
CCG001RV C      SUBASSEMBLY      DROPPED BY      CORE GRIPPER
$END
$DATA
5.500E-04 3.0    CCG001RV
$END
```

6/21/90 14:05

HD-40:PRA Codes:DROP.trd

Page 1

TREDRA FILE CREATED BY MODULE

* OPTION

PRINTPLO 6 7 F F

END

* TREE

TOP OR 0 1 CCG001RV

END

* TITLE

CORE REFUELING ANALYSIS

SUBASSEMBLY DROPPED INTO CORE

21-JUN-90

END

* INFORM

8.50 11.00NORMAL F F 11.00 1.00

END

* EVENTS

TOP SUBASSEMBLY DROPPED INTO CORE
CCG001RV 2 SUBASSEMBLY DROPPED BY CORE GRIPPER

END

FINISH

STOP

Output from module

Thu, Jun 21, 199

```

      PROGRAM MODULE ENTERED:  21-JUN-90
ENTER THE INPUT FILE NAME  :
DROP.SA
ENTER THE OUTPUT FILE NAME :
DROP
      READ DATA           :      25.0 SEC REQUIRED
ENTER TOP EVENT NAME      :
TOP
      WRITE TREDRA INPUT   :      4.0 SEC REQUIRED
      MODULARIZATION       :      0.0 SEC REQUIRED
ENTER CUT-OFF PROBA.     :
1.0E-06
ENTER MAX. TERM NUM.     :
5
      MCS :                1 TERMS GENERATED
DO YOU WANT TO ANALYZE IMPORTANCE ?
Y
      IMPORTANCE :          4.0 SEC REQUIRED
DO YOU WANT TO ANALYZE UNCERTAINTY ?
Y
      SAMPLING :          7.0 SEC REQUIRED
      SORTING :          0.0 SEC REQUIRED
      UNCERTAINTY :        1.0 SEC REQUIRED
      BOTTOM-UP :        26.0 SEC REQUIRED
DO YOU WANT TO EXPAND MODULARIZED MCS ?
Y
      MCS :                1 TERMS GENERATED
      CUTSET :          9.0 SEC REQUIRED
DO YOU WANT TO ANALYZE IMPORTANCE ?
Y
      IMPORTANCE :          3.0 SEC REQUIRED
DO YOU WANT TO ANALYZE UNCERTAINTY ?
Y
      SAMPLING :          9.0 SEC REQUIRED
      SORTING :          1.0 SEC REQUIRED
      UNCERTAINTY :        0.0 SEC REQUIRED
      EXPAND :        22.0 SEC REQUIRED
      MODULE :        77.0 SEC REQUIRED

END OF MODULE RUN
```

STOP

MODULE VERSION 2.0 RUN AT ANL ON: 21-JUN-90

CUT SETS FOR THE TOP EVENT: TOP

PROBABILITY = 5.500E-04

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
CCG001RV	5.500E-04	1.0000	1818.1819	INF

1

NO. OF MINIMAL CUT SETS = 1

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	5.500E-04	1.0000	CCG001RV

PROBABILITY = 5.500E-04

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
CCG001RV	5.500E-04	1.0000	1818.1819	INF

1

NO. OF MINIMAL CUT SETS = 1

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	5.500E-04	1.0000	CCG001RV

MODULE VERSION 2.0 RUN AT ANL ON: 21-JUN-90

FAULT TREE INPUT

TOP + CCG001RV
 " SUBASSEMBLY DROPPED INTO CORE
 CCG001RV C SUBASSEMBLY DROPPED BY CORE GRIPPER

1

THE FOLLOWING ARE TOP EVENTS : 1

TOP

THE FOLLOWING ARE GATES : 1

TOP

THE FOLLOWING ARE EVENTS : 1

CCG001RV

RELIABILITY INPUT DATA

5.500E-04 3.0 CCG001RV
 1

MODULARIZED FAULT TREE

TOP + CCG001RV

CUT SETS FOR THE TOP EVENT: TOP

NO. OF EVENTS IN FAULT TREE = 1

CCG001RV 5.500E-04 3.000

CUTSETS : 14.0 SEC REQUIRED
 1

PROBABILITY = 5.500E-04

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
CCG001RV	5.500E-04	1.0000	1818.1819	INF

1

NO. OF MINIMAL CUT SETS = 1

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	5.500E-04	1.0000	CCG001RV

1

RESULTS SUMMARIZED

DISTRIBUTION VALUES

NUMBER OF SAMPLE	=	1200
ACCURACY ON 95% CONFIDENCE INTERVAL	=	1.03496E+00
POINT MEAN VALUE	=	5.50000E-04
MEDIAN VALUE	=	4.35914E-04
ERROR FACTOR	=	3.09291E+00
MEAN VALUE	=	5.40850E-04
STANDARD DEVIATION	=	3.92685E-04
5% BOUND	=	1.40940E-04
95% BOUND	=	1.28369E-03

0

DISTRIBUTION CONFIDENCE LIMITS

CONFIDENCE (PERCENT)	FUNCTION VALUE
0.5	7.37897E-05
1.0	8.63303E-05
2.5	1.13787E-04
5.0	1.40940E-04
10.0	1.84873E-04
20.0	2.51670E-04
25.0	2.82896E-04
30.0	3.11810E-04
40.0	3.73625E-04
50.0	4.35914E-04
60.0	5.15643E-04
70.0	6.27091E-04
75.0	6.86523E-04
80.0	7.48532E-04
90.0	1.00099E-03
95.0	1.28369E-03
97.5	1.55480E-03
99.0	1.99240E-03
99.5	2.30860E-03

1

PROBABILITY = 5.500E-04

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
CCG001RV	5.500E-04	1.0000	1818.1819	INF

A.2-41

1

NO. OF MINIMAL CUT SETS = 1

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	5.500E-04	1.0000	CCG001RV

R E S U L T S S U M M A R I Z E D

DISTRIBUTION VALUES

NUMBER OF SAMPLE	=	1200
ACCURACY ON 95% CONFIDENCE INTERVAL	=	1.03496E+00
POINT MEAN VALUE	=	5.50000E-04
MEDIAN VALUE	=	4.41413E-04
ERROR FACTOR	=	3.02220E+00
MEAN VALUE	=	5.47243E-04
STANDARD DEVIATION	=	4.19874E-04
5% BOUND	=	1.46057E-04
95% BOUND	=	1.32422E-03

0

DISTRIBUTION CONFIDENCE LIMITS

CONFIDENCE (PERCENT)	FUNCTION VALUE
0.5	8.08218E-05
1.0	9.23343E-05
2.5	1.17396E-04
5.0	1.46057E-04
10.0	1.80883E-04
20.0	2.43257E-04
25.0	2.71528E-04
30.0	3.00120E-04
40.0	3.62864E-04
50.0	4.41413E-04
60.0	5.16096E-04
70.0	6.05961E-04
75.0	6.70424E-04
80.0	7.86980E-04
90.0	1.03828E-03
95.0	1.32422E-03
97.5	1.66594E-03
99.0	2.24485E-03
99.5	2.61122E-03

MODULE : 77.0 SEC EXECUTION TIME

END OF MODULE RUN

A.2-43

CORE REFUELING ANALYSIS	
SUBASSEMBLY DROPPED INTO CORE	
21-JUN-90	PAGE 1 OF 1

APPENDIX A.3

EVENT TREE ANALYSES

- Dropped Subassembly
- Trip Signal Failure
- Safety Rod Trip Failure

***** S E Q 1 *****

SEQUENCE FOR THE INITIATING EVENT iel *Dropped Subassembly*

FAULTED TREES:

init.tree

ps.tree

drop.tree

SUCCESS TREES:

1

PROBABILITY = 2.518E-11

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
iel	1.000E+00	1.0000	1.0000	0.0000
CCG001RV	5.500E-04	1.0000	1818.1819	0.0000
FHU001RZ	7.330E-04	0.9607	1310.6552	25.4332
CFA005RL	4.000E-05	0.6667	16667.0000	3.0000
CCG005RG	2.000E-05	0.3333	16667.3340	1.5000
CND001RM	3.000E-05	0.0393	1311.5763	1.0409
CND002RW	1.000E-10	0.0000	1378.9127	1.0000

1

NO. OF MINIMAL CUT SETS = 6

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	1.613E-11	0.6405	iel CFA005RL FHU001RZ CCG001RV
2	8.063E-12	0.3202	iel CCG005RG FHU001RZ CCG001RV
3	6.600E-13	0.0262	iel CFA005RL CND001RM CCG001RV
4	3.300E-13	0.0131	iel CCG005RG CND001RM CCG001RV
5	2.200E-18	0.0000	iel CFA005RL CND002RW CCG001RV
6	1.100E-18	0.0000	iel CCG005RG CND002RW CCG001RV

IMPORTANCE : 1.2 SEC REQUIRED

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SAMPLING : 6.0 SEC REQUIRED
SORTING : 0.6 SEC REQUIRED

1

R E S U L T S S U M M A R I Z E D

DISTRIBUTION VALUES

MEAN (Calculated with 99 % of the points) 0.187E-10

NUMBER OF SAMPLE = 5000
ACCURACY ON 95% CONFIDENCE INTERVAL = 5.07023E-01
POINT MEAN VALUE = 2.51790E-11

MEDIAN VALUE = 5.62783E-12
 ERROR FACTOR = 1.78990E+01
 MEAN VALUE = 2.69645E-11
 STANDARD DEVIATION = 1.12705E-10
 5% BOUND = 3.77253E-13
 95% BOUND = 1.00733E-10

G

DISTRIBUTION CONFIDENCE LIMITS

CONFIDENCE (PERCENT),	FUNCTION VALUE
0.5	7.86542E-14
1.0	1.22042E-13
2.5	2.29575E-13
5.0	3.77253E-13
10.0	6.95661E-13
20.0	1.42768E-12
25.0	1.88154E-12
30.0	2.44445E-12
40.0	3.72366E-12
50.0	5.62783E-12
60.0	8.37268E-12
70.0	1.30402E-11
75.0	1.72049E-11
80.0	2.29074E-11
90.0	5.00025E-11
95.0	1.00733E-10
97.5	1.76130E-10
99.0	3.49334E-10
99.5	5.85939E-10
UNCERTAINTY	: 0.1 SEC REQUIRED

***** S E Q 2 *****

FILE: SEQ5 ROGLANS A1 ANLVM VM/SP 508 CMS

PAGE 00003

SEQUENCE FOR THE INITIATING EVENT ie1

FAULTED TREES:
 init.tree
 ps.tree

SUCCESS TREES:
 drop.tree

1

PROBABILITY = 4.578E-08

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
ie1	1.000E+00	1.0000	1.0000	0.0000
FRUCCO12	7.330E-04	0.9607	1310.6552	25.4332
CFAC05P1	4.110E-05	0.6667	16667.0020	3.0000
CGC00000	2.110E-05	0.3333	16667.3359	1.5000

POINT MEAN VALUE = 6.00000E-05
 MEDIAN VALUE = 2.54613E-05
 ERROR FACTOR = 8.13502E+00
 MEAN VALUE = 6.03831E-05
 STANDARD DEVIATION = 1.48534E-04
 5% BOUND = 3.79920E-06
 95% BOUND = 2.07128E-04

0

DISTRIBUTION CONFIDENCE LIMITS

CONFIDENCE FUNCTION
 (PERCENT), VALUE

FILE: SEQS ROGLANS A1 ANLVM VM/SP 508 CMS

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0.5	1.22796E-06
1.0	1.74866E-06
2.5	2.69552E-06
5.0	3.79920E-06
10.0	5.82636E-06
20.0	9.53138E-06
25.0	1.14114E-05
30.0	1.36944E-05
40.0	1.86214E-05
50.0	2.54613E-05
60.0	3.46341E-05
70.0	4.88213E-05
75.0	5.76464E-05
80.0	7.16117E-05
90.0	1.23354E-04
95.0	2.07128E-04
97.5	3.40122E-04
99.0	5.71166E-04
99.5	8.46995E-04
UNCERTAINTY	:

0.1 SEC REQUIRED

***** S E Q 4 *****

SEQUENCE FOR THE INITIATING EVENT 1e2 *I-PS-S Sequence*

FAULTED TREES:

init.tree
 ps.tree
 tr.tree

SUCCESS TREES:

1

PROBABILITY = 2.490E-13

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
1e2	1.000E+00	1.0000	1.0000	0.0000
FRUGG1RZ	2.330E-04	0.9607	1310.6552	25.4333
FRUGG2RZ	5.000E-06	0.9191	*****	12.3636

CFA005RL	4.000E-05	0.6667	16666.9980	3.0000
CCG005RG	2.000E-05	0.3333	16667.3301	1.5000
CND003RH	4.400E-07	0.0809	*****	1.0880
CND001RM	3.000E-05	0.0393	1311.5730	1.0409

1

FILE: SEQ5 ROGLANS A1 ANLVM VM/SP 508 CMS

PAGE 00007

NO. OF MINIMAL CUT SETS = 12

MINIMAL CUT SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	1.466E-13	0.5887 1e2	CFA005RL FHU001RZ FHU002RZ
2	7.330E-14	0.2943 1e2	CCG005RG FHU001RZ FHU002RZ
3	1.290E-14	0.0518 1e2	CFA005RL FHU001RZ CND003RH
4	6.450E-15	0.0259 1e2	CCG005RG FHU001RZ CND003RH
5	6.000E-15	0.0241 1e2	CFA005RL CND001RM FHU002RZ
6	3.000E-15	0.0120 1e2	CCG005RG CND001RM FHU002RZ
7	5.280E-16	0.0021 1e2	CFA005RL CND001RM CND003RH
8	2.640E-16	0.0011 1e2	CCG005RG CND001RM CND003RH
9	2.000E-20	0.0000 1e2	CFA005RL CND002RW FHU002RZ
10	1.000E-20	0.0000 1e2	CCG005RG CND002RW FHU002RZ
11	1.760E-21	0.0000 1e2	CFA005RL CND002RW CND003RH
12	8.800E-22	0.0000 1e2	CCG005RG CND002RW CND003RH

IMPORTANCE : 1.2 SEC REQUIRED

SAMPLING : 7.3 SEC REQUIRED

SORTING : 0.5 SEC REQUIRED

1

RESULTS SUMMARIZED

DISTRIBUTION VALUES

MEAN (Calculated with 99 % of the points) 0.159E-12

NUMBER OF SAMPLE	=	5000
ACCURACY ON 95% CONFIDENCE INTERVAL	=	5.07023E-01
POINT MEAN VALUE	=	2.49043E-13
MEDIAN VALUE	=	4.80708E-14
ERROR FACTOR	=	1.68544E+01
MEAN VALUE	=	2.61337E-13
STANDARD DEVIATION	=	2.68467E-12
5% BOUND	=	2.93824E-15
95% BOUND	=	8.10204E-13

0

DISTRIBUTION CONFIDENCE LIMITS

CONFIDENCE (PERCENT),	FUNCTION VALUE
0.5	6.5766E-16
1.0	1.06766E-15
2.5	1.7841E-15
5.0	2.93824E-15

4	1.260E-12	0.0111	1e2	CCG005RG	CND001RM	CSF001RW
5	8.400E-18	0.0000	1e2	CFA005RL	CND002RW	CSF001RW
6	4.200E-18	0.0000	1e2	CCG005RG	CND002RW	CSF001RW

IMPORTANCE : 0.1 SEC REQUIRED

SAMPLING : 5.8 SEC REQUIRED

SORTING : 0.5 SEC REQUIRED

1

R E S U L T S S U M M A R I Z E D

DISTRIBUTION VALUES

MEAN (Calculated with 99 % of the points) 0.676E-10

NUMBER OF SAMPLE	=	5000
ACCURACY ON 95% CONFIDENCE INTERVAL	=	5.07023E-01
POINT MEAN VALUE	=	9.61380E-11
MEDIAN VALUE	=	2.15358E-11
ERROR FACTOR	=	1.64624E+01
MEAN VALUE	=	1.01555E-10
STANDARD DEVIATION	=	4.82961E-10
5% BOUND	=	1.56976E-12
95% BOUND	=	3.54531E-10

0

DISTRIBUTION CONFIDENCE LIMITS

CONFIDENCE (PERCENT),	FUNCTION VALUE
0.5	3.63275E-13
1.0	5.17699E-13
2.5	9.43252E-13
5.0	1.56976E-12
10.0	2.67717E-12
20.0	5.34060E-12
25.0	6.98678E-12
30.0	8.86005E-12
40.0	1.40628E-11
50.0	2.15358E-11
60.0	3.17958E-11
70.0	4.88775E-11
75.0	6.17467E-11

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80.0	8.04972E-11
90.0	1.74338E-10
95.0	3.54531E-10
97.5	6.71383E-10
99.0	1.41793E-09
99.5	2.52218E-09

UNCERTAINTY : 0.1 SEC REQUIRED

***** S E Q 6 *****

10.0 5.25908E-15
 20.0 1.12815E-14
 25.0 1.49546E-14
 30.0 1.94630E-14
 40.0 3.06105E-14
 50.0 4.80708E-14
 60.0 7.58014E-14
 70.0 1.18915E-13
 75.0 1.53297E-13
 80.0 2.06987E-13
 90.0 4.28671E-13
 95.0 8.10204E-13
 97.5 1.60390E-12
 99.0 3.01260E-12
 99.5 4.73156E-12
 UNCERTAINTY : 0.1 SEC REQUIRED

***** S E Q 5 *****

SEQUENCE FOR THE INITIATING EVENT ie2

I-PS-T Sequence

FAULTED TREES:

init.tree
 ps.tree
 rt.tree

SUCCESS TREES:

tr.tree

1

PROBABILITY = 9.614E-11

EVENT DATA INFORMATION

NAME	MEAN	IMPORTANCE	RAW	RRW
CSF001RW	2.100E-03	1.0000	476.1905	0.0000
ie2	1.000E+00	1.0000	1.0000	0.0000
FHU001RZ	7.330E-04	0.9607	1310.6552	25.4332
CFA005RL	4.000E-05	0.6667	16667.0000	3.0000
CCG005RG	2.000E-05	0.3333	16667.3340	1.5000
CND001RM	3.000E-05	0.0393	1311.5757	1.0409
CND002RW	1.000E-10	0.0000	1444.5277	1.0000

1

NO. OF MINIMAL CUT SETS = 6

MINIMAL CUT SETS INFORMATION

FILE: SEQ5 ROGLANS A1 ANLVM VM/SP 508 CMS

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NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS
1	6.157E-11	0.6405 ie2	CFA005RL FHU001RZ CSF001RW
2	3.079E-11	0.3202 ie2	CCG005RG FHU001RZ CSF001RW
3	2.000E-12	0.0262 ie2	CFA005RL CND001RM CSF001RW

Appendix B
FAULT TREES

Appendix B contains all of the fault trees and input data used for quantification of the EBR-II PRA. The figures on pages B.1-nnn are associated with the initiating events. The figures on pages B.2-nnn are configurations associated with top events. The table on pages B.3-n contain the minimum cutsets of the dominant sequences.

<u>B.1 - INITIATING EVENTS</u>	<u>ASSOCIATED TOP EVENT(S)</u>	<u>PAGE</u>
<u>LCRA, LCDL</u>		
BFTF		B.1-1
Primary sodium purification system siphon		B.1-3
IHX tube and secondary storage tank failure		B.1-5
<u>LOFs</u>		
LF1D	P3055	B.1-7
LF2A	P2001	B.1-9
LF2B	P1022	B.1-14
LF2C	P1033	B.1-26
LF2D	P1055	B.1-33
LF2E	P2002	B.1-35
LF2F	P2003	B.1-56
<u>Reactivity Insertions</u>		
RISA, RISB, RISC, RISD		
	C9010	B.1-87
	C9020	B.1-92
	C9030	B.1-105

B.2 - TOP EVENTS

<u>Fault Tree TOP Events</u>	<u>Event Tree Top Events</u>	<u>Page</u>
<u>Reactor Safety System (RSS)</u>		
R00A2	RSIG	B.2-1
R00C2	RROD	B.2-52
R01A1	FSIG	B.2-114
R00C1	FROD	B.2-155
R00C3	LROD	B.2-204
R0003	MRSS	B.2-206
<u>Primary Pumps</u>		
P1000	PKRU	B.2-222
P7777	PRUN	B.2-245
<u>Secondary Na and Balance of Plant</u>		
B1000L	BPFR	B.2-247
N7000	SPTR	B.2-254
<u>Decay Heat Removal Trees</u>		
D1104L	DHRS,DRSB	B.2-256
D1104	DRLB	B.2-268
D5000	DRLA	B.2-280
D4000	DHRL	B.2-289
D6000L	DRSC	B.2-291
<u>Electrical Systems</u>		
E0100		B.2-299
E0102		B.2-301
E0105		B.2-303
E0120		B.2-306
E0132		B.2-308
E0150		B.2-311
E0200		B.2-319
E0250		B.2-327
E0300		B.2-330
E0306		B.2-332
E0308		B.2-335
E0360		B.2-338

B.2 - TOP EVENTS (Cont'd)

Fault Tree TOP Events

Page

J0100	B.2-340
J0101	B.2-342
J0201	B.2-348
J0301	B.2-354
J0500	B.2-359
J0530	B.2-364
J0550	B.2-369
J0555	B.2-371
J0560	B.2-373
J0565	B.2-375
J0570	B.2-377
J0575	B.2-379
J0580	B.2-381
L0100	B.2-383
L0200	B.2-388
L0300	B.2-390
L0400	B.2-392
L0500	B.2-394
L0510	B.2-396
L0520	B.2-398
L0530	B.2-400
U0050	B.2-402
U0084	B.2-404
U0101	B.2-406
U0150	B.2-408
U0184	B.2-410
U0201	B.2-412
ED100	B.2-414
ED102	B.2-416
ED105	B.2-418
JD500	B.2-421
JD530	B.2-426
LD100	B.2-431
LD200	B.2-436
LD500	B.2-438
LD510	B.2-440

B.2 - TOP EVENTS (Cont'd)

Fault Tree TOP Events

Page

Example of Short Term Electrical System Tree

E0200L	B.2-442
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Instrument and Plant Air Systems

I0001	B.2-450
I0200	B.2-453
I0301	B.2-461
I0310	B.2-463
I0501	B.2-468
I0900	B.2-472

Water Systems

W0001	B.2-474
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Shield Cooling System

H1000	B.2-479
H8100	B.2-537

Stack Exhaust System

Y1000	B.2-553
Y1001	B.2-561

Thimble Cooling System

T1000	B.2-569
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B.3 - MINIMAL CUT SETS OF THE DOMINANT DAMAGE SEQUENCES

B.3-1

1	3	3
BFTF		
X9000	BFTF	
	+ 0 3	XPF100RF XPF101RF XCOPMZ00
XPF100RF	0 0	1 OUTER BOUNDARY BREACH
XPF101RF	0 0	2 INNER BOUNDARY BREACH
XCOPMZ00	2 0	3 PUMP REMAINS ON
	0.720E+04	
	0.720E+04	
	1 2	0.100E-05 0.100E+02
	2 0	0.100E+00 0.300E+01 0.000E+00
	3 T	0.100E+01 0.100E+01

B.1-3

PRIMARY SODIUM PURIFICATION SYSTEM SYPHON	
BREAK	
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      3  4  4
PRIMARY SODIUM PURIFICATION SYSTEM SYPHON BREAK
X0990  PRIMARY PURIFICATION SYSTEM INDUCES LOS
      * 1 1 X1000  XPP090LF
X1000  FAILURE OF SYPHON BREAK SYSTEM
      * 1 1 X1001  XVV000RLP
X1001  FAILURE OF SYPHON BREAK LINE 1
      + 0 2 XVV001LP XVV002LP
XPP090LF 0 0 1 PIPE BREACH IN PURIFICATION SYSTEM
XVV000RLP 0 0 2 FAILURE OF REDUNDANT SYSTEM; VALVE FAILS TO OPEN
XVV001LP 0 0 3 FAILURE OF VALVE 1 TO OPEN
XVV002LP 0 0 4 FAILURE OF VALVE 2 TO OPEN
0.720E+04
0.720E+04
1 2 0.300E-06 0.100E+02
2 0 0.100E-01 0.100E+02 0.000E+00
3 0 0.100E-01 0.100E+02 0.000E+00
4 0 0.100E-01 0.100E+02 0.000E+00

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5 9 9
 IHX Tube and Secondary Storage Tank Failure
 X0001 IHX TUBE AND SECONDARY STORAGE TANK FAILURE
 * 3 1 X0002 X0003 00001 XHE001RF
 X0002 SODIUM DRAIN LINES REMAIN OPEN
 * 1 1 X0004 X0000Z00
 X0003 SODIUM STORAGE TANK CONTAINMENT FAILURE
 + 0 2 XTK001NF XRV001NK
 00001 ARGON VENT SYSTEM FAILS
 * 0 2 OTG001OE OTG002OE
 X0004 DRAIN VALVES FAIL TO CLOSE
 * 0 3 X0011Z00 X0022Z00 X0033Z00
 XHE001RF 0 0 1 IXH TUBE FAILURE
 X0000Z00 2 0 2 DRAIN VALVES OPENED
 XTK001NF 0 0 3 SODIUM STORAGE TANK RUPTURES
 XRV001NK 0 0 4 PRESSURE RELIEF ALLOWS SODIUM OVERFLOW; OPENS AND REMAINS SO
 OTG001OE 0 0 5 VENT LINE TO IHX OUTLET CLOSED
 OTG002OE 0 0 6 VENT LINE TO IHX INLET CLOSED
 X0011Z00 2 0 7 IHX OUTLET DRAIN VALVE OPEN
 X0022Z00 2 0 8 SURGE TANK DRAIN VALVE OPEN
 X0033Z00 2 0 9 EM PUMP DRAIN VALVE OPEN
 0.720E+04
 0.720E+04
 1 2 0.100E-05 0.100E+02
 2 T 0.100E+01 0.100E+01
 3 2 0.100E-05 0.100E+02
 4 0 0.100E-02 0.100E+02 0.000E+00
 5 2 0.250E-06 0.100E+02 0.830E-01
 6 0 0.100E+01 0.100E+01 0.000E+00
 7 T 0.100E+01 0.100E+01
 8 T 0.100E+01 0.100E+01
 9 T 0.100E+01 0.100E+01

Primary Pumps System	
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1	2	2		
Primary Pumps System				
P3055	Pump 1 or Pump 2 Shaft Fails			LF1D
	+ 0 2 PPM001RR PPM002RR			
PPM001RR	0 0 1 Primary Pump 1 Shaft Broken or Seized			
PPM002RR	0 0 2 Primary Pump 2 Shaft Broken or Seized			
	0.720E+04			
	0.100E+00			
	1 2 0.190E-05 0.840E+01			
	2 1 0.190E-05 0.840E+01			

B.1-9

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11 13 13
Primary Pumps System
P0010 Loss of MG Set 1 Motor Output
      + 1 1 P0011 PM0001PW
P0011 Loss of 2400 V Supply to MG Set 1 Motor
      + 0 4 J0301 JSG201VO PWR201YB PWR201YC
P0110 Loss of MG Set 2 Motor Output
      + 1 1 P0111 PM0002PW
P0111 Loss of 2400 V Supply to MG Set 2 Motor
      + 0 4 J0301 JSG202VO PWR202YB PWR202YC
P1011 MG 1 and 2 Motor Fails
      + 1 1 P2011 JSG212VU
P1014 MG 1 Motor Fails, then Pump 2 Shaft Binds
      * 1 1 P0010 PPM002RG
P1041 Pump 1 Shaft Binds, then MG 2 Motor Fails
      * 1 1 P0110 PPM001RG
P1044 Pump 1 and 2 Shaft Bind
      + 1 1 P2044 PPM012RU
P2001 Double Pump LOF: Slow LF2A
      + 4 2 P1011 P1014 P1041 P1044 JSG212VU PPM012RU
P2011 MG 1 Motor Fails, then MG 2 Motor
      * 2 0 P0010 P0110
P2044 Pump 1 Shaft Binds, then Pump 2 Shaft Binds
      * 0 2 PPM001RG PPM002RG
J0301 3 0 1 Loss of 2400 V Bus Power
PM0001PW 0 0 2 MG Set 1 Motor Failure
JSG201VO 0 0 3 MG Set 1 Input Breaker Fails Open
PWR201YB 0 0 4 2400 V Feeder to MG Set 1 Fails Open
PWR201YC 0 0 5 2400 V Feeder to MG Set 1 Short Circuit
PM0002PW 0 0 6 MG Set 2 Motor Failure
JSG202VO 0 0 7 MG Set 2 Input Breaker Fails Open
PWR202YB 0 0 8 2400 V Feeder to MG Set 2 Fails Open
PWR202YC 0 0 9 2400 V Feeder to MG Set 2 Short Circuit
PPM001RG 0 0 10 Primary Pump 1 Shaft Binding
PPM012RU 0 0 11 Primary Pump 1 and 2 Shaft Binding, Common Cause
PPM002RG 0 0 12 Primary Pump 2 Shaft Binding
JSG212VU 0 0 13 MG Set 1 and 2 Input Breaker Fails Open, Common Cause
0.720E+04
0.100E+00
1 0 0.100E-09 0.100E+02 0.000E+00
2 2 0.300E-04 0.300E+01
3 2 0.300E-06 0.100E+02
4 2 0.300E-05 0.100E+02
5 2 0.300E-06 0.100E+02
6 1 0.300E-04 0.300E+01
7 1 0.300E-06 0.100E+02
8 1 0.300E-05 0.100E+02
9 1 0.300E-06 0.100E+02
10 2 0.290E-04 0.260E+01
11 2 0.290E-05 0.100E+02
12 1 0.290E-04 0.260E+01
13 2 0.300E-07 0.100E+02

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B.1-15

B.1-18

Primary Pumps System	
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B.1-21

Primary Pumps System	
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Primary Pumps System

P0020 Loss of MG Set 1 Clutch Coupling
 + 1 1 P0021 PCL001PW
 P0021 Loss of Output from HP Power Supply No. 1
 + 2 2 P0022 P0035 PDC001PB PSC001PB
 P0022 Loss of Power to HP Power Supply No. 1
 + 4 2 P0023 P0024 P0025 P0028 PKS001PB PWR01APB
 P0023 Loss of Constant Power Panel 1 Supply
 + 1 4 P0029 U0050 UCB125QO PWR022YB PWR022YC
 P0024 Primary Pump START/STOP Control Failure
 * 2 0 P0026 P0027
 P0025 Relay EX1 Failure
 + 0 2 PREEX1PB PCZX15PB
 P0026 Primary Pump STOP Switches or Wiring Failure
 + 0 3 PSW01APB PSW02ACB PWR02APB
 P0027 Relay CPC-2 Failure
 + 0 3 PREP2AQB PCZCP6QB PWR03AQB
 P0028 Relay PBS Failure
 + 0 3 PREPBSPB PCZPB1PB PCZPB8PB
 P0029 Loss of CPS, STOP Switch Short Circuit, Pump 1
 + 0 2 PSW01APD PSW02ACD
 P0035 Controller Module No. 1 Signal Error
 + 0 3 PSC01BPW PST001PW PSC01CPW
 P0120 Loss of MG Set 2 Clutch Coupling
 + 1 1 P0121 PCL002PW
 P0121 Loss of Output from HP Power Supply No. 2
 + 2 2 P0122 P0135 PDC002PB PSC002PB
 P0122 Loss of Power to HP Power Supply No. 2
 + 3 1 P0123 P0124 P0125 PWR01BPB
 P0123 Loss of Constant Power Panel 2 Supply
 + 1 4 P0129 U0150 UCB217QO PWR122YB PWR122YC
 P0124 Primary Pump START/STOP Control Failure
 * 1 2 P0126 PCZCP7QB PWR03BQB
 P0125 Relay EX2 Failure
 + 0 3 PREEX2PB PCZX24PB PCZX25PB
 P0126 Primary Pump STOP Switches or Wiring Failure
 + 0 3 PSW01BPB PSW02BCB PWR02BPB
 P0129 Loss of CPS, STOP Switch Short Circuit, Pump 2
 + 0 2 PSW01BPD PSW02BCD
 P0135 Controller Module No. 2 Signal Error
 + 0 3 PSC02BPW PST002PW PSC02CPW
 P1022 MG 1 and 2 Clutch Fail LF2B
 + 2 4 P2022 P4022 PSW012PU PDC012PU PSW012CU PWR012PU
 P2022 MG 1 Clutch Fails, then MG 2 Clutch
 * 2 0 P0020 P0120
 P4022 CPC-2 Common Cause Failures
 * 0 2 PWR022PU PWR032QU
 U0050 3 0 1 LOSS OF 120 VAC CONTINUOUS POWER PANEL No.1
 U0150 3 0 1 LOSS OF 120 VAC CONTINUOUS POWER PANEL No.2
 PCL001PW 0 0 2 MG Set 1 Clutch Failure
 PDC001PB 0 0 3 HP Power Supply No. 1 Failure
 PSC001PB 0 0 4 MG Set Controller 1 Overspeed Cutoff
 PKS001PB 0 0 5 Primary Pump Key Switch Fails Open
 PWR01APB 0 0 6 Primary Pump 1 START/STOP Circuit Wiring Fails Open
 UCB125QO 0 0 7 CPS Panel 1 Breaker 25 Fails Open
 PWR022YB 0 0 8 CPS Feeder to HP Power Supply No. 1 Fails Open
 PWR022YC 0 0 9 CPS Feeder to HP Power Supply No. 1 Short Circuit
 PREEX1PB 0 0 10 Relay EX1 Coil Fails Open
 PCZX15PB 0 0 11 Contacts 5 of Relay EX1 Fail Open

PSW01APB 0 0 12 Primary Panel STOP Switch Fails Open, Pump 1
 PSW02ACB 0 0 13 Control Room STOP Switch Fails Open, Pump 1
 PWR02APB 0 0 14 Wiring between STOP Switches Fails Open, Pump 1
 PREP2AQB 0 0 15 Relay CPC-2 Coil Fails Open
 PC2CP6QB 0 0 16 Contacts 6 of Relay CPC-2 Fail Open
 PWR03AQB 0 0 17 Wiring to/from CPC-2 Relay Fails Open, Pump 1
 PREPBSPB 0 0 18 Relay PBS Coil Fails Open
 PC2PB1PB 0 0 19 Contacts 1 of Relay PBS Fail Open
 PC2PB8PB 0 0 20 Contacts 8 of Relay PBS Fail Open
 PSW01APD 0 0 21 Primary Panel STOP Switch Short, Loss of CPS, Pump 1
 PSW02ACD 0 0 22 Control Room STOP Switch Short, Loss of CPS, Pump 1
 PDC012PU 0 0 23 HP Power Supply No. 1 and 2 Fails, Common Cause
 PSC01BPW 0 0 24 MG Set 1 Controller Faulty Demand Signal
 PST001PW 0 0 25 No. 1 Feedback Signal Circuit Fails With High Signal
 PSC01CPW 0 0 26 DCS No. 1 Fails with Low Demand Signal
 PCL002PW 0 0 27 MG Set 2 Clutch Failure
 PDC002PB 0 0 28 HP Power Supply No. 2 Failure
 PSC002PB 0 0 29 MG Set Controller 2 Overspeed Cutoff
 PWR01BPB 0 0 30 Primary Pump 2 START/STOP Circuit Wiring Fails Open
 UCB217QO 0 0 31 CPS Panel 2 Breaker 17 Fails Open
 PWR122YB 0 0 32 CPS Feeder to HP Power Supply No. 2 Fails Open
 PWR122YC 0 0 33 CPS Feeder to HP Power Supply No. 2 Short Circuit
 PC2CP7QB 0 0 34 Contacts 7 of Relay CPC-2 Fail Open
 PWR03BQB 0 0 35 Wiring to/from CPC-2 Relay Fails Open, Pump 2
 PREEX2PB 0 0 36 Relay EX2 Coil Fails Open
 PCZX24PB 0 0 37 Contacts 4 of Relay EX2 Fail Open
 PCZX25PB 0 0 38 Contacts 5 of Relay EX2 Fail Open
 PSW01BPB 0 0 39 Primary Panel STOP Switch Fails Open, Pump 2
 PSW02BCB 0 0 40 Control Room STOP Switch Fails Open, Pump 2
 PWR02BPB 0 0 41 Wiring between STOP Switches Fails Open, Pump 2
 PSW01BPD 0 0 42 Primary Panel STOP Switch Short, Loss of CPS, Pump 2
 PSW02BCD 0 0 43 Control Room STOP Switch Short, Loss of CPS, Pump 2
 PSC02BPW 0 0 44 MG Set 2 Controller Faulty Demand Signal
 PST002PW 0 0 45 No. 2 Feedback Signal Circuit Fails With High Signal
 PSC02CPW 0 0 46 DCS No. 2 Fails with Low Demand Signal
 PSW012CU 0 0 47 Control Room STOP Switch Short, Common Cause Loss of CPS
 PSW012PU 0 0 48 Primary Panel STOP Switch Short, Common Cause Loss of CPS
 PWR012PU 0 0 49 Pumps 1 & 2 START/STOP Circuit Wiring Fail Open Common Cause
 PWR022PU 0 0 50 Wiring between STOP Switches Fails Open, Pumps 1 & 2 Common
 PWR032QU 0 0 51 Wiring to/from CPC-2 Relay Fails Open, Pumps 1 & 2 Common

0.720E+04

0.100E+00

1	0	0.100E-09	0.100E+02	0.000E+00
2	2	0.500E-04	0.300E+02	
3	2	0.300E-05	0.100E+02	
4	2	0.300E-05	0.100E+02	
5	2	0.100E-06	0.100E+02	
6	2	0.300E-05	0.100E+02	
7	2	0.300E-06	0.100E+02	
8	2	0.300E-05	0.100E+02	
9	2	0.300E-06	0.100E+02	
10	2	0.500E-06	0.100E+02	
11	2	0.100E-06	0.100E+02	
12	2	0.100E-06	0.100E+02	
13	2	0.100E-06	0.100E+02	
14	2	0.300E-05	0.100E+02	
15	2	0.500E-06	0.100E+02	
16	2	0.100E-06	0.100E+02	
17	2	0.600E-05	0.100E+02	
18	2	0.500E-06	0.100E+02	

19	2	0.100E-06	0.100E+02
20	2	0.100E-06	0.100E+02
21	2	0.100E-06	0.100E+02
22	2	0.100E-06	0.100E+02
23	2	0.300E-06	0.100E+02
24	2	0.300E-05	0.100E+02
25	2	0.300E-05	0.100E+02
26	2	0.300E-05	0.100E+02
27	1	0.500E-04	0.300E+02
28	1	0.300E-05	0.100E+02
29	1	0.300E-05	0.100E+02
30	1	0.300E-05	0.100E+02
31	1	0.300E-06	0.100E+02
32	1	0.300E-05	0.100E+02
33	1	0.300E-06	0.100E+02
34	2	0.100E-06	0.100E+02
35	2	0.600E-05	0.100E+02
36	1	0.500E-06	0.100E+02
37	1	0.100E-06	0.100E+02
38	1	0.100E-06	0.100E+02
39	2	0.100E-06	0.100E+02
40	2	0.100E-06	0.100E+02
41	2	0.300E-05	0.100E+02
42	2	0.100E-06	0.100E+02
43	2	0.100E-06	0.100E+02
44	1	0.300E-05	0.100E+02
45	1	0.300E-05	0.100E+02
46	1	0.300E-05	0.100E+02
47	2	0.100E-07	0.100E+02
48	2	0.100E-07	0.100E+02
49	2	0.300E-06	0.100E+02
50	2	0.300E-06	0.100E+02
51	2	0.600E-06	0.100E+02

Primary Pumps System	
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Primary Pumps System

P0039 Primary Pump 1 Motor Failure
 + 1 1 P0040 RMO001RR
 P0040 Loss of MG Set 1 Generator Output
 + 2 1 P0041 P0051 PGE001PW
 P0041 Loss of MG Set 1 Generator Exciter Current
 + 2 2 P0042 P0043 PEX001PW PDC006PB
 P0042 Exciter to MG Set 1 Generator Wiring Failure
 + 0 2 PWR042PB PWR042PC
 P0043 Loss of Power to MG Set 1 Exciter Power Supply
 + 1 2 P0044 U0050 UCB125QO
 P0044 CPS to MG Set 1 Exciter Wiring Failure
 + 0 2 PWR044PB PWR044PC
 P0051 MG Set 1 Output Wiring to Pump 1 Failure
 + 0 3 PCB001PO PWR051TB PWR051TC
 P0139 Primary Pump 2 Motor Failure
 + 1 1 P0140 RMO002RR
 P0140 Loss of MG Set 2 Generator Output
 + 2 1 P0141 P0151 PGE002PW
 P0141 Loss of MG Set 2 Generator Exciter Current
 + 2 2 P0142 P0143 PEX002PW PDC007PB
 P0142 Exciter to MG Set 2 Generator Wiring Failure
 + 0 2 PWR142PB PWR142PC
 P0143 Loss of Power to MG Set 2 Exciter Power Supply
 + 1 2 P0144 U0150 UCB217QO
 P0144 CPS to MG Set 2 Exciter Wiring Failure
 + 0 2 PWR144PB PWR144PC
 P0151 MG Set 2 Output Wiring to Pump 2 Failure
 + 0 3 PCB002PO PWR151TB PWR151TC
 P1033 Pump 1 and 2 Motor Fail LF2C
 + 1 1 P2033 PCB012PU
 P2033 Pump 1 Motor Fails, then Pump 2
 * 2 0 P0039 P0139
 U0050 3 0 1 LOSS OF 120 VAC CONTINUOUS POWER PANEL No.1
 U0150 3 0 1 LOSS OF 120 VAC CONTINUOUS POWER PANEL No.2
 UCB125QO 0 0 2 CPS Panel 1 Breaker 25 Fails Open
 RMO001RR 0 0 3 Primary Pump 1 Motor Fails to Run
 PGE001PW 0 0 4 MG Set 1 Generator Failure
 PEX001PW 0 0 5 MG Set 1 Exciter Failure
 PDC006PB 0 0 6 Exciter No. 1 Power Supply Failure
 PWR042PB 0 0 7 Exciter No. 1 Wiring to Generator Fails Open
 PWR042PC 0 0 8 Exciter No. 1 Wiring to Generator Short Circuit
 PWR044PB 0 0 9 CPS Feeder to Exciter No. 1 PS Fails Open
 PWR044PC 0 0 10 CPS Feeder to Exciter No. 1 PS Short Circuit
 PCB001PO 0 0 11 MG Set 1 Output Breaker Fails Open
 PWR051TB 0 0 12 MG Set 1 Wiring to Pump 1 Fails Open
 PWR051TC 0 0 13 MG Set 1 Wiring to Pump 1 Short Circuit
 UCB217QO 0 0 14 CPS Panel 2 Breaker 17 Fails Open
 RMO002RR 0 0 15 Primary Pump 2 Motor Fails to Run
 PGE002PW 0 0 16 MG Set 2 Generator Failure
 PEX002PW 0 0 17 MG Set 2 Exciter Failure
 PDC007PB 0 0 18 Exciter No. 2 Power Supply Failure
 PWR142PB 0 0 19 Exciter No. 2 Wiring to Generator Fails Open
 PWR142PC 0 0 20 Exciter No. 2 Wiring to Generator Short Circuit
 PWR144PB 0 0 21 CPS Feeder to Exciter No. 2 PS Fails Open
 PWR144PC 0 0 22 CPS Feeder to Exciter No. 2 PS Short Circuit
 PCB002PO 0 0 23 MG Set 2 Output Breaker Fails Open
 PWR151TB 0 0 24 MG Set 2 Wiring to Pump 2 Fails Open
 PWR151TC 0 0 25 MG Set 2 Wiring to Pump 2 Short Circuit

PCB012PU 0 0 26 MG Set 1 and 2 Output Breaker Fails Open, Common Cause

0.720E+04

0.100E+00

1	0	0.100E-09	0.100E+02	0.000E+00
2	2	0.300E-06	0.100E+02	
3	2	0.200E-04	0.300E+01	
4	2	0.300E-05	0.100E+02	
5	2	0.300E-05	0.100E+02	
6	2	0.300E-05	0.100E+02	
7	2	0.300E-05	0.100E+02	
8	2	0.300E-06	0.100E+02	
9	2	0.300E-05	0.100E+02	
10	2	0.300E-06	0.100E+02	
11	2	0.300E-06	0.100E+02	
12	2	0.300E-05	0.100E+02	
13	2	0.300E-06	0.100E+02	
14	1	0.300E-06	0.100E+02	
15	1	0.200E-04	0.300E+01	
16	1	0.300E-05	0.100E+02	
17	1	0.300E-05	0.100E+02	
18	1	0.300E-05	0.100E+02	
19	1	0.300E-05	0.100E+02	
20	1	0.300E-06	0.100E+02	
21	1	0.300E-05	0.100E+02	
22	1	0.300E-06	0.100E+02	
23	1	0.300E-06	0.100E+02	
24	1	0.300E-05	0.100E+02	
25	1	0.300E-06	0.100E+02	
26	2	0.300E-07	0.100E+02

Primary Pumps System	
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1 2 2
Primary Pumps System
P1055 Pump 1 Shaft Fails, then Pump 2 Shaft Fails LF2D
* 0 2 PPM001RR PPM002RR
PPM001RR 0 0 1 Primary Pump 1 Shaft Broken or Seized
PPM002RR 0 0 2 Primary Pump 2 Shaft Broken or Seized
0.720E+04
0.100E+00
1 2 0.190E-05 0.840E+01
2 1 0.190E-05 0.840E+01

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Primary Pumps System	
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Primary Pumps System	
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Primary Pumps System

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Primary Pumps System	
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Primary Pumps System

P0010 Loss of MG Set 1 Motor Output
+ 1 1 P0011 PMO001PW

P0011 Loss of 2400 V Supply to MG Set 1 Motor
+ 0 4 J0301 JSG201VO PWR201YB PWR201YC

P0020 Loss of MG Set 1 Clutch Coupling
+ 1 1 P0021 PCL001PW

P0021 Loss of Output from HP Power Supply No. 1
+ 2 2 P0022 P0035 PDC001PB PSC001PB

P0022 Loss of Power to HP Power Supply No. 1
+ 4 2 P0023 P0024 P0025 P0028 PKS001PB PWR01APB

P0023 Loss of Constant Power Panel 1 Supply
+ 1 4 P0029 U0050 UCB125QO PWR022YB PWR022YC

P0024 Primary Pump START/STOP Control Failure
* 2 0 P0026 P0027

P0025 Relay EX1 Failure
+ 0 2 PREEX1PB PCZX15PB

P0026 Primary Pump STOP Switches or Wiring Failure
+ 0 3 PSW01APB PSW02ACB PWR02APB

P0027 Relay CPC-2 Failure
+ 0 3 PREP2AQB PCZCP6QB PWR03AQB

P0028 Relay PBS Failure
+ 0 3 PREPBSB PCZPB1PB PCZPB8PB

P0029 Loss of CPS, STOP Switch Short Circuit, Pump 1
+ 0 2 PSW01APD PSW02ACD

P0035 Controller Module No. 1 Signal Error
+ 0 3 PSC01BPW PST001PW PSC01CPW

P0039 Primary Pump 1 Motor Failure
+ 1 1 P0040 RMO001RR

P0040 Loss of MG Set 1 Generator Output
+ 2 1 P0041 P0051 PGE001PW

P0041 Loss of MG Set 1 Generator Exciter Current
+ 2 2 P0042 P0043 PEX001PW PDC006PB

P0042 Exciter to MG Set 1 Generator Wiring Failure
+ 0 2 PWR042PB PWR042PC

P0043 Loss of Power to MG Set 1 Exciter Power Supply
+ 1 2 P0044 U0050 UCB125QO

P0044 CPS to MG Set 1 Exciter Wiring Failure
+ 0 2 PWR044PB PWR044PC

P0051 MG Set 1 Output Wiring to Pump 1 Failure
+ 0 3 PCB001PO PWR051TB PWR051TC

P0110 Loss of MG Set 2 Motor Output
+ 1 1 P0111 PMO002PW

P0111 Loss of 2400 V Supply to MG Set 2 Motor
+ 0 4 J0301 JSG202VO PWR202YB PWR202YC

P0120 Loss of MG Set 2 Clutch Coupling
+ 1 1 P0121 PCL002PW

P0121 Loss of Output from HP Power Supply No. 2
+ 2 2 P0122 P0135 PDC002PB PSC002PB

P0122 Loss of Power to HP Power Supply No. 2
+ 3 1 P0123 P0124 P0125 PWR01BPB

P0123 Loss of Constant Power Panel 2 Supply
+ 1 4 P0129 U0150 UCB217QO PWR122YB PWR122YC

P0124 Primary Pump START/STOP Control Failure
* 1 2 P0126 PCZCP7QB PWR03BQB

P0125 Relay EX2 Failure
+ 0 3 PREEX2PB PCZX24PB PCZX25PB

P0126 Primary Pump STOP Switches or Wiring Failure
+ 0 3 PSW01BPB PSW02BCB PWR02BPB

P0129	Loss of CPS, STOP Switch Short Circuit, Pump 2	
	+ 0 2 PSW01BPD PSW02BCD	
P0135	Controller Module No. 2 Signal Error	
	+ 0 3 PSC02BPW PST002PW PSC02CPW	
P0139	Primary Pump 2 Motor Failure	
	+ 1 1 P0140 RMO02RR	
P0140	Loss of MG Set 2 Generator Output	
	+ 2 1 P0141 P0151 PGE002PW	
P0141	Loss of MG Set 2 Generator Exciter Current	
	+ 2 2 P0142 P0143 PEX002PW PDC007PB	
P0142	Exciter to MG Set 2 Generator Wiring Failure	
	+ 0 2 PWR142PB PWR142PC	
P0143	Loss of Power to MG Set 2 Exciter Power Supply	
	+ 1 2 P0144 U0150 UCB217QO	
P0144	CPS to MG Set 2 Exciter Wiring Failure	
	+ 0 2 PWR144PB PWR144PC	
P0151	MG Set 2 Output Wiring to Pump 2 Failure	
	+ 0 3 PCB002PO PWR151TB PWR151TC	
P1015	MG 1 Motor Fails, then Pump 2 Shaft Fails	
	* 1 1 P0010 PPM002RR	
P1025	MG 1 Clutch Fails, then Pump 2 Shaft Fails	
	* 1 1 P0020 PPM002RR	
P1035	Pump 1 Motor Fails, then Pump 2 Shaft Fails	
	* 1 1 P0039 PPM002RR	
P1045	Pump 1 Shaft Binds, then Pump 2 Shaft Fails	
	* 0 2 PPM001RG PPM002RR	
P1051	Pump 1 Shaft Fails, then MG 2 Motor	
	* 1 1 P0110 PPM001RR	
P1052	Pump 1 Shaft Fails, then MG 2 Clutch	
	* 1 1 P0120 PPM001RR	
P1053	Pump 1 Shaft Fails, then Pump 2 Motor	
	* 1 1 P0139 PPM001RR	
P1054	Pump 1 Shaft Fails, then Pump 2 Shaft Binds	
	* 0 2 PPM002RG PPM001RR	
P2002	Double Pump LOF: Mixed seize & other LF2E	
	+ 2 0 P2004 P2005	
P2004	Half of P2002	
	+ 4 0 P1015 P1025 P1035 P1045	
P2005	Half of P2002	
	+ 4 0 P1051 P1052 P1053 P1054	
J0301	3 0 1 Loss of 2400 V Bus Power	
U0050	3 0 1 LOSS OF 120 VAC CONTINUOUS POWER PANEL No.1	
U0150	3 0 1 LOSS OF 120 VAC CONTINUOUS POWER PANEL No.2	
PM0001PW	0 0 2 MG Set 1 Motor Failure	
JSG201VO	0 0 3 MG Set 1 Input Breaker Fails Open	
PWR201YB	0 0 4 2400 V Feeder to MG Set 1 Fails Open	
PWR201YC	0 0 5 2400 V Feeder to MG Set 1 Short Circuit	
PCL001PW	0 0 6 MG Set 1 Clutch Failure	
PDC001PB	0 0 7 HP Power Supply No. 1 Failure	
PSC001PB	0 0 8 MG Set Controller 1 Overspeed Cutoff	
PKS001PB	0 0 9 Primary Pump Key Switch Fails Open	
PWR01APB	0 0 10 Primary Pump 1 START/STOP Circuit Wiring Fails Open	
UCB125QO	0 0 11 CPS Panel 1 Breaker 25 Fails Open	
PWR022YC	0 0 12 CPS Feeder to HP Power Supply No. 1 Fails Open	
PWR022YC	0 0 13 CPS Feeder to HP Power Supply No. 1 Short Circuit	
PREEX1PB	0 0 14 Relay EX1 Coil Fails Open	
PCZX15PB	0 0 15 Contacts 5 of Relay EX1 Fail Open	
PSW01APB	0 0 16 Primary Panel STOP Switch Fails Open, Pump 1	
PSW02ACB	0 0 17 Control Room STOP Switch Fails Open, Pump 1	
PWR02APB	0 0 18 Wiring between STOP Switches Fails O, Pump 1	

PREP2AOB	0	0	19 Relay CPC-2 Coil Fails Open
PCZCP6QB	0	0	20 Contacts 6 of Relay CPC-2 Fail Open
PWR03AOB	0	0	21 Wiring to/from CPC-2 Relay Fails Open, Pump 1
PREPBSPB	0	0	22 Relay PBS Coil Fails Open
PCZPB1PB	0	0	23 Contacts 1 of Relay PBS Fail Open
PCZPB8PB	0	0	24 Contacts 8 of Relay PBS Fail Open
PSW01APD	0	0	25 Primary Panel STOP Switch Short, Loss of CPS, Pump 1
PSW02ACD	0	0	26 Control Room STOP Switch Short, Loss of CPS, Pump 1
PSC01BPW	0	0	27 MG Set 1 Controller Faulty Demand Signal
PST001PW	0	0	28 No. 1 Feedback Signal Circuit Fails With High Signal
PSC01CPW	0	0	29 DCS No. 1 Fails With Low Demand Signal
RMO01RR	0	0	30 Primary Pump 1 Motor Fails to Run
PGE001PW	0	0	31 MG Set 1 Generator Failure
PEX001PW	0	0	32 MG Set 1 Exciter Failure
PDC006PB	0	0	33 Exciter No. 1 Power Supply Failure
PWR042PB	0	0	34 Exciter No. 1 Wiring to Generator Fails Open
PWR042PC	0	0	35 Exciter No. 1 Wiring to Generator Short Circuit
PWR044PB	0	0	36 CPS Feeder to Exciter No. 1 PS Fails Open
PWR044PC	0	0	37 CPS Feeder to Exciter No. 1 PS Short Circuit
PCH001PO	0	0	38 MG Set 1 Output Breaker Fails Open
PWR051TB	0	0	39 MG Set 1 Wiring to Pump 1 Fails Open
PWR051TC	0	0	40 MG Set 1 Wiring to Pump 1 Short Circuit
PMO002PW	0	0	41 MG Set 2 Motor Failure
JSC202VO	0	0	42 MG Set 2 Input Breaker Fails Open
PWR202YB	0	0	43 2400 V Feeder to MG Set 2 Fails Open
PWR202YC	0	0	44 2400 V Feeder to MG Set 2 Short Circuit
PCI0002PW	0	0	45 MG Set 2 Clutch Failure
PDC002PB	0	0	46 HP Power Supply No. 2 Failure
PSC002PB	0	0	47 MG Set Controller 2 Overspeed Cutoff
PWR01BPB	0	0	48 Primary Pump 2 START/STOP Circuit Wiring Fails Open
UCB217QO	0	0	49 CPS Panel 2 Breaker 17 Fails Open
PWR122YB	0	0	50 CPS Feeder to HP Power Supply No. 2 Fails Open
PWR122YC	0	0	51 CPS Feeder to HP Power Supply No. 2 Short Circuit
PCZCP7QB	0	0	52 Contacts 7 of Relay CPC-2 Fail Open
PWR03BQB	0	0	53 Wiring to/from CPC-2 Relay Fails Open, Pump 2
PREX22PB	0	0	54 Relay EX2 Coil Fails Open
PCZX24PB	0	0	55 Contacts 4 of Relay EX2 Fail Open
PCZX25PB	0	0	56 Contacts 5 of Relay EX2 Fail Open
PSW01BPB	0	0	57 Primary Panel STOP Switch Fails Open, Pump 2
PSW02BCB	0	0	58 Control Room STOP Switch Fails Open, Pump 2
PWR02BPB	0	0	59 Wiring between STOP Switches Fails Open, Pump 2
PSW01BPD	0	0	60 Primary Panel STOP Switch Short, Loss of CPS, Pump 2
PSW02BCD	0	0	61 Control Room STOP Switch Short, Loss of CPS, Pump 2
PSC02BPW	0	0	62 MG Set 2 Controller Faulty Demand Signal
PST002PW	0	0	63 No. 2 Feedback Signal Circuit Fails With High Signal
PSC02CPW	0	0	64 DCS No. 2 Fails With Low Demand Signal
RMO002RR	0	0	65 Primary Pump 2 Motor Fails to Run
PGE002PW	0	0	66 MG Set 2 Generator Failure
PEX002PW	0	0	67 MG Set 2 Exciter Failure
PDC007PB	0	0	68 Exciter No. 2 Power Supply Failure
PWR142PB	0	0	69 Exciter No. 2 Wiring to Generator Fails Open
PWR142PC	0	0	70 Exciter No. 2 Wiring to Generator Short Circuit
PWR144PB	0	0	71 CPS Feeder to Exciter No. 2 PS Fails Open
PWR144PC	0	0	72 CPS Feeder to Exciter No. 2 PS Short Circuit
PCH002PO	0	0	73 MG Set 2 Output Breaker Fails Open
PWR151TB	0	0	74 MG Set 2 Wiring to Pump 2 Fails Open
PWR151TC	0	0	75 MG Set 2 Wiring to Pump 2 Short Circuit
PPM001RG	0	0	76 Primary Pump 1 Shaft Binding
PPM001RR	0	0	77 Primary Pump 1 Shaft Broken or Seized
PPM002RG	0	0	78 Primary Pump 2 Shaft Binding

PPM002RR 0 0 79 Primary Pump 2 Shaft Broken or Seized

0.720E+04

0.100E+00

0.000E+00

1	0	0.100E-09	0.100E+02	0.000E+00
2	2	0.300E-04	0.300E+01	
3	2	0.300E-06	0.100E+02	
4	2	0.300E-05	0.100E+02	
5	2	0.300E-06	0.100E+02	
6	2	0.500E-04	0.300E+02	
7	2	0.300E-05	0.100E+02	
8	2	0.300E-05	0.100E+02	
9	2	0.100E-06	0.100E+02	
10	2	0.300E-05	0.100E+02	
11	2	0.300E-06	0.100E+02	
12	2	0.300E-05	0.100E+02	
13	2	0.300E-06	0.100E+02	
14	2	0.500E-06	0.100E+02	
15	2	0.100E-06	0.100E+02	
16	2	0.100E-06	0.100E+02	
17	2	0.100E-06	0.100E+02	
18	2	0.300E-05	0.100E+02	
19	2	0.500E-06	0.100E+02	
20	2	0.100E-06	0.100E+02	
21	2	0.600E-05	0.100E+02	
22	2	0.500E-06	0.100E+02	
23	2	0.100E-06	0.100E+02	
24	2	0.100E-06	0.100E+02	
25	2	0.100E-06	0.100E+02	
26	2	0.100E-06	0.100E+02	
27	2	0.300E-05	0.100E+02	
28	2	0.300E-05	0.100E+02	
29	2	0.300E-05	0.100E+02	
30	2	0.200E-04	0.300E+01	
31	2	0.300E-05	0.100E+02	
32	2	0.300E-05	0.100E+02	
33	2	0.300E-05	0.100E+02	
34	2	0.300E-05	0.100E+02	
35	2	0.300E-06	0.100E+02	
36	2	0.300E-05	0.100E+02	
37	2	0.300E-06	0.100E+02	
38	2	0.300E-06	0.100E+02	
39	2	0.300E-05	0.100E+02	
40	2	0.300E-06	0.100E+02	
41	1	0.300E-04	0.300E+01	
42	1	0.300E-06	0.100E+02	
43	1	0.300E-05	0.100E+02	
44	1	0.300E-06	0.100E+02	
45	1	0.500E-04	0.300E+02	
46	1	0.300E-05	0.100E+02	
47	1	0.300E-05	0.100E+02	
48	1	0.300E-05	0.100E+02	
49	1	0.300E-06	0.100E+02	
50	1	0.300E-05	0.100E+02	
51	1	0.300E-06	0.100E+02	
52	2	0.100E-06	0.100E+02	
53	2	0.600E-05	0.100E+02	
54	1	0.500E-06	0.100E+02	
55	1	0.100E-06	0.100E+02	
56	1	0.100E-06	0.100E+02	
57	2	0.100E-06	0.100E+02	

58 2 0.100E-06 0.100E+02
59 2 0.300E-05 0.100E+02
60 2 0.100E-06 0.100E+02
61 2 0.100E-06 0.100E+02
62 1 0.300E-05 0.100E+02
63 1 0.300E-05 0.100E+02
64 1 0.300E-05 0.100E+02
65 1 0.200E-04 0.300E+01
66 1 0.300E-05 0.100E+02
67 1 0.300E-05 0.100E+02
68 1 0.300E-05 0.100E+02
69 1 0.300E-05 0.100E+02
70 1 0.300E-06 0.100E+02
71 1 0.300E-05 0.100E+02
72 1 0.300E-06 0.100E+02
73 1 0.300E-06 0.100E+02
74 1 0.300E-05 0.100E+02
75 1 0.300E-06 0.100E+02
76 2 0.290E-04 0.260E+01
77 2 0.190E-05 0.840E+01
78 1 0.290E-04 0.260E+01
79 1 0.190E-05 0.840E+01

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Primary Pumps System

P0010 Loss of MG Set 1 Motor Output
 + 1 1 P0011 PMO001PW
 P0011 Loss of 2400 V Supply to MG Set 1 Motor
 + 0 4 J0301 JSG201VO PWR201YB PWR201YC
 P0020 Loss of MG Set 1 Clutch Coupling
 + 1 1 P0021 PCL001PW
 P0021 Loss of Output from HP Power Supply No. 1
 + 2 2 P0022 P0035 PDC001PB PSC001PB
 P0022 Loss of Power to HP Power Supply No. 1
 + 4 2 P0023 P0024 P0025 P0028 PKS001PB PWR01APB
 P0023 Loss of Constant Power Panel 1 Supply
 + 1 4 P0029 U0050 UCB125QO PWR022YB PWR022YC
 P0024 Primary Pump START/STOP Control Failure
 + 2 0 P0026 P0027
 P0025 Relay EX1 Failure
 + 0 2 PREX1PB PCZX15PB
 P0026 Primary Pump STOP Switches or Wiring Failure
 + 0 3 PSW01APB PSW02ACB PWR02APB
 P0027 Relay CPC-2 Failure
 + 0 3 PREP2AQB PCZCP6QB PWR03AQB
 P0028 Relay PBS Failure
 + 0 3 PREPBSB PCZPB1PB PCZPB8PB
 P0029 Loss of CPS, STOP Switch Short Circuit, Pump 1
 + 0 2 PSW01APD PSW02ACD
 P0035 Controller Module No. 1 Signal Error
 + 0 3 PSC01BPW PST001PW PSC01CPW
 P0039 Primary Pump 1 Motor Failure
 + 1 1 P0040 RMO001RR
 P0040 Loss of MG Set 1 Generator Output
 + 2 1 P0041 P0051 PGE001PW
 P0041 Loss of MG Set 1 Generator Exciter Current
 + 2 2 P0042 P0043 PEX001PW PDC006PB
 P0042 Exciter to MG Set 1 Generator Wiring Failure
 + 0 2 PWR042PB PWR042PC
 P0043 Loss of Power to MG Set 1 Exciter Power Supply
 + 1 2 P0044 U0050 UCB125QO
 P0044 CPS to MG Set 1 Exciter Wiring Failure
 + 0 2 PWR044PB PWR044PC
 P0051 MG Set 1 Output Wiring to Pump 1 Failure
 + 0 3 PCB001PO PWR051TB PWR051TC
 P0110 Loss of MG Set 2 Motor Output
 + 1 1 P0111 PMO002PW
 P0111 Loss of 2400 V Supply to MG Set 2 Motor
 + 0 4 J0301 JSG202VO PWR202YB PWR202YC
 P0120 Loss of MG Set 2 Clutch Coupling
 + 1 1 P0121 PCL002PW
 P0121 Loss of Output from HP Power Supply No. 2
 + 2 2 P0122 P0135 PDC002PB PSC002PB
 P0122 Loss of Power to HP Power Supply No. 2
 + 3 1 P0123 P0124 P0125 PWR01BPB
 P0123 Loss of Constant Power Panel 2 Supply
 + 1 4 P0129 U0150 UCB217QO PWR122YB PWR122YC
 P0124 Primary Pump START/STOP Control Failure
 + 1 2 P0126 PCZCP7QB PWR03BQB
 P0125 Relay EX2 Failure
 + 0 3 PREX2PB PCZX24PB PCZX25PB
 P0126 Primary Pump STOP Switches or Wiring Failure
 + 0 3 PSW01BPB PSW02BCB PWR02BPB

P0129	Loss of CPS, STOP Switch Short Circuit, Pump 2			
P0135	+ 0 2 PSW01BPD PSW02BCD Controller Module No. 2 Signal Error			
P0139	+ 0 3 PSC02BPW PST002PW PSC02CPW Primary Pump 2 Motor Failure			
P0140	+ 1 1 P0140 RMO002RR Loss of MG Set 2 Generator Output			
P0141	+ 2 1 P0141 P0151 PGE002PW Loss of MG Set 2 Generator Exciter Current			
P0142	+ 2 2 P0142 P0143 PEX002PW PDC007PB Exciter to MG Set 2 Generator Wiring Failure			
P0143	+ 0 2 PWR142PB PWR142PC Loss of Power to MG Set 2 Exciter Power Supply			
P0144	+ 1 2 P0144 U0150 UCB217QO CPS to MG Set 2 Exciter Wiring Failure			
P0151	+ 0 2 PWR144PB PWR144PC MG Set 2 Output Wiring to Pump 2 Failure			
P1012	+ 0 3 PCB002PO PWR151TB PWR151TC MG 1 Motor Fails, then MG 2 Clutch			
P1013	* 2 0 P0010 P0120 MG 1 Motor Fails, then Pump 2 Motor			
P1014	* 2 0 P0010 P0139 MG 1 Motor Fails, then Pump 2 Shaft Binds			
P1021	* 1 1 P0010 PPM002RG MG 1 Clutch Fails, then MG 2 Motor			
P1023	* 2 0 P0020 P0110 MG 1 Clutch Fails, then Pump 2 Motor			
P1024	* 2 0 P0020 P0139 MG 1 Clutch Fails, then Pump 2 Shaft Binds			
P1031	* 1 1 P0020 PPM002RG Pump 1 Motor Fails, then MG 2 Motor			
P1032	* 2 0 P0039 P0110 Pump 1 Motor Fails, then MG 2 Clutch			
P1034	* 2 0 P0039 P0120 Pump 1 Motor Fails, then Pump 2 Shaft Binds			
P1041	* 1 1 P0039 PPM002RG Pump 1 Shaft Binds, then MG 2 Motor Fails			
P1042	* 1 1 P0110 PPM001RG Pump 1 Shaft Binds, then MG 2 Clutch Fails			
P1043	* 1 1 P0120 PPM001RG Pump 1 Shaft Binds, then Pump 2 Motor Fails			
P2003	* 1 1 P0139 PPM001RG Double Pump LOF: Other mixed non-seize LF2F			
P2006	+ 2 0 P2006 P2007 Half of P2003			
P2007	+ 6 0 P1012 P1013 P1014 P1023 P1024 P1034 Half of P2003			
J0301	+ 6 0 P1021 P1031 P1032 P1041 P1042 P1043 3 0 1 Loss of 2400 V Bus Power			
U0050	3 0 1 LOSS OF 120 VAC CONTINUOUS POWER PANEL No.1			
U0150	3 0 1 LOSS OF 120 VAC CONTINUOUS POWER PANEL No.2			
PM0001PW	0 0 2 MG Set 1 Motor Failure			
JSG201VO	0 0 3 MG Set 1 Input Breaker Fails Open			
PWR201YB	0 0 4 2400 V Feeder to MG Set 1 Fails Open			
PWR201YC	0 0 5 2400 V Feeder to MG Set 1 Short Circuit			
PCL001PW	0 0 6 MG Set 1 Clutch Failure			
PDC001PB	0 0 7 HP Power Supply No. 1 Failure			
PSC001PB	0 0 8 MG Set Controller 1 Overspeed Cutoff			
PKS001PB	0 0 9 Primary Pump Key Switch Fails Open			
PWR01APB	0 0 10 Primary Pump 1 START/STOP Circuit Wiring Fails Open			

UCB125QO	0	0	11 CPS Panel 1 Breaker 25 Fails Open
PWR022YB	0	0	12 CPS Feeder to HP Power Supply No. 1 Fails Open
PWR022YC	0	0	13 CPS Feeder to HP Power Supply No. 1 Short Circuit
PREEX1PB	0	0	14 Relay EX1 Coil Fails Open
PCZX15PB	0	0	15 Contacts 5 of Relay EX1 Fail Open
PSW01APB	0	0	16 Primary Panel STOP Switch Fails Open, Pump 1
PSW02ACB	0	0	17 Control Room STOP Switch Fails Open, Pump 1
PWR02APB	0	0	18 Wiring between STOP Switches Fails Open, Pump 1
PREP2AQB	0	0	19 Relay CPC-2 Coil Fails Open
PCZCP6QB	0	0	20 Contacts 6 of Relay CPC-2 Fail Open
PWR03AQB	0	0	21 Wiring to/from CPC-2 Relay Fails Open, Pump 1
PREPBSPB	0	0	22 Relay PBS Coil Fails Open
PCZPB1PB	0	0	23 Contacts 1 of Relay PBS Fail Open
PCZPB8PB	0	0	24 Contacts 8 of Relay PBS Fail Open
PSW01APD	0	0	25 Primary Panel STOP Switch Short, Loss of CPS, Pump 1
PSW02ACD	0	0	26 Control Room STOP Switch Short, Loss of CPS, Pump 1
PSC01BFW	0	0	27 MG Set 1 Controller Faulty Demand Signal
PST001PW	0	0	28 No. 1 Feedback Signal Circuit Fails With High Signal
PSC01CPW	0	0	29 DCS No. 1 Fails With Low Demand Signal
RMO001RR	0	0	30 Primary Pump 1 Motor Fails to Run
PGE001PW	0	0	31 MG Set 1 Generator Failure
PEX001PW	0	0	32 MG Set 1 Exciter Failure
PDC006PB	0	0	33 Exciter No. 1 Power Supply Failure
PWR042PB	0	0	34 Exciter No. 1 Wiring to Generator Fails Open
PWR042PC	0	0	35 Exciter No. 1 Wiring to Generator Short Circuit
PWR044PB	0	0	36 CPS Feeder to Exciter No. 1 PS Fails Open
PWR044PC	0	0	37 CPS Feeder to Exciter No. 1 PS Short Circuit
PCB001PO	0	0	38 MG Set 1 Output Breaker Fails Open
PWR051PB	0	0	39 MG Set 1 Wiring to Pump 1 Fails Open
PWR051TC	0	0	40 MG Set 1 Wiring to Pump 1 Short Circuit
PMO002PW	0	0	41 MG Set 2 Motor Failure
JSG202VO	0	0	42 MG Set 2 Input Breaker Fails Open
PWR202YB	0	0	43 2400 V Feeder to MG Set 2 Fails Open
PWR202YC	0	0	44 2400 V Feeder to MG Set 2 Short Circuit
PCL002PW	0	0	45 MG Set 2 Clutch Failure
PDC002PB	0	0	46 HP Power Supply No. 2 Failure
PSC002PB	0	0	47 MG Set Controller 2 Overspeed Cutoff
PWR01BPB	0	0	48 Primary Pump 2 START/STOP Circuit Wiring Fails Open
UCB217QO	0	0	49 CPS Panel 2 Breaker 17 Fails Open
PWR122YB	0	0	50 CPS Feeder to HP Power Supply No. 2 Fails Open
PWR122YC	0	0	51 CPS Feeder to HP Power Supply No. 2 Short Circuit
PCZCP7QB	0	0	52 Contacts 7 of Relay CPC-2 Fail Open
PWR03BQB	0	0	53 Wiring to/from CPC-2 Relay Fails Open, Pump 2
PREEX2PB	0	0	54 Relay EX2 Coil Fails Open
PCZX24PB	0	0	55 Contacts 4 of Relay EX2 Fail Open
PCZX25PB	0	0	56 Contacts 5 of Relay EX2 Fail Open
PSW01BPB	0	0	57 Primary Panel STOP Switch Fails Open, Pump 2
PSW02BCB	0	0	58 Control Room STOP Switch Fails Open, Pump 2
PWR02BPB	0	0	59 Wiring between STOP Switches Fails Open, Pump 2
PSW01BPD	0	0	60 Primary Panel STOP Switch Short, Loss of CPS, Pump 2
PSW02BCD	0	0	61 Control Room STOP Switch Short, Loss of CPS, Pump 2
PSC02BFW	0	0	62 MG Set 2 Controller Faulty Demand Signal
PST002PW	0	0	63 No. 2 Feedback Signal Circuit Fails With High Signal
PSC02CPW	0	0	64 DCS No. 2 Fails With Low Demand Signal
RMO002RR	0	0	65 Primary Pump 2 Motor Fails to Run
PGE002PW	0	0	66 MG Set 2 Generator Failure
PEX002PW	0	0	67 MG Set 2 Exciter Failure
PDC007PB	0	0	68 Exciter No. 2 Power Supply Failure
PWR142PB	0	0	69 Exciter No. 2 Wiring to Generator Fails Open
PWR142PC	0	0	70 Exciter No. 2 Wiring to Generator Short Circuit

PWRI44PB 0 0 71 CPS Feeder to Exciter No. 2 PS Fails C
 PWRI44PC 0 0 72 CPS Feeder to Exciter No. 2 PS Short Circuit
 PCB002PO 0 0 73 MG Set 2 Output Breaker Fails Open
 PWRI51TB 0 0 74 MG Set 2 Wiring to Pump 2 Fails Open
 PWRI51TC 0 0 75 MG Set 2 Wiring to Pump 2 Short Circuit
 PPM001RG 0 0 76 Primary Pump 1 Shaft Binding
 PPM002RG 0 0 77 Primary Pump 2 Shaft Binding

0.720E+04
 0.100E+00
 1 0 0.100E-09 0.100E+02 0.000E+00
 2 2 0.300E-04 0.300E+01
 3 2 0.300E-06 0.100E+02
 4 2 0.300E-05 0.100E+02
 5 2 0.300E-06 0.100E+02
 6 2 0.500E-04 0.300E+02
 7 2 0.300E-05 0.100E+02
 8 2 0.300E-05 0.100E+02
 9 2 0.100E-06 0.100E+02
 10 2 0.300E-05 0.100E+02
 11 2 0.300E-06 0.100E+02
 12 2 0.300E-05 0.100E+02
 13 2 0.300E-06 0.100E+02
 14 2 0.500E-06 0.100E+02
 15 2 0.100E-06 0.100E+02
 16 2 0.100E-06 0.100E+02
 17 2 0.100E-06 0.100E+02
 18 2 0.300E-05 0.100E+02
 19 2 0.500E-06 0.100E+02
 20 2 0.100E-06 0.100E+02
 21 2 0.600E-05 0.100E+02
 22 2 0.500E-06 0.100E+02
 23 2 0.100E-06 0.100E+02
 24 2 0.100E-06 0.100E+02
 25 2 0.100E-06 0.100E+02
 26 2 0.100E-06 0.100E+02
 27 2 0.300E-05 0.100E+02
 28 2 0.300E-05 0.100E+02
 29 2 0.300E-05 0.100E+02
 30 2 0.200E-04 0.300E+01
 31 2 0.300E-05 0.100E+02
 32 2 0.300E-05 0.100E+02
 33 2 0.300E-05 0.100E+02
 34 2 0.300E-05 0.100E+02
 35 2 0.300E-06 0.100E+02
 36 2 0.300E-05 0.100E+02
 37 2 0.300E-06 0.100E+02
 38 2 0.300E-06 0.100E+02
 39 2 0.300E-05 0.100E+02
 40 2 0.300E-06 0.100E+02
 41 1 0.300E-04 0.300E+01
 42 1 0.300E-06 0.100E+02
 43 1 0.300E-05 0.100E+02
 44 1 0.300E-06 0.100E+02
 45 1 0.500E-04 0.300E+02
 46 1 0.300E-05 0.100E+02
 47 1 0.300E-05 0.100E+02
 48 1 0.300E-05 0.100E+02
 49 1 0.300E-06 0.100E+02
 50 1 0.300E-05 0.100E+02
 51 1 0.300E-06 0.100E+02

52 2 0.100E-06 0.100E+02
53 2 0.600E-05 0.100E+02
54 1 0.500E-06 0.100E+02
55 1 0.100E-06 0.100E+02
56 1 0.100E-06 0.100E+02
57 2 0.100E-06 0.100E+02
58 2 0.100E-06 0.100E+02
59 2 0.300E-05 0.100E+02
60 2 0.100E-06 0.100E+02
61 2 0.100E-06 0.100E+02
62 1 0.300E-05 0.100E+02
63 1 0.300E-05 0.100E+02
64 1 0.300E-05 0.100E+02
65 1 0.200E-04 0.300E+01
66 1 0.300E-05 0.100E+02
67 1 0.300E-05 0.100E+02
68 1 0.300E-05 0.100E+02
69 1 0.300E-05 0.100E+02
70 1 0.300E-06 0.100E+02
71 1 0.300E-05 0.100E+02
72 1 0.300E-06 0.100E+02
73 1 0.300E-06 0.100E+02
74 1 0.300E-05 0.100E+02
75 1 0.300E-06 0.100E+02
76 2 0.290E-04 0.260E+01
77 1 0.290E-04 0.260E+01

14 20 20
ONE ROD RUNS IN
C9010 ONE ROD RUNS IN C9010 C9200 C9300 C9400
+ 4 0 C9100 C9200 C9300 C9400
C9100 MSW SWITCH HELD UP BY HUMAN ERROR
+ 2 0 C9110 C9120
C9200 MSW SWITCH FAILS CLOSED
+ 2 1 C9210 C9220 C9300 C9400
C9300 CU OR SU RELAY FAILS CLOSED
+ 0 2 CRECSUQP CRECSUQQ
C9400 ACRDS MOTOR CONTROLLER INDUCED INSERTION
+ 2 0 C9410 C9420
C9110 UNAUTHORIZED PERSON HOLDS MSW UP
* 0 3 C9100CZ C9110CZ C9120CZ C9130CZ C9140CZ
C9120 OPERATOR HOLDS MSW UP
* 0 2 C9120CZ C9130CZ C9140CZ C9150CZ C9160CZ
C9210 SPONTANEOUS MSW-1 CLOSURE COMPLETES CIRCUIT
* 0 2 C9210CZ C9220CZ C9230CZ C9240CZ C9250CZ
C9220 MSW FAILS TO REOPEN
* 0 2 C9220CZ C9230CZ C9240CZ C9250CZ C9260CZ
C9410 CONTROLLER FAILS WHILE RELAY OPEN, NO RECOVERY
* 0 2 C9410CZ C9420CZ C9430CZ C9440CZ C9450CZ
C9420 CONTROLLER FAILS WHILE RELAY CLOSED
+ 2 0 C9411 C9421
C9411 OPERATOR FAILS TO OPEN RELAY, THEN CONTROLLER FAILS
* 0 2 C9411CZ C9421CZ C9431CZ C9441CZ C9451CZ
C9421 RELAY CLOSED BY SWITCH OR RELAY FAILURE, THEN CONTROLLER FAI
* 1 1 C9422 C9423 C9424 C9425 C9426
C9422 SWITCH OR RELAY FAILS CLOSED
+ 0 4 C9422CZ C9423CZ C9424CZ C9425CZ C9426CZ
C9423 CRECSUQP 0 0 1 MSW-2 CLOSURES SPONTANEOUSLY
CRECSUQP 0 0 2 CU OR SU CONTACT FAILS TO REOPEN
CRECSUQP 0 0 3 CU OR SU RELAY CLOSURES SPONTANEOUSLY
C9424 C9425 C9426 C9427 C9428 C9429 C9430 C9431 C9432 C9433 C9434 C9435 C9436 C9437 C9438 C9439 C9440 C9441 C9442 C9443 C9444 C9445 C9446 C9447 C9448 C9449 C9450 C9451 C9452 C9453 C9454 C9455 C9456 C9457 C9458 C9459 C9460 C9461 C9462 C9463 C9464 C9465 C9466 C9467 C9468 C9469 C9470 C9471 C9472 C9473 C9474 C9475 C9476 C9477 C9478 C9479 C9480 C9481 C9482 C9483 C9484 C9485 C9486 C9487 C9488 C9489 C9490 C9491 C9492 C9493 C9494 C9495 C9496 C9497 C9498 C9499 C9500 C9501 C9502 C9503 C9504 C9505 C9506 C9507 C9508 C9509 C9510 C9511 C9512 C9513 C9514 C9515 C9516 C9517 C9518 C9519 C9520 C9521 C9522 C9523 C9524 C9525 C9526 C9527 C9528 C9529 C9530 C9531 C9532 C9533 C9534 C9535 C9536 C9537 C9538 C9539 C9540 C9541 C9542 C9543 C9544 C9545 C9546 C9547 C9548 C9549 C9550 C9551 C9552 C9553 C9554 C9555 C9556 C9557 C9558 C9559 C9560 C9561 C9562 C9563 C9564 C9565 C9566 C9567 C9568 C9569 C9570 C9571 C9572 C9573 C9574 C9575 C9576 C9577 C9578 C9579 C9580 C9581 C9582 C9583 C9584 C9585 C9586 C9587 C9588 C9589 C9590 C9591 C9592 C9593 C9594 C9595 C9596 C9597 C9598 C9599 C9600 C9601 C9602 C9603 C9604 C9605 C9606 C9607 C9608 C9609 C9610 C9611 C9612 C9613 C9614 C9615 C9616 C9617 C9618 C9619 C9620 C9621 C9622 C9623 C9624 C9625 C9626 C9627 C9628 C9629 C9630 C9631 C9632 C9633 C9634 C9635 C9636 C9637 C9638 C9639 C9640 C9641 C9642 C9643 C9644 C9645 C9646 C9647 C9648 C9649 C9650 C9651 C9652 C9653 C9654 C9655 C9656 C9657 C9658 C9659 C9660 C9661 C9662 C9663 C9664 C9665 C9666 C9667 C9668 C9669 C9670 C9671 C9672 C9673 C9674 C9675 C9676 C9677 C9678 C9679 C9680 C9681 C9682 C9683 C9684 C9685 C9686 C9687 C9688 C9689 C9690 C9691 C9692 C9693 C9694 C9695 C9696 C9697 C9698 C9699 C9700 C9701 C9702 C9703 C9704 C9705 C9706 C9707 C9708 C9709 C9710 C9711 C9712 C9713 C9714 C9715 C9716 C9717 C9718 C9719 C9720 C9721 C9722 C9723 C9724 C9725 C9726 C9727 C9728 C9729 C9730 C9731 C9732 C9733 C9734 C9735 C9736 C9737 C9738 C9739 C9740 C9741 C9742 C9743 C9744 C9745 C9746 C9747 C9748 C9749 C9750 C9751 C9752 C9753 C9754 C9755 C9756 C9757 C9758 C9759 C9760 C9761 C9762 C9763 C9764 C9765 C9766 C9767 C9768 C9769 C9770 C9771 C9772 C9773 C9774 C9775 C9776 C9777 C9778 C9779 C9780 C9781 C9782 C9783 C9784 C9785 C9786 C9787 C9788 C9789 C9790 C9791 C9792 C9793 C9794 C9795 C9796 C9797 C9798 C9799 C9800 C9801 C9802 C9803 C9804 C9805 C9806 C9807 C9808 C9809 C9810 C9811 C9812 C9813 C9814 C9815 C9816 C9817 C9818 C9819 C9820 C9821 C9822 C9823 C9824 C9825 C9826 C9827 C9828 C9829 C9830 C9831 C9832 C9833 C9834 C9835 C9836 C9837 C9838 C9839 C9840 C9841 C9842 C9843 C9844 C9845 C9846 C9847 C9848 C9849 C9850 C9851 C9852 C9853 C9854 C9855 C9856 C9857 C9858 C9859 C9860 C9861 C9862 C9863 C9864 C9865 C9866 C9867 C9868 C9869 C9870 C9871 C9872 C9873 C9874 C9875 C9876 C9877 C9878 C9879 C9880 C9881 C9882 C9883 C9884 C9885 C9886 C9887 C9888 C9889 C9890 C9891 C9892 C9893 C9894 C9895 C9896 C9897 C9898 C9899 C9900 C9901 C9902 C9903 C9904 C9905 C9906 C9907 C9908 C9909 C9910 C9911 C9912 C9913 C9914 C9915 C9916 C9917 C9918 C9919 C9920 C9921 C9922 C9923 C9924 C9925 C9926 C9927 C9928 C9929 C9930 C9931 C9932 C9933 C9934 C9935 C9936 C9937 C9938 C9939 C9940 C9941 C9942 C9943 C9944 C9945 C9946 C9947 C9948 C9949 C9950 C9951 C9952 C9953 C9954 C9955 C9956 C9957 C9958 C9959 C9960 C9961 C9962 C9963 C9964 C9965 C9966 C9967 C9968 C9969 C9970 C9971 C9972 C9973 C9974 C9975 C9976 C9977 C9978 C9979 C9980 C9981 C9982 C9983 C9984 C9985 C9986 C9987 C9988 C9989 C9990 C9991 C9992 C9993 C9994 C9995 C9996 C9997 C9998 C9999

9	3	0.167E-03	0.300E+01	0.180E+02
10	1	0.700E-07	0.200E+01	
11	0	0.140E-05	0.330E+01	0.240E+05
12	0	0.300E+00	0.200E+01	0.000E+00
13	1	0.400E-06	0.400E+01	
14	0	0.110E+00	0.300E+01	0.000E+00
15	3	0.167E-03	0.300E+01	0.180E+02
16	5	0.400E-06	0.400E+01	0.720E+04
17	0	0.100E-04	0.500E+01	0.400E+03
18	1	0.850E-07	0.100E+02	
19	0	0.300E-05	0.200E+01	0.400E+03
20	1	0.500E-06	0.100E+02

B.1-100

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TWO RODS RUN IN	
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31 26 26
TWO RODS RUN IN
C9020 TWO RODS RUN IN C9600 C9700 C9800 C9900 C9990
+ 6 0 C9500 C9600 C9700 C9800 C9900 C9990
C9500 2-ROD SHORT OF SELECTOR SWITCH AND MSW STAYS CLOSED
* 1 1 C9510 C9520 C9530 C9540 C9550 C9560 C9570 C9580 C9590 C9600
C9600 OPERATOR ERROR INSERTS SECOND ROD IN RESPONSE TO FIRST
* 2 0 C9610 C9620 C9630 C9640 C9650 C9660 C9670 C9680 C9690 C9700
C9700 MSW FAILURE AND 1 RELAY OR ACRDS CONTROLLER FAILS
* 2 0 C9710 C9720 C9730 C9740 C9750 C9760 C9770 C9780 C9790 C9800
C9800 1 CU OR SU RELAY FAILS CLOSED AND ACRDS FAILS
* 2 0 C9810 C9820 C9830 C9840 C9850 C9860 C9870 C9880 C9890 C9900
C9900 BOTH MSW SWITCHES FAIL CLOSED AND 2 RODS SELECTED
* 1 3 C9910 C9920 C9930 C9940 C9950 C9960 C9970 C9980 C9990 C9990
C9990 2 CU RELAYS OR 1 CU AND 1 SU RELAY FAIL CLOSED
* 1 1 C9991 C9992 C9993 C9994 C9995 C9996 C9997 C9998 C9999 C9999
C9510 MSW STAYS CLOSED
+ 2 0 C9100 C9200 C9300 C9400 C9500 C9600 C9700 C9800 C9900 C9990
C9610 OPERATOR INSERTS SECOND ROD; ERRONEOUS CORRECTIVE ACTION
* 0 2 CHUOPICL C9610 C9620 C9630 C9640 C9650 C9660 C9670 C9680 C9690 C9700
C9620 1 ROD RUNS IN: RELAY, CONTROLLER, OR COMPOUND MSW FAILURE
+ 3 0 C9621 C9622 C9623 C9624 C9625 C9626 C9627 C9628 C9629 C9630
C9710 MSW SWITCH FAILS CLOSED WITHIN 0.1 h
* 1 1 C9200 C9200 C9200 C9200 C9200 C9200 C9200 C9200 C9200 C9200
C9720 1 RELAY OR ACRDS CONTROLLER FAILS
+ 1 1 C9400 CRECSUQQ
C9400 ACRDS MOTOR CONTROLLER INDUCED INSERTION
+ 2 0 C9410 C9420 C9430 C9440 C9450 C9460 C9470 C9480 C9490 C9500
C9810 1 CU OR SU RELAY FAILS CLOSED WITHIN 0.1 h
* 1 1 C9300 CREDDUMQI
C9910 BOTH MSW SWITCHES FAIL CLOSED
* 1 1 C9911 C9912 C9913 C9914 C9915 C9916 C9917 C9918 C9919 C9920
C9991 2 CU (SU) RELAYS FAIL CLOSED
* 1 1 C9300 CRECSUQQ
C9100 MSW SWITCH HELD UP BY HUMAN ERROR
+ 2 0 C9110 C9120 C9130 C9140 C9150 C9160 C9170 C9180 C9190 C9200
C9200 MSW SWITCH FAILS CLOSED
+ 2 1 C9210 C9220 C9230 C9240 C9250 C9260 C9270 C9280 C9290 C9300
C9621 MSW CLOSED AND INTERLOCK BETWEEN MSW# FAILS
* 1 1 C9622 C9623 C9624 C9625 C9626 C9627 C9628 C9629 C9630 C9640
C9300 CU OR SU RELAY FAILS CLOSED
+ 0 2 CRECSUQQ CRECSUQQ
C9410 CONTROLLER FAILS WHILE RELAY OPEN, NO RECOVERY
* 0 2 CSTMC1QD C9410 C9420 C9430 C9440 C9450 C9460 C9470 C9480 C9490 C9500
C9420 CONTROLLER FAILS WHILE RELAY CLOSED
+ 2 0 C9411 C9421 C9431 C9441 C9451 C9461 C9471 C9481 C9491 C9501
C9911 MSW FAILS TO REOPEN OR CLOSURES SPONTANEOUSLY
+ 1 1 C9220 C9220 C9220 C9220 C9220 C9220 C9220 C9220 C9220 C9220
C9110 UNAUTHORIZED PERSON HOLDS MSW UP
* 0 3 C9110 C9110 C9110 C9110 C9110 C9110 C9110 C9110 C9110 C9110
C9120 OPERATOR HOLDS MSW UP
* 0 2 C9120 C9120 C9120 C9120 C9120 C9120 C9120 C9120 C9120 C9120
C9210 SPONTANEOUS MSW-1 CLOSURE COMPLETES CIRCUIT
* 0 2 C9210 C9210 C9210 C9210 C9210 C9210 C9210 C9210 C9210 C9210
C9220 MSW FAILS TO REOPEN
* 0 2 C9220 C9220 C9220 C9220 C9220 C9220 C9220 C9220 C9220 C9220
C9622 MSW CLOSED
+ 2 0 C9100 C9200 C9300 C9400 C9500 C9600 C9700 C9800 C9900 C9990
C9411 OPERATOR FAILS TO OPEN RELAY, THEN CONTROLLER FAILS
* 0 2 C9411 C9411 C9411 C9411 C9411 C9411 C9411 C9411 C9411 C9411

C9421 RELAY CLOSED BY SWITCH OR RELAY FAILURE, THEN CONTROLLER FAI
 * 1 1 C9422 CSTMC2QD

C9422 SWITCH OR RELAY FAILS CLOSED
 + 0 4 CSWPSBCP CSWPSBCQ CREKD7QP CREKD7QQ
 CSWSS2CD 0 0 1 2-ROD SHORT OF SELECTOR SWITCH
 CSWSS2CL 0 0 2 MSSW NOT SET TO NEUTRAL POSITION
 CSWDUMCI 0 0 3 SWITCH EVENT MUST OCCUR WITHIN 0.1 h
 CSWMS2CP 0 0 4 INTERLOCK BETWEEN MSW's FAILS
 CREKUNQI 0 0 5 RELAY EVENT MUST OCCUR WITHIN 0.1 h
 CHUOPICL 0 0 6 OPERATOR DECIDES TO COMPENSATE WITH 2nd ROD
 CSWMS4CZ 0 0 7 OPERATOR HOLDS MSW UP INSTEAD OF DOWN
 CRECSUQQ 0 0 8 CU OR SU RELAY CLOSES SPONTANEOUSLY
 CSWMS1CQ 0 0 9 MSW-1 CLOSES SPONTANEOUSLY
 CSWMS2CQ 0 0 10 MSW-2 CLOSES SPONTANEOUSLY
 CRECSUQP 0 0 11 CU OR SU CONTACT FAILS TO REOPEN
 CSTMC1QD 0 0 12 CONTROLLER SENDS UP-VOLTAGE SPONTANEOUSLY
 CSWMS3CZ 0 0 13 OP. CLOSES RELAY, FAILS TO COMPENSATE W/CONTROLLING ROD
 CSWCONCZ 0 0 14 CONSOLE ACCESS RESTRICTION VIOLATED
 CSWMS1CZ 0 0 15 MSW SWITCH HELD UP BY UNAUTHORIZED PERSON
 CSWMS1CL 0 0 16 OPERATORS FAIL TO FREE MSW FRM UNAUTHORIZED PERSON
 CSWMS2CZ 0 0 17 OPERATOR HOLDS UP MSW UNTIL ALARMS
 CSWMS2CL 0 0 18 OPERATORS FAIL TO RELEASE MSW WITHIN 1 MIN AFTER ALARM
 CSWMSWCP 0 0 19 MSW STAYS UP WHEN RELEASED
 CSWMSNCZ 0 0 20 OPERATOR FAILS TO FREE STUCK SWITCH
 CSWPSBCL 0 0 21 OPERATOR FAILS TO PUSH PSB2
 CSTMC2QD 0 0 22 CONTROLLER FAILS AFTER SWITCH OR RELAY
 CSWPSBCP 0 0 23 PSB2 SWITCH FAILS TO OPEN
 CSWPSBCQ 0 0 24 PSB2 SWITCH CLOSES SPONTANEOUSLY
 CREKD7QP 0 0 25 K7 RELAY FAILS TO REOPEN
 CREKD7QQ 0 0 26 K7 RELAY CLOSES SPONTANEOUSLY
 0.720E+04
 0.240E+02

1 5 0.146E-05 0.150E+01 0.720E+04
 2 3 0.167E-03 0.300E+01 0.180E+02
 3 0 0.139E-04 0.100E+01 0.000E+00
 4 0 0.100E-03 0.100E+02 0.000E+00
 5 0 0.139E-04 0.100E+01 0.000E+00
 6 0 0.100E+00 0.500E+01 0.000E+00
 7 0 0.250E-01 0.200E+01 0.000E+00
 8 1 0.143E-05 0.840E+01
 9 1 0.700E-07 0.200E+01
 10 1 0.700E-07 0.200E+01
 11 0 0.660E-06 0.840E+01 0.240E+05
 12 1 0.400E-06 0.400E+01
 13 0 0.110E+00 0.300E+01 0.000E+00
 14 0 0.500E-01 0.500E+01 0.000E+00
 15 0 0.210E+00 0.200E+01 0.000E+00
 16 0 0.500E-01 0.200E+01 0.000E+00
 17 0 0.200E-01 0.840E+01 0.000E+00
 18 0 0.100E-02 0.500E+01 0.000E+00
 19 0 0.140E-05 0.330E+01 0.240E+05
 20 0 0.300E+00 0.200E+01 0.000E+00
 21 3 0.167E-03 0.300E+01 0.180E+02
 22 5 0.400E-06 0.400E+01 0.720E+04
 23 0 0.100E-04 0.500E+01 0.400E+03
 24 1 0.850E-07 0.100E+02
 25 0 0.300E-05 0.200E+01 0.400E+03
 26 1 0.500E-06 0.100E+02

9	12	12	
MANY RODS RUN IN			
C9030	MANY RODS RUN IN		
	+ 1 1	C9031	CARCRPRD
C9031	MANY-ROD SHORT OF SELECTOR SWITCH AND MSW STAYS CLOSED		
	* 1 1	C9510	CSWSS3CD
C9510	MSW STAYS CLOSED		
	+ 2 0	C9100	C9200
C9100	MSW SWITCH HELD UP BY HUMAN ERROR		
	+ 2 0	C9110	C9120
C9200	MSW SWITCH FAILS CLOSED		
	+ 2 1	C9210	C9220
C9110	UNAUTHORIZED PERSON HOLDS MSW UP		
	* 0 3	CSWCONCZ	CSWMS1CZ
C9120	OPERATOR HOLDS MSW UP		
	* 0 2	CSWMS2CZ	CSWMS2CL
C9210	SPONTANEOUS MSW-1 CLOSURE COMPLETES CIRCUIT		
	* 0 2	CSWMS1CZ	CSWMS1CQ
C9220	MSW FAILS TO REOPEN		
	* 0 2	CSWMSWCP	CSWMSWCZ
CARCRPRD	0 0	1	LIFTING PLATFORM RAISED 1/4 IN
CSWSS3CD	0 0	2	MANY-ROD SHORT OF SELECTOR SWITCH
CSWMS2CQ	0 0	3	MSW-2 CLOSES SPONTANEOUSLY
CSWCONCZ	0 0	4	CONSOLE ACCESS RESTRICTION VIOLATED
CSWMS1CZ	0 0	5	MSW SWITCH HELD UP BY UNAUTHORIZED PERSON
CSWMS1CL	0 0	6	OPERATORS FAIL TO FREE MSW FRM UNAUTHORIZED PERSON
CSWMS2CZ	0 0	7	OPERATOR HOLDS UP MSW UNTIL ALARMS
CSWMS2CL	0 0	8	OPERATORS FAIL TO RELEASE MSW WITHIN 1 MIN AFTER ALARM
CSWMS1CQ	0 0	9	MSW NOT SET TO NEUTRAL POSITION
CSWMSWCP	0 0	10	MSW-1 CLOSES SPONTANEOUSLY
CSWMSWCZ	0 0	11	MSW STAYS UP WHEN RELEASED
	0 0	12	OPERATOR FAILS TO FREE STUCK SWITCH
	0.720E+04		
	0.240E+02		
1 0	0.600E-03	0.500E+01	0.000E+00
2 5	0.180E-07	0.100E+02	0.720E+04
3 1	0.700E-07	0.200E+01	
4 0	0.500E-01	0.500E+01	0.000E+00
5 0	0.210E+00	0.200E+01	0.000E+00
6 0	0.500E-01	0.200E+01	0.000E+00
7 0	0.200E-01	0.840E+01	0.000E+00
8 0	0.100E-02	0.500E+01	0.000E+00
9 3	0.167E-03	0.300E+01	0.180E+02
10 1	0.700E-07	0.200E+01	
11 0	0.140E-05	0.330E+01	0.240E+05
12 0	0.300E+00	0.200E+01	0.000E+00

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Scream Signal Failure under TOP Conditions	
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Scram Signal Failure under TOP Conditions

R00A2 No TOP Signals (neither Automatic nor Manual) to Strings A &
 * 3 0 R0550 R2750 C2600
 R0550 By Wide-Range Nuclear Channels
 * 2 0 R0300 R0400
 R2750 By SOT Channels
 * 2 0 R2500 R2600
 C2600 No Manl SCRAM Signal to both String A and String B
 + 1 1 C0610 RHU002CL
 R0300 No TOP Signal by High Flux Channels to String A
 + 3 0 R0310 R0311 R0312
 R0400 No TOP Signal by High Flux Channels to String B
 + 3 0 R0410 R0411 R0412
 R2500 No SOT Signal to String A
 + 4 0 R2510 R2511 R2512 R2513
 R2600 No SOT Signal to String B
 + 4 0 R2610 R2611 R2612 R2613
 C0610 IMPB and 3MPB Fail to send SCRAM Signal
 * 2 0 C0611 C0631
 R0310 Nucl Channels HFA and HFB Fail to open String A
 * 2 0 R0321 R0322
 R0311 Nucl Channels HFB and HFC Fail to open String A
 * 2 0 R0322 R0323
 R0312 Nucl Channels HFC and HFA Fail to open String A
 * 2 0 R0323 R0321
 R0410 Nucl Channels HFA and HFB Fail to open String B
 * 2 0 R0421 R0422
 R0411 Nucl Channels HFB and HFC Fail to open String B
 * 2 0 R0422 R0423
 R0412 Nucl Channels HFC and HFA Fail to open String B
 * 2 0 R0423 R0421
 R2510 T/C 503D, F, H Combintn Fails in String A
 * 3 0 R251D R251F R251H
 R2511 T/C 503F, H, C Combintn Fails in String A
 * 3 0 R251F R251H R251C
 R2512 T/C 503H, C, D Combintn Fails in String A
 * 3 0 R251H R251C R251D
 R2513 T/C 503C, D, F Combintn Fails in String A
 * 3 0 R251C R251D R251F
 R2610 T/C 503D, F, H Combintn Fails in String B
 * 3 0 R261D R261F R261H
 R2611 T/C 503F, H, C Combintn Fails in String B
 * 3 0 R261F R261H R261C
 R2612 T/C 503H, C, D Combintn Fails in String B
 * 3 0 R261H R261C R261D
 R2613 T/C 503C, D, F Combintn Fails in String B
 * 3 0 R261C R261D R261F
 C0611 IMPB Fails to Open both Strings
 * 2 0 C061A C061B
 C0631 3MPB Fails to Open both Strings
 * 2 0 C063A C063B
 R0321 Nucl Channel A Contacts Fail to Open in String A
 + 1 4 R0326 RRE2A1QP RC22A1QP RRE220QU RC2220QU
 R0322 Nucl Channel B Contacts Fail to Open in String A
 + 1 4 R0327 RRE2B1QP RC22B1QP RRE220QU RC2220QU
 R0323 Nucl Channel C Contacts Fail to Open in String A
 + 1 4 R0328 RRE2C1QP RC22C1QP RRE220QU RC2220QU
 R0421 Nucl Channel A Contacts Fail to Open in String B
 + 1 4 R0426 RRE2A2QP RC22A2QP RRE220QU 220QU

R0422	Nucl Channel B Contacts Fail to Open in String A	
	+ 1 4 R0427 RRE2B2QP RCZ2B2QP RRE2Z0QU RCZ2Z0QU	
R0423	Nucl Channel C Contacts Fail to Open in String B	
	+ 1 4 R0428 RRE2C2QP RCZ2C2QP RRE2Z0QU RCZ2Z0QU	
R251D	503D Contacts Fail to Open in String A	
	+ 1 4 R252D RRE3D1QP RCZ3D1QP RRE2Z0QU RCZ2Z0QU	
R251F	503F Contacts Fail to Open in String A	
	+ 1 4 R252F RRE3F1QP RCZ3F1QP RRE2Z0QU RCZ2Z0QU	
R251H	503H Contacts Fail to Open in String A	
	+ 1 4 R252H RRE3H1QP RCZ3H1QP RRE2Z0QU RCZ2Z0QU	
R251C	503C Contacts Fail to Open in String A	
	+ 1 4 R252C RRE3C1QP RCZ3C1QP RRE2Z0QU RCZ2Z0QU	
R261D	503D Contacts Fail to Open in String B	
	+ 1 4 R252D RRE3D2QP RCZ3D2QP RRE2Z0QU RCZ2Z0QU	
R261F	503F Contacts Fail to Open in String B	
	+ 1 4 R252F RRE3F2QP RCZ3F2QP RRE2Z0QU RCZ2Z0QU	
R261H	503H Contacts Fail to Open in String B	
	+ 1 4 R252H RRE3H2QP RCZ3H2QP RRE2Z0QU RCZ2Z0QU	
R261C	503C Contacts Fail to Open in String B	
	+ 1 4 R252C RRE3C2QP RCZ3C2QP RRE2Z0QU RCZ2Z0QU	
C061A	1MPB Fails to Open String A	
	+ 0 2 CSW100CP CSW10ACP	
C061B	1MPB Fails to Open String B	
	+ 0 2 CSW100CP CSW10BCP	
C063A	3MPB Fails to Open String A	
	+ 0 2 SSW300CP SSW30ACP	
C063B	3MPB Fails to Open String B	
	+ 0 2 SSW300CP SSW30BCP	
R0326	Relays in HF-A Circuit Remain Energized	RHU201QL
	+ 4 1 R0331 R0351 R0371 R0391	
R0327	Relays in HF-B Circuit Remain Energized	RHU201QL
	+ 4 1 R0332 R0352 R0372 R0392	
R0328	Relays in HF-C Circuit Remain Energized	RHU201QL
	+ 4 1 R0333 R0353 R0373 R0393	
R0426	Relays in HF-A Circuit Remain Energized	RHU201QL
	+ 4 1 R0431 R0451 R0371 R0391	
R0427	Relays in HF-B Circuit Remain Energized	RHU201QL
	+ 4 1 R0432 R0452 R0372 R0392	
R0428	Relays in HF-C Circuit Remain Energized	RHU201QL
	+ 4 1 R0433 R0453 R0373 R0393	
R252D	Relays in 503D Circuit Remain Energized	RHU201QL
	+ 1 5 R053D RBI3D1QW RBI3Z0QU RVI3D0QW RVI3Z0QU RRU302QL	
R252F	Relays in 503F Circuit Remain Energized	RHU302QL
	+ 1 5 R053F RBI3F1QW RBI3Z0QU RVI3F0QW RVI3Z0QU RRU302QL	
R252H	Relays in 503H Circuit Remain Energized	RHU302QL
	+ 1 5 R053H RBI3H1QW RBI3Z0QU RVI3H0QW RVI3Z0QU RRU302QL	
R252C	Relays in 503C Circuit Remain Energized	RHU302QL
	+ 1 5 R053C RBI3C1QW RBI3Z0QU RVI3C0QW RVI3Z0QU RRU302QL	
R0331	High Volt Crkt of Channel A Fails to dtct Volt Drop	
	* 2 0 R1331 R2331	
R0351	Compts of HF-A Trip Unit Failure	
	+ 0 6 RBI2A1QW RBI2Z1QU RRE2A3QP RRE2Z9QU RCZ2A3QP RCZ2Z9QU	
R0371	Compts of HF-A Trip Unit Failure	
	+ 0 4 RND2A0RW RND2Z0RU RCA2A0RW RCA2Z0RU	
R0391	Compts of HF-A Trip Unit Failure	
	+ 0 4 RAM2A1QW RAM2Z1QU RDC2A1QW RDC2Z1QU	
R0332	High Volt Crkt of Channel B Fails to dtct Volt Drop	
	* 2 0 R1332 R2332	
R0352	Compts of HF-B Trip Unit Failure	
	+ 0 6 RBI2B1QW RBI2Z1QU RRE2B3QP RRE2Z9QU RCZ2B3QP RCZ2Z9QU	

R0372	Compts of HF-B Trip Unit Failure	+ 0 4 RND2B0RW RND220RU RCA2B0RW RCA220RU	
R0392	Compts of HF-B Trip Unit Failure	+ 0 4 RAM2B1QW RAM221QU RDC2B1QW RDC221QU	
R0333	High Volt Crkt of Channel C Fails to dtct Volt Drop	* 2 0 RI333 R2333	
R0353	Compts of HF-C Trip Unit Failure	+ 0 6 RBY2C1QW RBY221QU RRE2C3QP RRE229QU RCZ2C3QP RCZ229QU	
R0373	Compts of HF-C Trip Unit Failure	+ 0 4 RND2C0RW RND220RU RCA2C0RW RCA220RU	
R0393	Compts of HF-C Trip Unit Failure	+ 0 4 RAM2C1QW RAM221QU RDC2C1QW RDC221QU	
R0431	High Volt Crkt of Channel A Fails to dtct Volt Drop	* 2 0 RI431 R2331	
R0451	Compts of HF-A Trip Unit Failure	+ 0 6 RBY2A2QW RBY221QU RRE2A4QP RRE229QU RCZ2A4QP RCZ229QU	
R0432	High Volt Crkt of Channel B Fails to dtct Volt Drop	* 2 0 RI432 R2332	
R0452	Compts of HF-B Trip Unit Failure	+ 0 6 RBY2B2QW RBY221QU RRE2B4QP RRE229QU RCZ2B4QP RCZ229QU	
R0433	High Volt Crkt of Channel C Fails to dtct Volt Drop	* 2 0 RI433 R2333	
R0453	Compts of HF-C Trip Unit Failure	+ 0 6 RBY2C2QW RBY221QU RRE2C4QP RRE229QU RCZ2C4QP RCZ229QU	
R053D	Failure of T/C or its Ref Junction	+ 0 4 RYC3D0RW RYC300RU RRT3D0AW RRT300AU	
R053F	Failure of T/C or its Ref Junction	+ 0 4 RYC3F0RW RYC300RU RRT3F0AW RRT300AU	
R053H	Failure of T/C or its Ref Junction	+ 0 4 RYC3H0RW RYC300RU RRT3H0AW RRT300AU	
R053C	Failure of T/C or its Ref Junction	+ 0 4 RYC3C0RW RYC300RU RRT3C0AW RRT300AU	
R1331	Compts of LV-A Trip Unit Failure	+ 0 7 RBY2A3QW RBY222QU RRE2A5QP RRE228QU RCZ2A5QP RCZ228QU RHU202QL	
R2331	High Volt DC Power Supply Fails Low	+ 0 2 RDC2A2QW RDC222QU	
R1332	Compts of LV-B Trip Unit Failure	+ 0 7 RBY2B3QW RBY222QU RRE2B5QP RRE228QU RCZ2B5QP RCZ228QU RHU202QL	
R2332	High Volt DC Power Supply Fails Low	+ 0 2 RDC2B2QW RDC222QU	
R1333	Compts of LV-C Trip Unit Failure	+ 0 7 RBY2C3QW RBY222QU RRE2C5QP RRE228QU RCZ2C5QP RCZ228QU RHU202QL	
R2333	High Volt DC Power Supply Fails Low	+ 0 2 RDC2C2QW RDC222QU	
R1431	Compts of LV-A Trip Unit Failure	+ 0 7 RBY2A4QW RBY222QU RRE2A6QP RRE228QU RCZ2A6QP RCZ228QU RHU202QL	
R1432	Compts of LV-B Trip Unit Failure	+ 0 7 RBY2B4QW RBY222QU RRE2B6QP RRE228QU RCZ2B6QP RCZ228QU RHU202QL	
R1433	Compts of LV-C Trip Unit Failure	+ 0 7 RBY2C4QW RBY222QU RRE2C6QP RRE228QU RCZ2C6QP RCZ228QU RHU202QL	
RHU002CL	1 0 Opertr Error - Failure to Push SCRAM Buttons		
RRE2A1QP	0 2 String A Relay Fails to Move to Open Position		
RCZ2A1QP	0 3 Contact 7 or 8 in String A Fails Closed		
RRE2Z0QU	0 0 4 CCF - All Trip Relays of both Strings Fail to Move		
RCZ2Z0QU	0 0 5 CCF - All Trip Contacts of both Strings Fail Closed		
RRE2B1QP	0 2 String A Relay Fails to Move to Open Position		
RCZ2B1QP	0 3 Contact 7 or 8 in String A Fails Closed		
RRE2C1QP	0 2 String A Relay Fails to Move to Open Position		
RCZ2C1QP	0 3 Contact 7 or 8 in String A Fails ed		
RRE2A2QP	0 2 String B Relay Fails to Move to Position		

RC22A2QP	0	3	Contact 7 or 8 in String A Fails	1
RRE2B2QP	0	2	String B Relay Fails to Move to Open Position	
RC22B2QP	0	3	Contact 7 or 8 in String A Fails Closed	
RRE2C2QP	0	2	String B Relay Fails to Move to Open Position	
RC22C2QP	0	3	Contact 7 or 8 in String A Fails Closed	
RRE3D1QP	0	2	String A Relay Fails to Change Position	
RC23D1QP	0	3	Contact 7 or 8 in String A Fails Closed	
RRE3F1QP	0	2	String A Relay Fails to Change Position	
RC23F1QP	0	3	Contact 7 or 8 in String A Fails Closed	
RRE3H1QP	0	2	String A Relay Fails to Change Position	
RC23H1QP	0	3	Contact 7 or 8 in String A Fails Closed	
RRE3C1QP	0	2	String A Relay Fails to Change Position	
RC23C1QP	0	3	Contact 7 or 8 in String A Fails Closed	
RRE3D2QP	0	2	String B Relay Fails to Change Position	
RC23D2QP	0	3	Contact 7 or 8 in String B Fails Closed	
RRE3F2QP	0	2	String B Relay Fails to Change Position	
RC23F2QP	0	3	Contact 7 or 8 in String B Fails Closed	
RRE3H2QP	0	2	String B Relay Fails to Change Position	
RC23H2QP	0	3	Contact 7 or 8 in String B Fails Closed	
RRE3C2QP	0	2	String B Relay Fails to Change Position	
RC23C2QP	0	3	Contact 7 or 8 in String B Fails Closed	
CSW100CP	0	16	6 MPB Mech Parts Fail	
CSW10ACP	0	15	7 MPB Switch of String A Fails to Open	
CSW10BCP	0	15	7 MPB Switch of String B Fails to Open	
SSW300CP	0	16	6 MPB Mech Parts Fail	
SSW30ACP	0	15	7 MPB Switch of String A Fails to Open	
SSW30BCP	0	15	7 MPB Switch of String B Fails to Open	
RHU201QL	1	0	8 CCF Human Error - Failure to Set HF Trip Points Correctly	
RBI3D1LQ	0	19	9 SOT Trip Unit of Strings A & B Fails	
RBI3D0Q	0	0	10 CCF - All Trip Units in All SOT Channels Fail	
RV13D0Q	0	18	11 MV/I Converter Fails	
RV13Z0Q	0	0	12 CCF - All MV/I Converters in SOT Channels Fail	
RHU302QL	1	0	13 CCF - Human Error - SOT Trip Points Set Too High	
RBI3F1LQ	0	19	9 SOT Trip Unit of Strings A & B Fails	
RV13F0Q	0	18	11 MV/I Converter Fails	
RBI3H1LQ	0	19	9 SOT Trip Unit of Strings A & B Fails	
RV13H0Q	0	18	11 MV/I Converter Fails	
RBI3C1LQ	0	19	9 SOT Trip Unit of Strings A & B Fails	
RV13C0Q	0	18	11 MV/I Converter Fails	
RBI2A1LQ	0	5	14 Bistable of String A Fails to detect High Flux	
RBI221LQ	0	0	15 CCF - All HF Bistables of All Channels Fail	
RRE2A3QP	0	9	16 K3 Relay Fails to Move to Open Position	
RRE229Q	0	0	17 CCF - HF K-Relays of All Channels Fail to Move	
RC22A3QP	0	12	18 Contacts of K3 Relay Fail Closed	
RRE229Q	0	0	19 CCF - All Contacts of HF K-Relays Fail Closed	
RND2A0RW	0	7	20 Graded Fission Chamber in J1 Thimble Fails	
RND220R	0	0	21 All Graded Fission Chambers Fail	
RCA2A0RW	0	6	22 Cable and Connector of Channel A Fail	
RCA220R	0	0	23 CCF - All Cables and Connectors Fail	
RAM2A1LQ	0	11	24 Current Amplifier Reduces Signal to Trip unit	
RAM221LQ	0	0	25 CCF - All Current Amplifier Reduce Signals to Trip units	
RDC2A1LQ	0	14	26 +15 V DC Power Supply Fails Low	
RDC221LQ	0	0	27 ALL +15 V DC Power Supplies Fail Low	
RBI2B1LQ	0	5	14 Bistable of String A Fails to detect High Flux	
RRE2B3QP	0	9	16 K3 Relay Fails to Move to Open Position	
RC22B3QP	0	12	18 Contacts of K3 Relay Fail Closed	
RND2B0RW	0	7	20 Graded Fission Chamber in J3 Thimble Fails	
RCA2B0RW	0	6	22 Cable and Connector of Channel B Fail	
RAM2B1LQ	0	11	24 Current Amplifier Reduces Signal to Trip unit	
RDC2B1LQ	0	14	26 +15 V DC Power Supply Fails Low	

RBI2C1QW 0 5	14 Bistable of String A Fails to dtct High Flux
RRE2C3QP 0 9	16 K3 Relay Fails to Move to Open Position
RC22C3QP 0 12	18 Contacts of K3 Relay Fail Closed
RND2C0RW 0 7	20 Grded Fission Chamber in J4 Thimble Fails
RC22C0RW 0 6	22 Cable and Connector of Channel C Fail
RAM2C1QW 0 11	24 Current Amplfr Reduces Signal to Trip unit
RDC2C1QW 0 14	26 +15 V DC Power Supply Fails Low
RBI2A2QW 0 5	14 Bistable of String B Fails to dtct High Flux
RRE2A4QP 0 9	16 K4 Relay Fails to Move to Open Position
RC22A4QP 0 12	18 Contacts of K4 Relay Fail Closed
RBI2B2QW 0 5	14 Bistable of String B Fails to dtct High Flux
RRE2B4QP 0 9	16 K4 Relay Fails to Move to Open Position
RC22B4QP 0 12	18 Contacts of K4 Relay Fail Closed
RBI2C2QW 0 5	14 Bistable of String B Fails to dtct High Flux
RRE2C4QP 0 9	16 K4 Relay Fails to Move to Open Position
RC22C4QP 0 12	18 Contacts of K4 Relay Fail Closed
RTC3D0RW 0 20	28 T/C 503D Fails Low - Short/Open
RTC300RU 0 0	29 CCF - All T/C's Fail Low
RRT3D0AW 0 21	30 T/C 503D Ref Junction Fails High
RRT300AU 0 0	31 CCF - All Ref Junctions Fail High
RTC3F0RW 0 20	28 T/C 503F Fails Low - Short/Open
RRT3F0AW 0 21	30 T/C 503F Ref Junction Fails High
RTC3H0RW 0 20	28 T/C 503H Fails Low - Short/Open
RRT3H0AW 0 21	30 T/C 503H Ref Junction Fails High
RTC3C0RW 0 20	28 T/C 503C Fails Low - Short/Open
RRT3C0AW 0 21	30 T/C 503C Ref Junction Fails High
RBI2A3QW 0 5	14 Bistable of String A Fails to dtct Low Voltage
RBI2Z22QU 0 0	32 CCF - All LV Bistables of All Channels Fail
RRE2A5QP 0 9	16 K5 Relay Fails to Move to Open Position
RRE2Z28QU 0 0	33 CCF - LV K-Relays of All Channels Fail to Move
RC22A5QP 0 12	18 Contacts of K5 Relay Fail Closed
RC22Z28QU 0 0	34 CCF - All Contacts of LV K-Relays Fail Closed
RHU202QL 1 0	35 CCF Human Error - Failure to Set LV Trip Points Correctly
RDC2A2QW 0 14	26 High Volt DC Power Supply Fails Low
RDC2Z22QU 0 0	36 CCF - All High Volt DC Power Supplies Fail Low
RBI2B3QW 0 5	14 Bistable of String A Fails to dtct Low Voltage
RRE2B5QP 0 9	16 K5 Relay Fails to Move to Open Position
RC22B5QP 0 12	18 Contacts of K5 Relay Fail Closed
RDC2B2QW 0 14	26 High Volt DC Power Supply Fails Low
RBI2C3QW 0 5	14 Bistable of String A Fails to dtct Low Voltage
RRE2C5QP 0 9	16 K5 Relay Fails to Move to Open Position
RC22C5QP 0 12	18 Contacts of K5 Relay Fail Closed
RDC2C2QW 0 14	26 High Volt DC Power Supply Fails Low
RBI2A4QW 0 5	14 Bistable of String B Fails to dtct Low Voltage
RRE2A6QP 0 9	16 K6 Relay Fails to Move to Open Position
RC22A6QP 0 12	18 Contacts of K6 Relay Fail Closed
RBI2B4QW 0 5	14 Bistable of String B Fails to dtct Low Voltage
RRE2B6QP 0 9	16 K6 Relay Fails to Move to Open Position
RC22B6QP 0 12	18 Contacts of K6 Relay Fail Closed
RBI2C4QW 0 5	14 Bistable of String B Fails to dtct Low Voltage
RRE2C6QP 0 9	16 K6 Relay Fails to Move to Open Position
RC22C6QP 0 12	18 Contacts of K6 Relay Fail Closed
0.240E+02	
0.240E+04	
1 0 0.100E+00	0.900E+01 0.000E+00
2 5 0.200E-06	0.100E+02 0.240E+04
3 0 0.300E-03	0.100E+02 0.000E+00
4 5 0.160E-08	0.100E+02 0.240E+04
5 0 0.240E-05	0.100E+02 0.000E+00
6 0 0.200E-05	0.100E+02 0.000E+00

8 0	0.150E-05	0.100E+02	0.000E+00
9 5	0.500E-06	0.100E+02	0.240E+04
10 5	0.100E-07	0.100E+02	0.240E+04
11 5	0.100E-05	0.100E+02	0.240E+04
12 5	0.200E-07	0.100E+02	0.240E+04
13 0	0.250E-01	0.100E+02	0.000E+00
14 5	0.500E-06	0.100E+02	0.240E+04
15 5	0.400E-08	0.100E+02	0.240E+04
16 5	0.200E-06	0.100E+02	0.240E+04
17 5	0.160E-08	0.100E+02	0.240E+04
18 0	0.300E-03	0.100E+02	0.000E+00
19 0	0.240E-05	0.100E+02	0.000E+00
20 5	0.300E-05	0.100E+02	0.240E+04
21 5	0.300E-06	0.100E+02	0.240E+04
22 5	0.400E-06	0.100E+02	0.240E+04
23 5	0.400E-07	0.100E+02	0.240E+04
24 5	0.100E-05	0.100E+02	0.240E+04
25 5	0.100E-06	0.100E+02	0.240E+04
26 5	0.300E-05	0.100E+02	0.240E+04
27 5	0.300E-06	0.100E+02	0.240E+04
28 5	0.100E-05	0.100E+02	0.240E+04
29 5	0.200E-07	0.100E+02	0.240E+04
30 5	0.200E-05	0.100E+02	0.240E+04
31 5	0.400E-07	0.100E+02	0.240E+04
32 5	0.400E-08	0.100E+02	0.240E+04
33 5	0.160E-08	0.100E+02	0.240E+04
34 0	0.240E-05	0.100E+02	0.000E+00
35 0	0.150E-05	0.100E+02	0.000E+00
36 5	0.300E-06	0.100E+02	0.240E+04

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Rod Drop Failure under TOP Conditions
 R00C2 Elect & Mech Failures of CRs & SRs
 * 2 0 C2500 S2010
 C2500 Mech or Elect Components Failure, but Signal is OK
 * 2 0 C3000 C5000
 S2010 Safety Rods Failure to SCRAM
 + 2 0 S2600 S0500
 C3000 Control Rods 3 to 6 affected by CPB-1 Contacts
 + 2 0 C3310 C3360
 C5000 Control Rods 7, 9 to 12 affected by CPB-2 Contacts
 + 2 0 C3310 C3760
 S2600 No Manl SCRAM Signal to both String A & String B
 + 2 0 S0631 S2637
 S0500 Mech or Elect Components Failure, but Signal is OK
 + 2 1 S0510 S0560 SCL000RP
 C3310 Mechanical Components Failure
 + 1 1 C3315 CCL00PRU
 C3360 Rod Release Contacts Fail to Open
 * 4 0 C3365 C3375 C3385 C3395
 C3760 Rod Release Contacts Fail to Open
 * 4 0 C3365 C3375 C3785 C3395
 S0631 3MPB Fails to Open both Strings
 * 2 0 S063A S063B
 S2637 Operator Failure to SCRAM SRs
 + 2 0 S0638 S2639
 S0510 Mechanical Components Failure
 + 1 1 S0515 SCD200RW
 S0560 Rods Release Contacts Fail to Open the 24 VDC Ckt
 * 2 0 S0570 S0580
 C3315 Force Larger than CR Wght is Required
 + 2 0 C3320 C3340
 C3365 Contacts CPC-1 Fail to Open
 + 0 4 RCZCCLQP RRECCLQP RCZC00QU RREC00QU
 C3375 Contacts CPA Fail to Open
 + 0 4 RCZCPAPQ RRECFAQP RCZC00QU RREC00QU
 C3385 Contacts CPB-1 Fail to Open
 + 0 4 RCZCB1QP RRECBIQP RCZC00QU RREC00QU
 C3395 Contacts CPY Fail to Open
 + 0 6 RCZCPYQP RRECPYQP RCZCP2QP RREC2QP RCZC00QU RREC00QU
 C3785 Contacts CPB-2 Fail to Open
 + 0 4 RCZCB2QP RREC2QP RCZC00QU RREC00QU
 S063A 3MPB Fails to Open String A
 + 0 2 SSW300CP SSW30ACP
 S063B 3MPB Fails to Open String B
 + 0 2 SSW300CP SSW30BCP
 S0638 With Successful Automatic SCRAM Signal
 + 0 1 RHU00ZCL
 S2639 With Failed Automatic SCRAM Signal
 * 1 1 R0540 RHU00ZCL
 S0515 Force Larger than SRs Wght & Spring Force is Required
 + 2 0 S0520 S0540
 S0570 Contacts SRO Fail to Open
 + 0 4 RRESROQP RCZSROQP RREC00QU RCZC00QU
 S0580 Contacts SRO-X Fail to Open
 + 0 4 RRESRXQP RCZSRXQP RREC00QU RCZC00QU
 C3320 Forces from CR Drive
 + 0 3 CCD10GRU CCD20WRU CCD30IRU
 C3340 Forces from CR and its Thimble
 + 0 2 CSR10GRU CSR30IRU

R0540 No TOP Signal to both String A and String F
 * 2 0 R0550 R2750
 S0520 Forces from SR Drive
 + 0 2 SCD100RG SCD300RI
 S0540 Forces from SRs and their Thimble
 + 0 4 SRF100RG CSR10GRU SSR300RI CSR30IRU
 R0550 By Wide-Range Nuclear Channels
 * 2 0 R0300 R0400
 R2750 By SOT Channels
 * 2 0 R2500 R2600
 R0300 No TOP Signal by High Flux Channels to String A
 + 3 0 R0310 R0311 R0312
 R0400 No TOP Signal by High Flux Channels to String B
 + 3 0 R0410 R0411 R0412
 R2500 No SOT Signal to String A
 + 4 0 R2510 R2511 R2512 R2513
 R2600 No SOT Signal to String B
 + 4 0 R2610 R2611 R2612 R2613
 R0310 Nucl Channels HFA and HFB Fail to open String A
 * 2 0 R0321 R0322
 R0311 Nucl Channels HFB and HFC Fail to open String A
 * 2 0 R0322 R0323
 R0312 Nucl Channels HFC and HFA Fail to open String A
 * 2 0 R0323 R0321
 R0410 Nucl Channels HFA and HEB Fail to open String B
 * 2 0 R0421 R0422
 R0411 Nucl Channels HFB and HFC Fail to open String B
 * 2 0 R0422 R0423
 R0412 Nucl Channels HFC and HFA Fail to open String B
 * 2 0 R0423 R0421
 R2510 T/C 503D, F, H Combintn Fails in String A
 * 3 0 R251D R251F R251H
 R2511 T/C 503F, H, C Combintn Fails in String A
 * 3 0 R251F R251H R251C
 R2512 T/C 503H, C, D Combintn Fails in String A
 * 3 0 R251H R251C R251D
 R2513 T/C 503C, D, F Combintn Fails in String A
 * 3 0 R251C R251D R251F
 R2610 T/C 503D, F, H Combintn Fails in String B
 * 3 0 R261D R261F R261H
 R2611 T/C 503F, H, C Combintn Fails in String B
 * 3 0 R261F R261H R261C
 R2612 T/C 503H, C, D Combintn Fails in String B
 * 3 0 R261H R261C R261D
 R2613 T/C 503C, D, F Combintn Fails in String B
 * 3 0 R261C R261D R261F
 R0321 Nucl Channel A Contacts Fail to Open in String A
 + 1 4 R0326 RRE2A1QP RCZ2A1QP RRE2Z0QU RCZ2Z0QU
 R0322 Nucl Channel B Contacts Fail to Open in String A
 + 1 4 R0327 RRE2B1QP RCZ2B1QP RRE2Z0QU RCZ2Z0QU
 R0323 Nucl Channel C Contacts Fail to Open in String A
 + 1 4 R0328 RRE2C1QP RCZ2C1QP RRE2Z0QU RCZ2Z0QU
 R0421 Nucl Channel A Contacts Fail to Open in String B
 + 1 4 R0426 RRE2A2QP RCZ2A2QP RRE2Z0QU RCZ2Z0QU
 R0422 Nucl Channel B Contacts Fail to Open in String B
 + 1 4 R0427 RRE2B2QP RCZ2B2QP RRE2Z0QU RCZ2Z0QU
 R0423 Nucl Channel C Contacts Fail to Open in String B
 + 1 4 R0428 RRE2C2QP RCZ2C2QP RRE2Z0QU RCZ2Z0QU
 R251D 503D Contacts Fail to Open in String A
 + 1 4 R252D RRE3D1QP RCZ3D1QP RRE2Z0QU RCZ2Z0QU

R251F 503F Contacts Fail to Open in String A
 + 1 4 R252F RRE3F1QP RCZ3F1QP RRE220QU RCZ220QU
 R251H 503H Contacts Fail to Open in String A
 + 1 4 R252H RRE3H1QP RCZ3H1QP RRE220QU RCZ220QU
 R251C 503C Contacts Fail to Open in String A
 + 1 4 R252C RRE3C1QP RCZ3C1QP RRE220QU RCZ220QU
 R261D 503D Contacts Fail to Open in String B
 + 1 4 R252D RRE3D2QP RCZ3D2QP RRE220QU RCZ220QU
 R261F 503F Contacts Fail to Open in String B
 + 1 4 R252F RRE3F2QP RCZ3F2QP RRE220QU RCZ220QU
 R261H 503H Contacts Fail to Open in String B
 + 1 4 R252H RRE3H2QP RCZ3H2QP RRE220QU RCZ220QU
 R261C 503C Contacts Fail to Open in String B
 + 1 4 R252C RRE3C2QP RCZ3C2QP RRE220QU RCZ220QU
 R0326 Relays in HF-A Circuit Remain Energized
 + 4 1 R0331 R0351 R0371 R0391 RHU201QL
 R0327 Relays in HF-B Circuit Remain Energized
 + 4 1 R0332 R0352 R0372 R0392 RHU201QL
 R0328 Relays in HF-C Circuit Remain Energized
 + 4 1 R0333 R0353 R0373 R0393 RHU201QL
 R0426 Relays in HF-A Circuit Remain Energized
 + 4 1 R0431 R0451 R0371 R0391 RHU201QL
 R0427 Relays in HF-B Circuit Remain Energized
 + 4 1 R0432 R0452 R0372 R0392 RHU201QL
 R0428 Relays in HF-C Circuit Remain Energized
 + 4 1 R0433 R0453 R0373 R0393 RHU201QL
 R252D Relays in 503D Circuit Remain Energized
 + 1 5 R053D RBI3D1QW RBI320QU RVI3D0QW RVI320QU RHU302QL
 R252F Relays in 503F Circuit Remain Energized
 + 1 5 R053F RBI3F1QW RBI320QU RVI3F0QW RVI320QU RHU302QL
 R252H Relays in 503H Circuit Remain Energized
 + 1 5 R053H RBI3H1QW RBI320QU RVI3H0QW RVI320QU RHU302QL
 R252C Relays in 503C Circuit Remain Energized
 + 1 5 R053C RBI3C1QW RBI320QU RVI3C0QW RVI320QU RHU302QL
 R0331 High Volt Crkt of Channel A Fails to dtct Volt Drop
 * 2 0 R1331 R2331
 R0351 Compts of HF-A Trip Unit Failure
 + 0 6 RBI2A1QW RBI221QU RRE2A3QP RRE229QU RCZ2A3QP RCZ229QU
 R0371 Compts of HF-A Trip Unit Failure
 + 0 4 RND2A0RW RND220RU RCA2A0RW RCA220RU
 R0391 Compts of HF-A Trip Unit Failure
 + 0 4 RAM2A1QW RAM221QU RDC2A1QW RDC221QU
 R0332 High Volt Crkt of Channel B Fails to dtct Volt Drop
 * 2 0 R1332 R2332
 R0352 Compts of HF-B Trip Unit Failure
 + 0 6 RBI2B1QW RBI221QU RRE2B3QP RRE229QU RCZ2B3QP RCZ229QU
 R0372 Compts of HF-B Trip Unit Failure
 + 0 4 RND2B0RW RND220RU RCA2B0RW RCA220RU
 R0392 Compts of HF-B Trip Unit Failure
 + 0 4 RAM2B1QW RAM221QU RDC2B1QW RDC221QU
 R0333 High Volt Crkt of Channel C Fails to dtct Volt Drop
 * 2 0 R1333 R2333
 R0353 Compts of HF-C Trip Unit Failure
 + 0 6 RBI2C1QW RBI221QU RRE2C3QP RRE229QU RCZ2C3QP RCZ229QU
 R0373 Compts of HF-C Trip Unit Failure
 + 0 4 RND2C0RW RND220RU RCA2C0RW RCA220RU
 R0393 Compts of HF-C Trip Unit Failure
 + 0 4 RAM2C1QW RAM221QU RDC2C1QW RDC221QU
 R0431 High Volt Crkt of Channel A Fails to dtct Volt Drop
 * 2 0 R1431 R2331

R0451	Compts of HF-A Trip Unit Failure	
	+ 0 6 RBI2A2QW RBI221Q RRE2A4QP RRE229QU RRA4QP RCZ229QU	
R0432	High Volt Crkt of Channel B Fails to dtct Volt Drop	
	* 2 0 R1432 R2332	
R0452	Compts of HF-B Trip Unit Failure	
	+ 0 6 RBI2B2QW RBI221Q RRE2B4QP RRE229QU RCZ2B4QP RCZ229QU	
R0433	High Volt Crkt of Channel C Fails to dtct Volt Drop	
	* 2 0 R1433 R2333	
R0453	Compts of HF-C Trip Unit Failure	
	+ 0 6 RBI2C2QW RBI221Q RRE2C4QP RRE229QU RCZ2C4QP RCZ229QU	
R053D	Failure of T/C or its Ref Junction	
	+ 0 4 RTC3D0RW RTC300RU RRT3D0AW RRT300AU	
R053F	Failure of T/C or its Ref Junction	
	+ 0 4 RTC3F0RW RTC300RU RRT3F0AW RRT300AU	
R053H	Failure of T/C or its Ref Junction	
	+ 0 4 RTC3H0RW RTC300RU RRT3H0AW RRT300AU	
R053C	Failure of T/C or its Ref Junction	
	+ 0 4 RTC3C0RW RTC300RU RRT3C0AW RRT300AU	
R1331	Compts of LV-A Trip Unit Failure	
	+ 0 7 RBI2A3QW RBI222Q RRE2A5QP RRE228QU RCZ2A5QP RCZ228QU RHU202QL	
R2331	High Volt DC Power Supply Fails Low	
	+ 0 2 RDC2A2QW RDC222QU	
R1332	Compts of LV-B Trip Unit Failure	
	+ 0 7 RBI2B3QW RBI222Q RRE2B5QP RRE228QU RCZ2B5QP RCZ228QU RHU202QL	
R2332	High Volt DC Power Supply Fails Low	
	+ 0 2 RDC2B2QW RDC222QU	
R1333	Compts of LV-C Trip Unit Failure	
	+ 0 7 RBI2C3QW RBI222Q RRE2C5QP RRE228QU RCZ2C5QP RCZ228QU RHU202QL	
R2333	High Volt DC Power Supply Fails Low	
	+ 0 2 RDC2C2QW RDC222QU	
R1431	Compts of LV-A Trip Unit Failure	
	+ 0 7 RBI2A4QW RBI222Q RRE2A6QP RRE228QU RCZ2A6QP RCZ228QU RHU202QL	
R1432	Compts of LV-B Trip Unit Failure	
	+ 0 7 RBI2B4QW RBI222Q RRE2B6QP RRE228QU RCZ2B6QP RCZ228QU RHU202QL	
R1433	Compts of LV-C Trip Unit Failure	
	+ 0 7 RBI2C4QW RBI222Q RRE2C6QP RRE228QU RCZ2C6QP RCZ228QU RHU202QL	
SCLO00RP 0 0	1 Magnetic Clutch Fails to Release SRs	
CCLO00PRU 0 0	2 CCF - Magnetic Clutches Fail to Release CRs	
SCD200RW 0 0	3 Failure of Rlse Mechanism, Connecting Shafts, Bearings	
RCZCC1QP 0 24	4 Contacts Fail Closed	
RRECC1QP 0 17	5 CPC-1 Relay Fails to Change Position	
RCZC00QU 0 0	6 CCF - Contacts of All CP-Relays Fail Closed	
RREC00QU 0 0	7 CCF - All CP-Relays on Strings A & B Fail	
RCZCPAQP 0 24	4 Contacts Fail Closed	
RRECPAQP 0 17	5 CPA Relay Fails to Change Position	
RCZCH1QP 0 24	4 Contacts Fail Closed	
RRECB1QP 0 17	5 CPB-1 Relay Fails to Change Position	
RCZCPYQP 0 24	4 Contacts Fail Closed	
RRECPLYQP 0 17	5 CPY Relay Fails to Change Position	
RCZCP2QP 0 24	4 Contacts Fail Closed	
RREC2QP 0 17	5 CP-2 Relay Fails to Change Position	
RCZCB2QP 0 24	4 Contacts Fail Closed	
RRECB2QP 0 17	5 CPB-2 Relay Fails to Change Position	
SSW300CP 0 16	8 3MPB Mech Parts Fail	
SSW30ACP 0 15	9 3MPB Switch of String A Fails to Open	
SSW30BCP 0 15	9 3MPB Switch of String B Fails to Open	
RHU00ZCL 1 0	10 Opertr Error - Failure to Push 3MPB SCRAM Button	
RHU002CL 1 0	11 Opertr Error - Failure to Push SCRAM Buttons	
RRESROQP 0 17	5 SRO Relay Fails to Change Position	
RCZSROQP 0 3	12 Contacts Fail Closed	

RRESRXQP 0 17 5 SKO-X Relay Fails to Change Position
 RCZSRXQP 0 3 12 Contacts Fail Closed
 CCD10GRU 0 0 13 CCF - Bowing and Binding / Inside Tank
 CCD20WUR 0 0 14 CCF - Bowing and Binding/Above Rotating Plug
 CCD30IRU 0 0 15 CCF - Forces due to Foreign Materials in Sodium
 CSR10GRU 0 0 16 CCF - Forces due to Bowing and Binding
 CSR30IRU 0 0 17 CCF - Forces due to Foreign Materials in Sodium
 SCD10ORG 0 0 18 Forces due to Bowing and Binding
 SCD30ORI 0 0 19 Forces due to Foreign Materials in Sodium
 SSR10ORG 0 0 20 Forces due to Bowing and Binding
 SSR30ORI 0 0 21 Forces due to Foreign Materials in Sodium
 RRE2A1QP 0 2 22 String A Relay Fails to Move to Open Position
 RCZ2A1QP 0 3 12 Contact 7 or 8 in String A Fails Closed
 RRE2Z0QU 0 0 23 CCF - All Trip Relays of both Strings Fail to Move
 RCZ2Z0QU 0 0 24 CCF - All Trip Contacts of both Strings Fail Closed
 RRE2B1QP 0 2 22 String A Relay Fails to Move to Open Position
 RCZ2B1QP 0 3 12 Contact 7 or 8 in String A Fails Closed
 RRE2C1QP 0 2 22 String A Relay Fails to Move to Open Position
 RCZ2C1QP 0 3 12 Contact 7 or 8 in String A Fails Closed
 RRE2A2QP 0 2 22 String B Relay Fails to Move to Open Position
 RCZ2A2QP 0 3 12 Contact 7 or 8 in String A Fails Closed
 RRE2B2QP 0 2 22 String B Relay Fails to Move to Open Position
 RCZ2B2QP 0 3 12 Contact 7 or 8 in String A Fails Closed
 RRE2C2QP 0 2 22 String B Relay Fails to Move to Open Position
 RCZ2C2QP 0 3 12 Contact 7 or 8 in String A Fails Closed
 RRE3D1QP 0 2 22 String A Relay Fails to Change Position
 RCZ3D1QP 0 3 12 Contact 7 or 8 in String A Fails Closed
 RRE3F1QP 0 2 22 String A Relay Fails to Change Position
 RCZ3F1QP 0 3 12 Contact 7 or 8 in String A Fails Closed
 RRE3H1QP 0 2 22 String A Relay Fails to Change Position
 RCZ3H1QP 0 3 12 Contact 7 or 8 in String A Fails Closed
 RRE3C1QP 0 2 22 String A Relay Fails to Change Position
 RCZ3C1QP 0 3 12 Contact 7 or 8 in String A Fails Closed
 RRE3D2QP 0 2 22 String B Relay Fails to Change Position
 RCZ3D2QP 0 3 12 Contact 7 or 8 in String B Fails Closed
 RRE3F2QP 0 2 22 String B Relay Fails to Change Position
 RCZ3F2QP 0 3 12 Contact 7 or 8 in String B Fails Closed
 RRE3H2QP 0 2 22 String B Relay Fails to Change Position
 RCZ3H2QP 0 3 12 Contact 7 or 8 in String B Fails Closed
 RRE3C2QP 0 2 22 String B Relay Fails to Change Position
 RCZ3C2QP 0 3 12 Contact 7 or 8 in String B Fails Closed
 RHU201QL 1 0 25 CCF Human Error - Failure to Set HF Trip Points Correctly
 RB13D1QW 0 19 26 SOT Trip Unit of Strings A & B Fails
 RB13Z0QU 0 0 27 CCF - All Trip Units in All SOT Channels Fail
 RV13D0QW 0 18 28 MV/I Converter Fails
 RV13Z0QU 0 0 29 CCF - All MV/I Converters in SOT Channels Fail
 RHU302QL 1 0 30 CCF - Human Error - SOT Trip Points Set Too High
 RB13F1QW 0 19 26 SOT Trip Unit of Strings A & B Fails
 RV13F0QW 0 18 28 MV/I Converter Fails
 RB13H1QW 0 19 26 SOT Trip Unit of Strings A & B Fails
 RV13H0QW 0 18 28 MV/I Converter Fails
 RB13C1QW 0 19 26 SOT Trip Unit of Strings A & B Fails
 RV13C0QW 0 18 28 MV/I Converter Fails
 RB12A1QW 0 5 31 Bistable of String A Fails to detect High Flux
 RB12Z1QU 0 0 32 CCF - All HF Bistables of All Channels Fail
 RRE2A3QP 0 9 33 K3 Relay Fails to Move to Open Position
 RRE2Z9QU 0 0 34 CCF - HF K-Relays of All Channels Fail to Move
 RCZ2A3QP 0 12 35 Contacts of K3 Relay Fail Closed
 RCZ2Z9QU 0 0 36 CCF - All Contacts of HF K-Relays Fail Closed
 RND2A0RW 0 7 37 Graded Fission Chamber in J1 Thim Fails

RND2Z0RU	0	0	38	All Graded Fission Chambers Fail
RCA2A0RW	0	6	39	Cable and Connector of Channel A Fail
RCA2Z0RU	0	0	40	CCF - All Cables and Connectors Fail
RAM2A1QW	0	11	41	Current Amplr Reduces Signal to Trip unit
RAM2Z1QU	0	0	42	CCF - All Current Amplr Reduce Signals to Trip units
RDC2A1QW	0	14	43	+15 V DC Power Supply Fails Low
RDC2Z1QU	0	0	44	All +15 V DC Power Supplies Fail Low
RBI2B1QW	0	5	31	Bistable of String A Fails to dtct High Flux
RRE2B3QP	0	9	33	K3 Relay Fails to Move to Open Position
RCZ2B3QP	0	12	35	Contacts of K3 Relay Fail Closed
RND2B0RW	0	7	37	Graded Fission Chamber in J3 Thimble Fails
RCA2B0RW	0	6	39	Cable and Connector of Channel B Fail
RAM2B1QW	0	11	41	Current Amplr Reduces Signal to Trip unit
RDC2B1QW	0	14	43	+15 V DC Power Supply Fails Low
RBI2C1QW	0	5	31	Bistable of String A Fails to dtct High Flux
RRE2C3QP	0	9	33	K3 Relay Fails to Move to Open Position
RCZ2C3QP	0	12	35	Contacts of K3 Relay Fail Closed
RND2C0RW	0	7	37	Graded Fission Chamber in J4 Thimble Fails
RCA2C0RW	0	6	39	Cable and Connector of Channel C Fail
RAM2C1QW	0	11	41	Current Amplr Reduces Signal to Trip unit
RDC2C1QW	0	14	43	+15 V DC Power Supply Fails Low
RBI2A2QW	0	5	31	Bistable of String B Fails to dtct High Flux
RRE2A4QP	0	9	33	K4 Relay Fails to Move to Open Position
RCZ2A4QP	0	12	35	Contacts of K4 Relay Fail Closed
RBI2B2QW	0	5	31	Bistable of String B Fails to dtct High Flux
RRE2B4QP	0	9	33	K4 Relay Fails to Move to Open Position
RCZ2B4QP	0	12	35	Contacts of K4 Relay Fail Closed
RBI2C2QW	0	5	31	Bistable of String B Fails to dtct High Flux
RRE2C4QP	0	9	33	K4 Relay Fails to Move to Open Position
RCZ2C4QP	0	12	35	Contacts of K4 Relay Fail Closed
RTC3D0RW	0	20	45	T/C 503D Fails Low - Short/Open
RTC300RU	0	0	46	CCF - All T/C's Fail Low
RRT3D0AW	0	21	47	T/C 503D Ref Junction Fails High
RRT300AU	0	0	48	CCF - All Ref Junctions Fail High
RTC3F0RW	0	20	45	T/C 503F Fails Low - Short/Open
RRT3F0AW	0	21	47	T/C 503F Ref Junction Fails High
RTC3H0RW	0	20	45	T/C 503H Fails Low - Short/Open
RRT3H0AW	0	21	47	T/C 503H Ref Junction Fails High
RTC3C0RW	0	20	45	T/C 503C Fails Low - Short/Open
RRT3C0AW	0	21	47	T/C 503C Ref Junction Fails High
RBI2A3QW	0	5	31	Bistable of String A Fails to dtct Low Voltage
RBI2Z2QU	0	0	49	CCF - All LV Bistables of All Channels Fail
RRE2A5QP	0	9	33	K5 Relay Fails to Move to Open Position
RCZ2Z2QU	0	0	50	CCF - LV K-Relays of All Channels Fail to Move
RRE2Z2QU	0	12	35	Contacts of K5 Relay Fail Closed
RCZ2Z2QU	0	0	51	CCF - All Contacts of LV K-Relays Fail Closed
RHU202QL	1	0	52	CCF Human Error - Failure to Set LV Trip Points Correctly
RDC2A2QW	0	14	43	High Volt DC Power Supply Fails Low
RBI2B3QW	0	5	31	Bistable of String A Fails to dtct Low Voltage
RRE2B5QP	0	9	33	K5 Relay Fails to Move to Open Position
RCZ2B5QP	0	12	35	Contacts of K5 Relay Fail Closed
RDC2B2QW	0	14	43	High Volt DC Power Supply Fails Low
RBI2C3QW	0	5	31	Bistable of String A Fails to dtct Low Voltage
RRE2C5QP	0	9	33	K5 Relay Fails to Move to Open Position
RCZ2C5QP	0	12	35	Contacts of K5 Relay Fail Closed
RDC2C2QW	0	14	43	High Volt DC Power Supply Fails Low
RBI2A4QW	0	5	31	Bistable of String B Fails to dtct Low Voltage
RRE2A6QP	0	9	33	K6 Relay Fails to Move to Open Position
RCZ2A6QP	0	12	35	Contacts of K6 Relay Fail Closed

RB12B4QW 0 5 31 Bistable of String B Fails to dtct Low Voltage
 RRE2B6QP 0 9 33 K6 Relay Fails to Move to Open Position
 RC22B6QP 0 12 35 Contacts of K6 Relay Fail Closed
 RB12C4QW 0 5 31 Bistable of String B Fails to dtct Low Voltage
 RRE2C6QP 0 9 33 K6 Relay Fails to Move to Open Position
 RC22C6QP 0 12 35 Contacts of K6 Relay Fail Closed
 0.240E+02
 0.240E+04

1 0	0.200E-03	0.200E+02	0.000E+00
2 0	0.160E-05	0.200E+02	0.000E+00
3 0	0.600E-04	0.200E+02	0.000E+00
4 0	0.300E-03	0.100E+02	0.000E+00
5 5	0.200E-06	0.100E+02	0.240E+04
6 0	0.240E-05	0.100E+02	0.000E+00
7 5	0.160E-08	0.100E+02	0.240E+04
8 0	0.200E-05	0.100E+02	0.000E+00
9 0	0.100E-04	0.500E+01	0.000E+00
10 0	0.750E-02	0.100E+02	0.000E+00
11 0	0.100E+00	0.900E+01	0.000E+00
12 0	0.300E-03	0.100E+02	0.000E+00
13 0	0.130E-05	0.200E+02	0.000E+00
14 0	0.100E-06	0.200E+02	0.000E+00
15 0	0.300E-06	0.200E+02	0.000E+00
16 0	0.100E-06	0.200E+02	0.000E+00
17 0	0.100E-06	0.200E+02	0.000E+00
18 0	0.600E-04	0.200E+02	0.000E+00
19 0	0.600E-04	0.200E+02	0.000E+00
20 0	0.600E-04	0.200E+02	0.000E+00
21 0	0.600E-04	0.200E+02	0.000E+00
22 5	0.200E-06	0.100E+02	0.240E+04
23 5	0.160E-08	0.100E+02	0.240E+04
24 0	0.240E-05	0.100E+02	0.000E+00
25 0	0.150E-05	0.100E+02	0.000E+00
26 5	0.500E-06	0.100E+02	0.240E+04
27 5	0.100E-07	0.100E+02	0.240E+04
28 5	0.100E-05	0.100E+02	0.240E+04
29 5	0.200E-07	0.100E+02	0.240E+04
30 0	0.250E-01	0.100E+02	0.000E+00
31 5	0.500E-06	0.100E+02	0.240E+04
32 5	0.400E-08	0.100E+02	0.240E+04
33 5	0.200E-06	0.100E+02	0.240E+04
34 5	0.160E-08	0.100E+02	0.240E+04
35 0	0.300E-03	0.100E+02	0.000E+00
36 0	0.240E-05	0.100E+02	0.000E+00
37 5	0.300E-05	0.100E+02	0.240E+04
38 5	0.300E-06	0.100E+02	0.240E+04
39 5	0.400E-06	0.100E+02	0.240E+04
40 5	0.400E-07	0.100E+02	0.240E+04
41 5	0.100E-05	0.100E+02	0.240E+04
42 5	0.100E-06	0.100E+02	0.240E+04
43 5	0.300E-05	0.100E+02	0.240E+04
44 5	0.300E-06	0.100E+02	0.240E+04
45 5	0.100E-05	0.100E+02	0.240E+04
46 5	0.200E-07	0.100E+02	0.240E+04
47 5	0.200E-05	0.100E+02	0.240E+04
48 5	0.400E-07	0.100E+02	0.240E+04
49 5	0.400E-08	0.100E+02	0.240E+04
50 5	0.160E-08	0.100E+02	0.240E+04
51 0	0.240E-05	0.100E+02	0.000E+00
52 0	0.150E-05	0.100E+02	0.000E+00

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Scram Signal Failure under LOF Conditions
 R01A1 No LOF Signals (neither Automatic nor Manual) to Strings A &
 * 3 0 R0250 R1750 C1600
 R0250 By Low-Flow Channels
 * 2 0 R0100 R0200
 R1750 By SOT Channels
 * 2 0 R1500 R1600
 C1600 No Man1 SCRAM Signal to both String A and String B
 + 1 1 C0610 RHU001CL
 R0100 No LOF Signal to String A
 + 4 0 R0110 R0111 R0112 R0113
 R0200 No LOF Signal to String B
 + 4 0 R0210 R0211 R0212 R0213
 R1500 No SOT Signal to String A
 + 4 0 R1510 R1511 R1512 R1513
 R1600 No SOT Signal to String B
 + 4 0 R1610 R1611 R1612 R1613
 C0610 1MPB and 3MPB Fail to send SCRAM Signal
 * 2 0 C0611 C0631
 R0110 TSOFF, LPFF, HPFF Combintn Fails in String A
 * 3 0 R012C R012B R018A
 R0111 UPFB, LPFF, TSOFF Combintn Fails in String A
 * 3 0 R012C R012B R012D
 R0112 UPFB, HPFF, TSOFF Combintn Fails in String A
 * 3 0 R012C R018A R012D
 R0113 LPFF, HPFF, UPFB Combintn Fails in String A
 * 3 0 R012B R018A R012D
 R0210 TSOFF, LPFF, HPFF Combintn Fails in String B
 * 3 0 R022C R022B R028A
 R0211 UPFB, LPFF, TSOFF Combintn Fails in String B
 * 3 0 R022C R022B R022D
 R0212 UPFB, HPFF, TSOFF Combintn Fails in String B
 * 3 0 R022C R028A R022D
 R0213 LPFF, HPFF, UPFB Combintn Fails in String B
 * 3 0 R022B R028A R022D
 R1510 T/C 503D, F, H Combintn Fails in String A
 * 3 0 R151D R151F R151H
 R1511 T/C 503F, H, C Combintn Fails in String A
 * 3 0 R151F R151H R151C
 R1512 T/C 503H, C, D Combintn Fails in String A
 * 3 0 R151H R151C R151D
 R1513 T/C 503C, D, F Combintn Fails in String A
 * 3 0 R151C R151D R151F
 R1610 T/C 503D, F, H Combintn Fails in String B
 * 3 0 R161D R161F R161H
 R1611 T/C 503F, H, C Combintn Fails in String B
 * 3 0 R161F R161H R161C
 R1612 T/C 503H, C, D Combintn Fails in String B
 * 3 0 R161H R161C R161D
 R1613 T/C 503C, D, F Combintn Fails in String B
 * 3 0 R161C R161D R161F
 C0611 1MPB Fails to Open both Strings
 * 2 0 C061A C061B
 C0631 3MPB Fails to Open both Strings
 * 2 0 C063A C063B
 R012C Contacts TSOFF Fail to Open in String A
 + 1 4 R013C RRE1C1QP RCZ1C1QP RRE1Z0QU RCZ1Z0QU
 R012B Contacts LPFF Fail to Open in String A
 + 1 4 R013B RRE1B1QP RCZ1B1QP RRE1Z0QU RCZ1Z

R015B	Trip Unit Failure	
	+ 0 2 RB11B0QW RB1120QU	
R016B	MV/I Converter Failure	
	+ 0 2 RV11B1QW RV1120QU	
R012A	Contacts HPPE Fail to Open in String A	
	+ 1 4 R013A RRE1A1QP RCZ1A1QP RRE120QU RCZ120QU	
R014D	Failure of Signal Transmission/Conversion	
	+ 0 2 RDC1D1RW RTS1DIRW	
R022A	Contacts HPPE Fail to Open in String B	
	+ 1 4 R013A RRE1A2QP RCZ1A2QP RRE120QU RCZ120QU	
R053D	Failure of T/C or its Ref Junction	
	+ 0 4 RTC3D0RW RTC300RU RRT3D0AW RRT300AU	
R053F	Failure of T/C or its Ref Junction	
	+ 0 4 RTC3F0RW RTC300RU RRT3F0AW RRT300AU	
R053H	Failure of T/C or its Ref Junction	
	+ 0 4 RTC3H0RW RTC300RU RRT3H0AW RRT300AU	
R053C	Failure of T/C or its Ref Junction	
	+ 0 4 RTC3C0RW RTC300RU RRT3C0AW RRT300AU	
R013A	Relays in HPPE circuit remain energized	
	+ 2 2 R015A R016A RFM1A0RW RFM1Z1RU	
R015A	Trip Unit Failure	
	+ 0 2 RB11A0QW RB1120QU	
R016A	MV/I Converter Failure	
	+ 0 3 RV11A1QW RV11A2QW RV1120QU	
RHU001CL	1 0 1 Opertr Error - Failure to Push SCRAM Buttons	
RRE1C1QP	0 23 2 String A Relay Fails to Change Position	
RCZ1C1QP	0 3 3 Contact 3,4 or 5 in String A Fails Closed	
RRE120QU	0 0 4 CCF - All Trip Relays Fail to Move	
RCZ120QU	0 0 5 CCF - All Trip Contacts of both Strings Fail Closed	
RRE1B1QP	0 23 2 String A Relay Fails to Change Position	
RCZ1B1QP	0 3 3 Contact 3,4 or 5 in String A Fails Closed	
ROO1A00	2 0 6 Flow Sensor Can Not Detect LOF Condition (Location)	
RRE1D1QP	0 23 2 String A Relay Fails to Change Position	
RCZ1D1QP	0 3 3 Contact 3,4 or 5 in String A Fails Closed	
RRE1C2QP	0 23 2 String B Relay Fails to Change Position	
RCZ1C2QP	0 3 3 Contact 3,4 or 5 in String B Fails Closed	
RRE1B2QP	0 23 2 String B Relay Fails to Change Position	
RCZ1B2QP	0 3 3 Contact 3,4 or 5 in String B Fails Closed	
RRE1D2QP	0 23 2 String B Relay Fails to Change Position	
RCZ1D2QP	0 3 3 Contact 3,4 or 5 in String B Fails Closed	
RRE3D1QP	0 2 7 String A Relay Fails to Change Position	
RCZ3D1QP	0 3 3 Contact 7 or 8 in String A Fails Closed	
RRE3F1QP	0 2 7 String A Relay Fails to Change Position	
RCZ3F1QP	0 3 3 Contact 7 or 8 in String A Fails Closed	
RRE3H1QP	0 2 7 String A Relay Fails to Change Position	
RCZ3H1QP	0 3 3 Contact 7 or 8 in String A Fails Closed	
RRE3C1QP	0 2 7 String A Relay Fails to Change Position	
RCZ3C1QP	0 3 3 Contact 7 or 8 in String A Fails Closed	
RRE3D2QP	0 2 7 String B Relay Fails to Change Position	
RCZ3D2QP	0 3 3 Contact 7 or 8 in String B Fails Closed	
RRE3F2QP	0 2 7 String B Relay Fails to Change Position	
RCZ3F2QP	0 3 3 Contact 7 or 8 in String B Fails Closed	
RRE3H2QP	0 2 7 String B Relay Fails to Change Position	
RCZ3H2QP	0 3 3 Contact 7 or 8 in String B Fails Closed	
RRE3C2QP	0 2 7 String B Relay Fails to Change Position	
RCZ3C2QP	0 3 3 Contact 7 or 8 in String B Fails Closed	
CSW100CP	0 16 8 LMPB Mech Parts Fail	
CSW10ACP	0 15 9 LMPB Switch of String A Fails to Open	
CSW10BCP	0 15 9 LMPB Switch of String B Fails to Oper	
SSW300CP	0 16 8 3MPB Mech Parts Fail	

SSW30ACP	0 15	9 3MPB Switch of String A Fails to Open
SSW30BCP	0 15	9 3MPB Switch of String B Fails to Open
RFMLC0RW	0 0	10 Flow Sensor Fails High or Fails to Respond
RFMB0RW	0 0	11 Flow Sensor Fails High or Fails to Respond
RFML21RU	0 0	12 CCF - Electro-Magnetic Flow Sensors Fail
ROO1AB00	2 0	13 Flow Sensor Can Detect LOF Condition (Location)
RFML0RW	0 0	14 Flow Sensor Fails High or Fails to Respond
RBIL0QW	0 0	15 Trip Unit of Strings A & B Fails
RDC1D2QW	0 0	16 Bistable DC Power Supply Fails
RBI3D1QW	0 19	17 SOT Trip Unit of Strings A & B Fails
RBII20QU	0 0	18 CCF - All Trip Units in All Channels Fail
RVI3D0QW	0 18	19 MV/I Converter Fails
RVII20QU	0 0	20 CCF - All MV/I Converters Fail
RHU302QL	1 0	21 CCF - Human Error - SOT Trip Points Set Too High
RB13F1QW	0 19	17 SOT Trip Unit of Strings A & B Fails
RV13F0QW	0 18	19 MV/I Converter Fails
RB13H1QW	0 19	17 SOT Trip Unit of Strings A & B Fails
RV13H0QW	0 18	19 MV/I Converter Fails
RB13C1QW	0 19	17 SOT Trip Unit of Strings A & B Fails
RV13C0QW	0 18	19 MV/I Converter Fails
RDC1C1AW	0 0	22 Transmitter DC Power Supply Fails
RTS1C1AW	0 0	23 Pressure Transmitters Fail
RBII0QW	0 25	24 Trip Unit of Strings A & B Fails
RVII1C1QW	0 26	25 MV/I Converter Fails
RBII0QW	0 25	24 Trip Unit of Strings A & B Fails
RVII1B1QW	0 26	25 MV/I Converter Fails
RRE1A1QP	0 23	2 String A Relay Fails to Change Position
RCZ1A1QP	0 3	3 Contact 3, 4 or 5 in String A Fails Closed
RDC1D1RW	0 0	26 Transmitter DC Power Supply Fails
RTS1D1RW	0 0	27 Pressure Transmitters Fail
RRE1A2QP	0 23	2 String B Relay Fails to Change Position
RCZ1A2QP	0 3	3 Contact 3, 4 or 5 in String B Fails Closed
RTC3D0RW	0 20	28 T/C 503D Fails Low - Short/Open
RTC300RU	0 0	29 CCF - All T/C's Fail Low
RRT3D0AW	0 21	30 T/C 503D Ref Junction Fails High
RRT300AU	0 0	31 CCF - All Ref Junctions Fail High
RTC3F0RW	0 20	28 T/C 503F Fails Low - Short/Open
RRT3F0AW	0 21	30 T/C 503F Ref Junction Fails High
RTC3H0RW	0 20	28 T/C 503H Fails Low - Short/Open
RRT3H0AW	0 21	30 T/C 503H Ref Junction Fails High
RTC3C0RW	0 20	28 T/C 503C Fails Low - Short/Open
RRT3C0AW	0 21	30 T/C 503C Ref Junction Fails High
RFM1A0RW	0 0	32 Flow Sensor Fails High or Fails to Respond
RBII1A0QW	0 25	24 Trip Unit of Strings A & B Fails
RVII1A1QW	0 26	25 MV/I Converter Fails
RVII1A2QW	0 26	25 DMV/I Converter Fails
0.240E+02		
0.240E+04		
1 0	0.500E-01	0.100E+02 0.000E+00
2 5	0.200E-06	0.100E+02 0.168E+03
3 0	0.300E-03	0.100E+02 0.000E+00
4 5	0.160E-08	0.100E+02 0.168E+03
5 0	0.240E-05	0.100E+02 0.000E+00
6 F	0.000E+00	0.100E+01
7 5	0.200E-06	0.100E+02 0.240E+04
8 0	0.200E-05	0.100E+02 0.000E+00
9 0	0.100E-04	0.500E+01 0.000E+00
10 5	0.300E-05	0.100E+02 0.168E+03
11 5	0.300E-05	0.100E+02 0.168E+03
12 5	0.300E-06	0.100E+02 0.168E+03

13	T	0.100E+01	0.100E+01	0.168E+03
14	5	0.300E-05	0.100E+02	0.168E+03
15	5	0.500E-06	0.100E+02	0.168E+03
16	5	0.300E-05	0.100E+02	0.168E+03
17	5	0.500E-06	0.100E+02	0.240E+04
18	5	0.400E-08	0.100E+02	0.168E+03
19	5	0.100E-05	0.100E+02	0.240E+04
20	5	0.800E-08	0.100E+02	0.168E+03
21	0	0.250E-01	0.100E+02	0.000E+00
22	5	0.300E-05	0.100E+02	0.168E+03
23	5	0.200E-04	0.100E+02	0.168E+03
24	5	0.500E-06	0.100E+02	0.168E+03
25	5	0.100E-05	0.100E+02	0.168E+03
26	5	0.300E-05	0.100E+02	0.168E+03
27	5	0.200E-04	0.100E+02	0.168E+03
28	5	0.100E-05	0.100E+02	0.240E+04
29	5	0.200E-07	0.100E+02	0.240E+04
30	5	0.200E-05	0.100E+02	0.240E+04
31	5	0.400E-07	0.100E+02	0.240E+04
32	5	0.300E-05	0.100E+02	0.168E+03

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Rod Drop Failure under LOF Conditions
R00C1 Elect & Mech Failures of CRs & SRs
C2500 * 2 0 C2500 S1010
Mech or Elect Components Failure, but Signal is OK
* 2 0 C3000 C5000
S1010 Safety Rods Failure to SCRAM
+ 2 0 S1600 S0500
C3000 Control Rods 3 to 6 affected by CPB-1 Contacts
+ 2 0 C3310 C3360
C5000 Control Rods 7, 9 to 12 affected by CPB-2 Contacts
+ 2 0 C3310 C3760
S1600 No Man1 SCRAM Signal to both String A & String B
+ 2 0 S0631 S1637
S0500 Mech or Elect Components Failure, but Signal is OK
+ 2 1 S0510 S0560 SCL000RP
C3310 Mechanical Components Failure
+ 1 1 C3315 CCL00PRU
C3360 Rod Release Contacts Fail to Open
* 4 0 C3365 C3375 C3385 C3395
C3760 Rod Release Contacts Fail to Open
* 4 0 C3365 C3375 C3785 C3395
S0631 3MPB Fails to Open both Strings
* 2 0 S063A S063B
S1637 Operator Failure to SCRAM SRs
+ 2 0 S0638 S1639
S0510 Mechanical Components Failure
+ 1 1 S0515 SCD200RW
S0560 Rods Release Contacts Fail to Open the 24 VDC Ckt
* 2 0 S0570 S0580
C3315 Force Larger than CR Wght is Required
+ 2 0 C3320 C3340
C3365 Contacts CPC-1 Fail to Open
+ 0 4 RCZCC1QP RREC1QP RCZC00QU RREC00QU
C3375 Contacts CPA Fail to Open
+ 0 4 RCZCPAQP RRECPAQP RCZC00QU RREC00QU
C3385 Contacts CPB-1 Fail to Open
+ 0 4 RCZCB1QP RRECBIQP RCZC00QU RREC00QU
C3395 Contacts CPY Fail to Open
+ 0 6 RCZCPYQP RRECPIQP RCZCP2QP RREC2QP RCZC00QU RREC00QU
C3785 Contacts CPB-2 Fail to Open
+ 0 4 RCZCB2QP RRECBIQP RCZC00QU RREC00QU
S063A 3MPB Fails to Open String A
+ 0 2 SSW300CP SSW30ACP
S063B 3MPB Fails to Open String B
+ 0 2 SSW300CP SSW30BCP
S0638 With Successful Automatic SCRAM Signal
+ 0 1 RHU00ZCL
S1639 With Failed Automatic SCRAM Signal
* 1 1 R0240 RHU001CL
S0515 Force Larger than SRs Wght & Spring Force is Required
+ 2 0 S0520 S0540
S0570 Contacts SRO Fail to Open
+ 0 4 RRESROQP RCZSROQP RREC00QU RCZC00QU
S0580 Contacts SRO-X Fail to Open
+ 0 4 RRESRXQP RCZSRXQP RREC00QU RCZC00QU
C3320 Forces from CR Drive
+ 0 3 CCD10GRU CCD20WRU CCD30IRU
C3340 Forces from CR and its Thimble
+ 0 2 CSR10GRU CSR30IRU

NO LOW SIGNAL TO BOTH STRING A AND STRING B

R0240 * 2 0 R0250 R1750
S0520 Forces from SR Drive
+ 0 2 SCD100RG SCD300RI
S0540 Forces from SRs and their Thimble
+ 0 4 SSR100RG CSR10GRU SSR300RI CSR30IRU
R0250 By Low-Flow Channels
* 2 0 R0100 R0200
R1750 By SOT Channels
* 2 0 R1500 R1600
R0100 No LOF Signal to String A
+ 4 0 R0110 R0111 R0112 R0113
R0200 No LOF Signal to String B
+ 4 0 R0210 R0211 R0212 R0213
R1500 No SOT Signal to String A
+ 4 0 R1510 R1511 R1512 R1513
R1600 No SOT Signal to String B
+ 4 0 R1610 R1611 R1612 R1613
R0110 TSOF, LPPE, HPPE Combintn Fails in String A
* 3 0 R012C R012B R018A
R0111 UPFB, LPPE, TSOF Combintn Fails in String A
* 3 0 R012C R012B R012D
R0112 UPFB, HPPE, TSOF Combintn Fails in String A
* 3 0 R012C R018A R012D
R0113 LPPE, HPPE, UPFB Combintn Fails in String A
* 3 0 R012B R018A R012D
R0210 TSOF, LPPE, HPPE Combintn Fails in String B
* 3 0 R022C R022B R028A
R0211 UPFB, LPPE, TSOF Combintn Fails in String B
* 3 0 R022C R022B R022D
R0212 UPFB, HPPE, TSOF Combintn Fails in String B
* 3 0 R022C R028A R022D
R0213 LPPE, HPPE, UPFB Combintn Fails in String B
* 3 0 R022B R028A R022D
R1510 T/C 503D, F, H Combintn Fails in String A
* 3 0 R151D R151F R151H
R1511 T/C 503F, H, C Combintn Fails in String A
* 3 0 R151F R151H R151C
R1512 T/C 503H, C, D Combintn Fails in String A
* 3 0 R151H R151C R151D
R1513 T/C 503C, D, F Combintn Fails in String A
* 3 0 R151C R151D R151F
R1610 T/C 503D, F, H Combintn Fails in String B
* 3 0 R161D R161F R161H
R1611 T/C 503F, H, C Combintn Fails in String B
* 3 0 R161F R161H R161C
R1612 T/C 503H, C, D Combintn Fails in String B
* 3 0 R161H R161C R161D
R1613 T/C 503C, D, F Combintn Fails in String B
* 3 0 R161C R161D R161F
R012C Contacts TSOF Fail to Open in String A
+ 1 4 R013C RRE1C1QP RCZ1C1QP RRE1Z0QU RCZ1Z0QU
R012B Contacts LPPE Fail to Open in String A
+ 1 4 R013B RRE1B1QP RCZ1B1QP RRE1Z0QU RCZ1Z0QU
R018A Channel HPPE Fails to Respond
+ 1 1 R017A R001AA00
R012D Contacts UPFB Fail to Open in String A
+ 1 4 R013D RRE1D1QP RCZ1D1QP RRE1Z0QU RCZ1Z0QU
R022C Contacts TSOF Fail to Open in String B
+ 1 4 R013C RRE1C2QP RCZ1C2QP RRE1Z0QU RCZ1Z0QU

R022B	Contacts LPFF Fail to Open in String B	
R028A	+ 1 4 R013B RRE1B2QP RCZ1B2QP RRE1Z0QU RCZ1Z0QU	
	Channel HPFF Fails to Respond	
	+ 1 1 R027A R001AA00	
R022D	Contacts UPFB Fail to Open in String B	
	+ 1 4 R013D RRE1D2QP RCZ1D2QP RRE1Z0QU RCZ1Z0QU	
R151D	503D Contacts Fail to Open in String A	
	+ 1 4 R152D RRE3D1QP RCZ3D1QP RRE1Z0QU RCZ1Z0QU	
R151F	503F Contacts Fail to Open in String A	
	+ 1 4 R152F RRE3F1QP RCZ3F1QP RRE1Z0QU RCZ1Z0QU	
R151H	503H Contacts Fail to Open in String A	
	+ 1 4 R152H RRE3H1QP RCZ3H1QP RRE1Z0QU RCZ1Z0QU	
R151C	503C Contacts Fail to Open in String A	
	+ 1 4 R152C RRE3C1QP RCZ3C1QP RRE1Z0QU RCZ1Z0QU	
R161D	503D Contacts Fail to Open in String B	
	+ 1 4 R152D RRE3D2QP RCZ3D2QP RRE1Z0QU RCZ1Z0QU	
R161F	503F Contacts Fail to Open in String B	
	+ 1 4 R152F RRE3F2QP RCZ3F2QP RRE1Z0QU RCZ1Z0QU	
R161H	503H Contacts Fail to Open in String B	
	+ 1 4 R152H RRE3H2QP RCZ3H2QP RRE1Z0QU RCZ1Z0QU	
R161C	503C Contacts Fail to Open in String B	
	+ 1 4 R152C RRE3C2QP RCZ3C2QP RRE1Z0QU RCZ1Z0QU	
R013C	Relays in TSOE Remain Energized	
	+ 3 1 R014C R015C R016C RFM1C0RW	
R013B	Relays in LPFF circuit remain energized	
	+ 2 2 R015B R016B RFM1B0RW RFM1Z1RU	
R017A	Componets of HPFF Channel Fail to Open String A	
	* 1 1 R012A R001AB00	
R013D	Relays in UPFB Remain Energized	
	+ 1 3 R014D RFM1D0RW RB11D0QW RDC1D2QW	
R027A	Componets of HPFF Channel Fail to Open String B	
	* 1 1 R022A R001AB00	
R152D	Relays in 503D Circuit Remain Energized	
	+ 1 5 R053D RB13D1QW RB11Z0QU RVI3D0QW RVI1Z0QU RHU302QL	
R152F	Relays in 503F Circuit Remain Energized	
	+ 1 5 R053F RB13F1QW RB11Z0QU RVI3F0QW RVI1Z0QU RHU302QL	
R152H	Relays in 503H Circuit Remain Energized	
	+ 1 5 R053H RB13H1QW RB11Z0QU RVI3H0QW RVI1Z0QU RHU302QL	
R152C	Relays in 503C Circuit Remain Energized	
	+ 1 5 R053C RB13C1QW RB11Z0QU RVI3C0QW RVI1Z0QU RHU302QL	
R014C	Failure of Signal Transmission / Conversion	
	+ 0 2 RDC1C1AW Rts1C1AW	
R015C	Trip Unit Failure	
	+ 0 2 RB11C0QW RB11Z0QU	
R016C	MV/I Converter Failure	
	+ 0 2 RVI1C1QW RVI1Z0QU	
R015B	Trip Unit Failure	
	+ 0 2 RB11B0QW RB11Z0QU	
R016B	MV/I Converter Failure	
	+ 0 2 RVI1B1QW RVI1Z0QU	
R012A	Contacts HPFF Fail to Open in String A	
	+ 1 4 R013A RRE1A1QP RCZ1A1QP RRE1Z0QU RCZ1Z0QU	
R014D	Failure of Signal Transmission/Conversion	
	+ 0 2 RDC1D1RW Rts1D1RW	
R022A	Contacts HPFF Fail to Open in String B	
	+ 1 4 R013A RRE1A2QP RCZ1A2QP RRE1Z0QU RCZ1Z0QU	
R053D	Failure of T/C or its Ref Junction	
	+ 0 4 RTC3D0RW RTC300RU RRT3D0AW RRT300AU	
R053F	Failure of T/C or its Ref Junction	
	+ 0 4 RTC3F0RW RTC300RU RRT3F0AW RRT300AU	

R053C	+ 0 4 RTC3H0RW RTC300RU RRT3H0AW RRT300AU	
	Failure of T/C or its Ref Junction	
R013A	+ 0 4 RTC3C0RW RTC300RU RRT3C0AW RRT300AU	
	Relays in HPPF circuit remain energized	
R015A	+ 2 2 R015A R016A RFM1A0RW RFM121RU	
	Trip Unit Failure	
R016A	+ 0 2 RB11A0QW RB1120QU	
	MV/I Converter Failure	
	+ 0 3 RV11A1QW RV11A2QW RV1120QU	
SCL000RP	0 0 1 Magnetic Clutch Fails to Release SRs	
CCLO0PRU	0 0 2 CCF - Magnetic Clutches Fail to Release CRs	
SCD200RW	0 0 3 Failure of Rise Mechanism, Connecting Shafts, Bearings	
RCZCC1QP	0 24 4 Contacts Fail Closed	
RRECC1QP	0 17 5 CP-1 Relay Fails to Change Position	
RCZC00QU	0 0 6 CCF - Contacts of All CP-Relays Fail Closed	
RREC00QU	0 0 7 CCF - All CP-Relays on Strings A & B Fail	
RCZCPAQP	0 24 4 Contacts Fail Closed	
RRECPAQP	0 17 5 CPA Relay Fails to Change Position	
RCZCB1QP	0 24 4 Contacts Fail Closed	
RRECB1QP	0 17 5 CPB-1 Relay Fails to Change Position	
RCZCPYQP	0 24 4 Contacts Fail Closed	
RRECPYQP	0 17 5 CPY Relay Fails to Change Position	
RCZCP2QP	0 24 4 Contacts Fail Closed	
RRECP2QP	0 17 5 CP-2 Relay Fails to Change Position	
RCZCB2QP	0 24 4 Contacts Fail Closed	
RRECB2QP	0 17 5 CPB-2 Relay Fails to Change Position	
SSW300CP	0 16 8 3MPB Mech Parts Fail	
SSW30ACP	0 15 9 3MPB Switch of String A Fails to Open	
SSW30BCP	0 15 9 3MPB Switch of String B Fails to Open	
RHU00ZCL	1 0 10 Opertr Error - Failure to Push 3MPB SCRAM Button	
RRESROQP	0 17 5 SRO Relay Fails to Change Position	
RCZSROQP	0 3 12 Contacts Fail Closed	
RRESRXQP	0 17 5 SRO-X Relay Fails to Change Position	
RCZSRXQP	0 3 12 Contacts Fail Closed	
CCD10GRU	0 0 13 CCF - Bowing and Binding / Inside Tank	
CCD20WRU	0 0 14 CCF - Bowing and Binding/Above Rotating Plug	
CCD30IRU	0 0 15 CCF - Forces due to Foreign Materials in Sodium	
CSR10GRU	0 0 16 CCF - Forces due to Bowing and Binding	
CSR30IRU	0 0 17 CCF - Forces due to Foreign Materials in Sodium	
SCD10ORG	0 0 18 Forces due to Bowing and Binding	
SCD300RI	0 0 19 Forces due to Foreign Materials in Sodium	
SSR10ORG	0 0 20 Forces due to Bowing and Binding	
SSR300RI	0 0 21 Forces due to Foreign Materials in Sodium	
RREL1C1QP	0 23 22 String A Relay Fails to Change Position	
RCZ1C1QP	0 3 12 Contact 3,4 or 5 in String A Fails Closed	
RREL120QU	0 0 23 CCF - All Trip Relays Fail to Move	
RCZ120QU	0 0 24 CCF - All Trip Contacts of both Strings Fail Closed	
RREL1B1QP	0 23 22 String A Relay Fails to Change Position	
RCZ1B1QP	0 3 12 Contact 3,4 or 5 in String A Fails Closed	
ROOLAA00	2 0 25 Flow Sensor Can Not Detect LOF Condition (Location)	
RREL1D1QP	0 23 22 String A Relay Fails to Change Position	
RCZ1D1QP	0 3 12 Contact 3,4 or 5 in String A Fails Closed	
RREL1C2QP	0 23 22 String B Relay Fails to Change Position	
RCZ1C2QP	0 3 12 Contact 3,4 or 5 in String B Fails Closed	
RREL1B2QP	0 23 22 String B Relay Fails to Change Position	
RCZ1B2QP	0 3 12 Contact 3,4 or 5 in String B Fails Closed	
RREL1D2QP	0 23 22 String B Relay Fails to Change Position	
RCZ1D2QP	0 3 12 Contact 3,4 or 5 in String B Fails Closed	

RRE3D1QP	0	2	26 String A Relay Fails to Change Position
RCZ3D1QP	0	3	12 Contact 7 or 8 in String A Fails Closed
RRE3F1QP	0	2	26 String A Relay Fails to Change Position
RCZ3F1QP	0	3	12 Contact 7 or 8 in String A Fails Closed
RRE3H1QP	0	2	26 String A Relay Fails to Change Position
RCZ3H1QP	0	3	12 Contact 7 or 8 in String A Fails Closed
RRE3C1QP	0	2	26 String A Relay Fails to Change Position
RCZ3C1QP	0	3	12 Contact 7 or 8 in String A Fails Closed
RRE3D2QP	0	2	26 String B Relay Fails to Change Position
RCZ3D2QP	0	3	12 Contact 7 or 8 in String B Fails Closed
RRE3F2QP	0	2	26 String B Relay Fails to Change Position
RCZ3F2QP	0	3	12 Contact 7 or 8 in String B Fails Closed
RRE3H2QP	0	2	26 String B Relay Fails to Change Position
RCZ3H2QP	0	3	12 Contact 7 or 8 in String B Fails Closed
RRE3C2QP	0	2	26 String B Relay Fails to Change Position
RCZ3C2QP	0	3	12 Contact 7 or 8 in String B Fails Closed
RFM1CORW	0	0	27 Flow Sensor Fails High or Fails to Respond
RFM1BORW	0	0	28 Flow Sensor Fails High or Fails to Respond
RFM1ZIRU	0	0	29 CCF - Electro-Magnetic Flow Sensors Fail
ROOLAB00	2	0	30 Flow Sensor Can Detect LOF Condition (Location)
RFM1DORW	0	0	31 Flow Sensor Fails High or Fails to Respond
RBILDOQW	0	0	32 Trip Unit of Strings A & B Fails
RDC1D2QW	0	0	33 Bistable DC Power Supply Fails
RB13D1QW	0	19	34 SOT Trip Unit of Strings A & B Fails
RB1120QU	0	0	35 CCF - All Trip Units in All Channels Fail
RV13D0QW	0	18	36 MV/I Converter Fails
RV1120QU	0	0	37 CCF - All MV/I Converters Fail
RHU302QL	1	0	38 CCF - Human Error - SOT Trip Points Set Too High
RB13F1QW	0	19	34 SOT Trip Unit of Strings A & B Fails
RV13F0QW	0	18	36 MV/I Converter Fails
RB13H1QW	0	19	34 SOT Trip Unit of Strings A & B Fails
RV13H0QW	0	18	36 MV/I Converter Fails
RB13C1QW	0	19	34 SOT Trip Unit of Strings A & B Fails
RV13C0QW	0	18	36 MV/I Converter Fails
RDC1C1AW	0	0	39 Transmitter DC Power Supply Fails
RTS1C1AW	0	0	40 Pressure Transmitters Fail
RB1LC0QW	0	25	41 Trip Unit of Strings A & B Fails
RV1LC1QW	0	26	42 MV/I Converter Fails
RB1LB0QW	0	25	41 Trip Unit of Strings A & B Fails
RV1LB1QW	0	26	42 MV/I Converter Fails
RRE1A1QP	0	23	22 String A Relay Fails to Change Position
RCZ1A1QP	0	3	12 Contact 3, 4 or 5 in String A Fails Closed
RTC3D0RW	0	20	45 T/C 503D Fails Low - Short/Open
RTC300RU	0	0	46 CCF - All T/C's Fail Low
RRT3D0AW	0	21	47 T/C 503D Ref Junction Fails High
RRT300AU	0	0	48 CCF - All Ref Junctions Fail High
RTC3F0RW	0	20	45 T/C 503F Fails Low - Short/Open
RRT3F0AW	0	21	47 T/C 503F Ref Junction Fails High
RTC3H0RW	0	21	45 T/C 503H Fails Low - Short/Open
RRT3H0AW	0	21	47 T/C 503H Ref Junction Fails High
RTC3C0RW	0	20	45 T/C 503C Fails Low - Short/Open
RRT3C0AW	0	21	47 T/C 503C Ref Junction Fails High
RFM1A0RW	0	0	49 Flow Sensor Fails High or Fails to Respond
RB1LA0QW	0	25	41 Trip Unit of Strings A & B Fails
RV1LA1QW	0	26	42 MV/I Converter Fails
RV1LA2QW	0	26	42 DMV/I Converter Fails

0.240E+04

1	0	0.200E-03	0.200E+02	0.000E+00
2	0	0.160E-05	0.200E+02	0.000E+00
3	0	0.600E-04	0.200E+02	0.000E+00
4	0	0.300E-03	0.100E+02	0.000E+00
5	5	0.200E-06	0.100E+02	0.240E+04
6	0	0.240E-05	0.100E+02	0.000E+00
7	5	0.160E-08	0.100E+02	0.240E+04
8	0	0.200E-05	0.100E+02	0.000E+00
9	0	0.100E-04	0.500E+01	0.000E+00
10	0	0.750E-02	0.100E+02	0.000E+00
11	0	0.500E-01	0.100E+02	0.000E+00
12	0	0.300E-03	0.100E+02	0.000E+00
13	0	0.130E-05	0.200E+02	0.000E+00
14	0	0.100E-06	0.200E+02	0.000E+00
15	0	0.300E-06	0.200E+02	0.000E+00
16	0	0.100E-06	0.200E+02	0.000E+00
17	0	0.100E-06	0.200E+02	0.000E+00
18	0	0.600E-04	0.200E+02	0.000E+00
19	0	0.600E-04	0.200E+02	0.000E+00
20	0	0.600E-04	0.200E+02	0.000E+00
21	0	0.600E-04	0.200E+02	0.000E+00
22	5	0.200E-06	0.100E+02	0.168E+03
23	5	0.160E-08	0.100E+02	0.168E+03
24	0	0.240E-05	0.100E+02	0.000E+00
25	F	0.000E+00	0.100E+01	
26	5	0.200E-06	0.100E+02	0.240E+04
27	5	0.300E-05	0.100E+02	0.168E+03
28	5	0.300E-05	0.100E+02	0.168E+03
29	5	0.300E-06	0.100E+02	0.168E+03
30	T	0.100E+01	0.100E+01	
31	5	0.300E-05	0.100E+02	0.168E+03
32	5	0.500E-06	0.100E+02	0.168E+03
33	5	0.300E-05	0.100E+02	0.168E+03
34	5	0.500E-06	0.100E+02	0.240E+04
35	5	0.400E-08	0.100E+02	0.168E+03
36	5	0.100E-05	0.100E+02	0.240E+04
37	5	0.800E-08	0.100E+02	0.168E+03
38	0	0.250E-01	0.100E+02	0.000E+00
39	5	0.300E-05	0.100E+02	0.168E+03
40	5	0.200E-04	0.100E+02	0.168E+03
41	5	0.500E-06	0.100E+02	0.168E+03
42	5	0.100E-05	0.100E+02	0.168E+03
43	5	0.300E-05	0.100E+02	0.168E+03
44	5	0.200E-04	0.100E+02	0.168E+03
45	5	0.100E-05	0.100E+02	0.240E+04
46	5	0.200E-07	0.100E+02	0.240E+04
47	5	0.200E-05	0.100E+02	0.240E+04
48	5	0.400E-07	0.100E+02	0.240E+04
49	5	0.300E-05	0.100E+02	0.168E+03

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RSS, Fault Tree of Rod Drop Failure under LOF Condition due to LOCP

R00C3	Elect & Mech Failure of CRs & SRs in Event of LOCP			
	* 2 0 C3310 S1011			
C3310	Mechanical Components Failure			
	+ 1 1 C3315 CCL00PRU			
S1011	Mech or Elect Components Failure, but Signal is OK			
	+ 1 1 S0510 SCL000RP			
C3315	Force Larger than CR Wght is Required			
	+ 2 0 C3320 C3340			
S0510	Mechanical Components Failure			
	+ 1 1 S0515 SCD200RW			
C3320	Forces from CR Drive			
	+ 0 3 CCD10GRU CCD20WRU CCD30IRU			
C3340	Forces from CR and its Thimble			
	+ 0 2 CSR10GRU CSR30IRU			
S0515	Force Larger than SRs Wght & Spring Force is Required			
	+ 2 0 S0520 S0540			
S0520	Forces from SR Drive			
	+ 0 2 SCD100RG SCD300RI			
S0540	Forces from SRs and their Thimble			
	+ 0 4 SSR100RG CSR10GRU SSR300RI CSR30IRU			
CCL00PRU	0 0 1 CCF - Magnetic Clutches Fail to Release CRs			
SCL000RP	0 0 2 Magnetic Clutch Fails to Release SRs			
SCD200RW	0 0 3 Failure of Rise Mechanism, Connecting Shafts, Bearings			
CCD10GRU	0 0 4 CCF - Bowing and Binding / Inside Tank			
CCD20WRU	0 0 5 CCF - Bowing and Binding/Above Rotating Plug			
CCD30IRU	0 0 6 CCF - Forces due to Foreign Materials in Sodium			
CSR10GRU	0 0 7 CCF - Forces due to Bowing and Binding			
CSR30IRU	0 0 8 CCF - Forces due to Foreign Materials in Sodium			
SCD100RG	0 0 9 Forces due to Bowing and Binding			
SCD300RI	0 0 10 Forces due to Foreign Materials in Sodium			
SSR100RG	0 0 11 Forces due to Bowing and Binding			
SSR300RI	0 0 12 Forces due to Foreign Materials in Sodium			
	0.240E+02			
	0.240E+04			
	1 0 0.160E-05 0.200E+02 0.000E+00			
	2 0 0.200E-03 0.200E+02 0.000E+00			
	3 0 0.600E-04 0.200E+02 0.000E+00			
	4 0 0.130E-05 0.200E+02 0.000E+00			
	5 0 0.100E-06 0.200E+02 0.000E+00			
	6 0 0.300E-06 0.200E+02 0.000E+00			
	7 0 0.100E-06 0.200E+02 0.000E+00			
	8 0 0.100E-06 0.200E+02 0.000E+00			
	9 0 0.600E-04 0.200E+02 0.000E+00			
	10 0 0.600E-04 0.200E+02 0.000E+00			
	11 0 0.600E-04 0.200E+02 0.000E+00			
	12 0 0.600E-04 0.200E+02 0.000E+00

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Failure to SCRAM under LHOS Conditions
 R0003 CRs and SRs Failure to SCRAM under LOHS Conditions
 * 2 0 C4060 S4010
 C4060 CRs Manual SCRAM Failure
 + 2 0 C2500 C4600
 S4010 Safety Rods Failure to SCRAM
 + 2 0 S4600 S0500
 C2500 Mech or Elect Components Failure, but Signal is OK
 * 2 0 C3000 C5000
 C4600 No Manual SCRAM Signal to Both String A and String B
 + 1 1 C0610 RHU006CL
 S4600 No Manual SCRAM Signal to Both String A and String B
 + 1 1 S0631 RHU006CL
 S0500 Mech or Elect Components Failure, but Signal is OK
 + 2 1 S0510 S0560 SCL000RP
 C3000 Control Rods 3 to 6 affected by CPB-1 Contacts
 + 2 0 C3310 C3360
 C5000 Control Rods 7, 9 to 12 affected by CPB-2 Contacts
 + 2 0 C3310 C3760
 C0610 1MPB and 3MPB Fail to send SCRAM Signal
 * 2 0 C0611 C0631
 S0631 3MPB Fails to Open both Strings
 * 2 0 S063A S063B
 S0510 Mechanical Components Failure
 + 1 1 S0515 SCD200RW
 S0560 Rods Release Contacts Fail to Open the 24 VDC Ckt
 * 2 0 S0570 S0580
 C3310 Mechanical Components Failure
 + 1 1 C3315 CCL00PRU
 C3360 Rod Release Contacts Fail to Open
 * 4 0 C3365 C3375 C3385 C3395
 C3760 Rod Release Contacts Fail to Open
 * 4 0 C3365 C3375 C3785 C3395
 C0611 1MPB Fails to Open both Strings
 * 2 0 C061A C061B
 C0631 3MPB Fails to Open both Strings
 * 2 0 C063A C063B
 S063A 3MPB Fails to Open String A
 + 0 2 SSW300CP SSW30ACP
 S063B 3MPB Fails to Open String B
 + 0 2 SSW300CP SSW30BCP
 S0515 Force Larger than SRs Wght & Spring Force is Required
 + 2 0 S0520 S0540
 S0570 Contacts SRO Fail to Open
 + 0 4 RRESROQP RCZSROQP RREC000QU RCZC000QU
 S0580 Contacts SRO-X Fail to Open
 + 0 4 RRESRXQP RCZSRXQP RREC000QU RCZC000QU
 C3315 Force Larger than CR Wght is Required
 + 2 0 C3320 C3340
 C3365 Contacts CPC-1 Fail to Open
 + 0 4 RCZCCLQP RRECCIQP RCZC000QU RREC000QU
 C3375 Contacts CPA Fail to Open
 + 0 4 RCZCPAQP RRECPAQP RCZC000QU RREC000QU
 C3385 Contacts CPB-1 Fail to Open
 + 0 4 RCZCB1QP RRECB1QP RCZC000QU RREC000QU
 C3395 Contacts CPY Fail to Open
 + 0 6 RCZCPYQP RRECYPQP RCZCP2QP RREC2QP RCZC000QU RREC000QU
 C3785 Contacts CPB-2 Fail to Open
 + 0 4 RCZCB2QP RRECB2QP RCZC000QU RREC000QU

C061A	1MPB Fails to Open String A	
	+ 0 2 CSW100CP CSW10ACP	
C061B	1MPB Fails to Open String B	
	+ 0 2 CSW100CP CSW10BCP	
C063A	3MPB Fails to Open String A	
	+ 0 2 SSW300CP SSW30ACP	
C063B	3MPB Fails to Open String B	
	+ 0 2 SSW300CP SSW30BCP	
S0520	Forces from SR Drive	
	+ 0 2 SCD100RG SCD300RI	
S0540	Forces from SRs and their Thimble	
	+ 0 4 SSR100RG CSR10GRU SSR300RI CSR30IRU	
C3320	Forces from CR Drive	
	+ 0 3 CCD10GRU CCD20WRU CCD30IRU	
C3340	Forces from CR and its Thimble	
	+ 0 2 CSR10GRU CSR30IRU	
RHU006CL	0 0 1 Operator Failure to Push SCRAM Buttons u/LOHS Conditions	
SC1000RP	0 0 2 Magnetic Clutch Fails to Release SRs	
SCD200RW	0 0 3 Failure of Rise Mechanism, Connecting Shafts, Bearings	
CCL00PRU	0 0 4 CCF - Magnetic Clutches Fail to Release CRs	
SSW300CP	0 16 5 3MPB Mech Parts Fail	
SSW30ACP	0 15 6 3MPB Switch of String A Fails to Open	
SSW30BCP	0 15 6 3MPB Switch of String B Fails to Open	
RRESROQP	0 17 7 SRO Relay Fails to Change Position	
RCZSROQP	0 3 8 Contacts Fail Closed	
RREC00QU	0 0 9 CCF - All CP-Relays on Strings A & B Fail	
RCZC00QU	0 0 10 CCF - Contacts of All CP-Relays Fail Closed	
RRESRXQP	0 17 7 SRO-X Relay Fails to Change Position	
RCZSRXQP	0 3 8 Contacts Fail Closed	
RCZCC1QP	0 24 11 Contacts Fail Closed	
RRECC1QP	0 17 7 CPC-1 Relay Fails to Change Position	
RCZCPAQP	0 24 11 Contacts Fail Closed	
RRECPAQP	0 17 7 CPA Relay Fails to Change Position	
RCZCB1QP	0 24 11 Contacts Fail Closed	
RECB1QP	0 17 7 CPB-1 Relay Fails to Change Position	
RCZCPYQP	0 24 11 Contacts Fail Closed	
RRECPYQP	0 17 7 CPY Relay Fails to Change Position	
RCZCP2QP	0 24 11 Contacts Fail Closed	
RRECP2QP	0 17 7 CP-2 Relay Fails to Change Position	
RCZCB2QP	0 24 11 Contacts Fail Closed	
RRECB2QP	0 17 7 CPB-2 Relay Fails to Change Position	
CSW100CP	0 16 5 1MPB Mech Parts Fail	
CSW10ACP	0 15 6 1MPB Switch of String A Fails to Open	
CSW10BCP	0 15 6 1MPB Switch of String B Fails to Open	
SCD100RG	0 0 12 Forces due to Bowing and Binding	
SCD300RI	0 0 13 Forces due to Foreign Materials in Sodium	
SSR100RG	0 0 14 Forces due to Bowing and Binding	
CSR10GRU	0 0 15 CCF - Forces due to Bowing and Binding	
SSR300RI	0 0 16 Forces due to Foreign Materials in Sodium	
CSR30IRU	0 0 17 CCF - Forces due to Foreign Materials in Sodium	
CCD10GRU	0 0 18 CCF - Bowing and Binding / Inside Tank	
CCD20WRU	0 0 19 CCF - Bowing and Binding/Above Rotating Plug	
CCD30IRU	0 0 20 CCF - Forces due to Foreign Materials in Sodium	
	0.240E+02	
	0.240E+04	
	1 0 0.100E-05 0.100E+02 0.000E+00	
	2 0 0.200E-03 0.200E+02 0.000E+00	
	3 0 0.600E-04 0.200E+02 0.000E+00	
	4 0 0.160E-05 0.200E+02 0.000E+00	
	5 0 0.200E-05 0.100E+02 0.000E+00	

6	0	0.100E-04	0.500E+01	0.000E+00
7	5	0.200E-06	0.100E+02	0.240E+04
8	0	0.300E-03	0.100E+02	0.000E+00
9	5	0.160E-08	0.100E+02	0.240E+04
10	0	0.240E-05	0.100E+02	0.000E+00
11	0	0.300E-03	0.100E+02	0.000E+00
12	0	0.600E-04	0.200E+02	0.000E+00
13	0	0.600E-04	0.200E+02	0.000E+00
14	0	0.600E-04	0.200E+02	0.000E+00
15	0	0.100E-06	0.200E+02	0.000E+00
16	0	0.600E-04	0.200E+02	0.000E+00
17	0	0.100E-06	0.200E+02	0.000E+00
18	0	0.130E-05	0.200E+02	0.000E+00
19	0	0.100E-06	0.200E+02	0.000E+00
20	0	0.300E-06	0.200E+02	0.000E+00

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Loss of Output Both Primary Pumps in a 1 hour window
 P1000 Loss of Primary Pump 1 and 2
 + 3 0 P1001 P1002 P4022
 P1001 Loss of Output from Primary Pump 1 then Pump 2
 * 2 0 P0400 P0500
 P1002 Common Cause Loss of Primary Pump 1 and 2
 + 0 7 JSG212VU PCB012PU PPM012RU PSW012CU PSW012PU PDC012PU PWR012PU
 P4022 CPC-2 Common Cause Failures.
 * 0 2 PWR0222PU PWR032QU
 P0400 Loss of Output from Pump 1
 + 2 0 P0002 P0202
 P0500 Loss of Output from Pump 2
 + 2 0 P0102 P0302
 P0002 Loss of Electrical Power to Primary Pump 1
 + 3 0 P0010 P0020 P0040
 P0202 Mechanical Failure of Pump 1
 + 1 1 P0203 RMO001RR
 P0102 Loss of Electrical Power to Primary Pump 2
 + 3 0 P0110 P0120 P0140
 P0302 Mechanical Failure of Pump 2
 + 1 1 P0303 RMO002RR
 P0010 Loss of MG Set 1 Motor Output
 + 1 1 P0011 PMO001PW
 P0020 Loss of MG Set 1 Clutch Coupling
 + 1 1 P0021 PCL001PW
 P0040 Loss of MG Set 1 Generator Output
 + 2 1 P0041 P0051 PGE001PW
 P0203 Pump 1 Shaft Failure
 + 0 2 PPM001RG PPM001RR
 P0110 Loss of MG Set 2 Motor Output
 + 1 1 P0111 PMO002PW
 P0120 Loss of MG Set 2 Clutch Coupling
 + 1 1 P0121 PCL002PW
 P0140 Loss of MG Set 2 Generator Output
 + 2 1 P0141 P0151 PGE002PW
 P0303 Pump 2 Shaft Failure
 + 0 2 PPM002RG PPM002RR
 P0011 Loss of 2400 V Supply to MG Set 1 Motor
 + 0 4 J0301 JSG201VO PWR201YB PWR201YC
 P0021 Loss of Output from HP Power Supply No. 1
 + 2 2 P0022 P0035 PDC001PB PSC001PB
 P0041 Loss of MG Set 1 Generator Exciter Current
 + 2 2 P0042 P0043 PEX001PW PDC006PB
 P0051 MG Set 1 Output Wiring to Pump 1 Failure
 + 0 3 PCB001PO PWR051TB PWR051TC
 P0111 Loss of 2400 V Supply to MG Set 2 Motor
 + 0 4 J0301 JSG202VO PWR202YB PWR202YC
 P0121 Loss of Output from HP Power Supply No. 2
 + 2 2 P0122 P0135 PDC002PB PSC002PB
 P0141 Loss of MG Set 2 Generator Exciter Current
 + 2 2 P0142 P0143 PEX002PW PDC007PB
 P0151 MG Set 2 Output Wiring to Pump 2 Failure
 + 0 3 PCB002PO PWR151TB PWR151TC
 P0022 Loss of Power to HP Power Supply No. 1
 + 4 2 P0023 P0024 P0025 P0028 PKS001PB PWR01APB
 P0035 Controller Module No. 1 Signal Error
 + 0 3 PSC01BPW PST001PW PSC01CPW
 P0042 Exciter to MG Set 1 Generator Wiring Failure
 + 0 2 PWR042PB PWR042PC

P0043	Loss of Power to MG Set 1 Exciter Power Supply
	+ 1 2 P0044 U0050 UCB125QO
P0122	Loss of Power to HP Power Supply No. 2
	+ 3 1 P0123 P0124 P0125 PWR01BPB
P0135	Controller Module No. 2 Signal Error
	+ 0 3 PSC02BPW PST002PW PSC02CPW
P0142	Exciter to MG Set 2 Generator Wiring Failure
	+ 0 2 PWR142PB PWR142PC
P0143	Loss of Power to MG Set 2 Exciter Power Supply
	+ 1 2 P0144 U0150 UCB217QO
P0023	Loss of Constant Power Panel 1 Supply
	+ 1 4 P0029 U0050 UCB125QO PWR022YB PWR022YC
P0024	Primary Pump START/STOP Control Failure
	* 2 0 P0026 P0027
P0025	Relay EX1 Failure
	+ 0 2 PREX1PB PCZX15PB
P0028	Relay PBS Failure
	+ 0 3 PREPBSPB PCZPB1PB PCZPB8PB
P0044	CPS to MG Set 1 Exciter Wiring Failure
	+ 0 2 PWR044PB PWR044PC
P0123	Loss of Constant Power Panel 2 Supply
	+ 1 4 P0129 U0150 UCB217QO PWR122YB PWR122YC
P0124	Primary Pump START/STOP Control Failure
	* 1 2 P0126 PCZCP7QB PWR03BQB
P0125	Relay EX2 Failure
	+ 0 3 PREX2PB PCZX24PB PCZX25PB
P0144	CPS to MG Set 2 Exciter Wiring Failure
	+ 0 2 PWR144PB PWR144PC
P0029	Loss of CPS, STOP Switch Short Circuit, Pump 1
	+ 0 2 PSW01APD PSW02ACD
P0026	Primary Pump STOP Switches or Wiring Failure
	+ 0 3 PSW01APB PSW02ACB PWR02APB
P0027	Relay CPC-2 Failure
	+ 0 3 PREP2AQB PCZCP6QB PWR03AQB
P0129	Loss of CPS, STOP Switch Short Circuit, Pump 2
	+ 0 2 PSW01BPD PSW02BCD
P0126	Primary Pump STOP Switches or Wiring Failure
	+ 0 3 PSW01BPB PSW02BCB PWR02BPB
J0301	3 0 1 Loss of 2400 V Bus Power
U0050	3 0 1 LOSS OF 120 VAC CONTINUOUS POWER PANEL No.1
U0150	3 0 1 LOSS OF 120 VAC CONTINUOUS POWER PANEL No.2
JSG212VU	0 0 2 MG Set 1 and 2 Input Breaker Fails Open, Common Cause
PCB012PU	0 0 3 MG Set 1 and 2 Output Breaker Fails Open, Common Cause
PPM012RU	0 0 4 Primary Pump 1 and 2 Shaft Binding, Common Cause
PSW012CU	0 0 5 Control Room STOP Switch Short, Common Cause Loss of CPS
PSW012PU	0 0 6 Primary Panel STOP Switch Short, Common Cause Loss of CPS
PDC012PU	0 0 7 HP Power Supply No. 1 and 2 Fails, Common Cause
PWR012PU	0 0 8 Pumps 1 & 2 START/STOP Circuit Wiring Fail Open Common Cause
PWR022PU	0 0 9 Wiring between STOP Switches Fails Open, Pumps 1 & 2 Common
PWR032QU	0 0 10 Wiring to/from CPC-2 Relay Fails Open, Pumps 1 & 2 Common
RMO001RR	0 0 11 Primary Pump 1 Motor Fails to Run
RMO002RR	0 0 12 Primary Pump 2 Motor Fails to Run
PMO001PW	0 0 13 MG Set 1 Motor Failure
PCL001PW	0 0 14 MG Set 1 Clutch Failure
PGE001PW	0 0 15 MG Set 1 Generator Failure
PPM001RG	0 0 16 Primary Pump 1 Shaft Binding
PPM001RR	0 0 17 Primary Pump 1 Shaft Broken or Seized
PMO002PW	0 0 18 MG Set 2 Motor Failure
PCL002PW	0 0 19 MG Set 2 Clutch Failure
PGE002PW	0 0 20 MG Set 2 Generator Failure

PPM002RG	0	0	21 Primary Pump 2 Shaft Binding
PPM002RR	0	0	22 Primary Pump 2 Shaft Broken or Seized
JSG201VO	0	0	23 MG Set 1 Input Breaker Fails Open
PWR201YB	0	0	24 2400 V Feeder to MG Set 1 Fails Open
PWR201YC	0	0	25 2400 V Feeder to MG Set 1 Short Circuit
PDC001PB	0	0	26 HP Power Supply No. 1 Failure
PSC001PB	0	0	27 MG Set Controller 1 Overspeed Cutoff
PEX001PW	0	0	28 MG Set 1 Exciter Failure
PDC006PB	0	0	29 Exciter No. 1 Power Supply Failure
PCB001PO	0	0	30 MG Set 1 Output Breaker Fails Open
PWR051TB	0	0	31 MG Set 1 Wiring to Pump 1 Fails Open
PWR051TC	0	0	32 MG Set 1 Wiring to Pump 1 Short Circuit
JSG202VO	0	0	33 MG Set 2 Input Breaker Fails Open
PWR202YB	0	0	34 2400 V Feeder to MG Set 2 Fails Open
PWR202YC	0	0	35 2400 V Feeder to MG Set 2 Short Circuit
PDC002PB	0	0	36 HP Power Supply No. 2 Failure
PSC002PB	0	0	37 MG Set Controller 2 Overspeed Cutoff
PEX002PW	0	0	38 MG Set 2 Exciter Failure
PDC007PB	0	0	39 Exciter No. 2 Power Supply Failure
PCB002PO	0	0	40 MG Set 2 Output Breaker Fails Open
PWR151TB	0	0	41 MG Set 2 Wiring to Pump 2 Fails Open
PWR151TC	0	0	42 MG Set 2 Wiring to Pump 2 Short Circuit
PKS001PB	0	0	43 Primary Pump Key Switch Fails Open
PWR01APB	0	0	44 Primary Pump 1 START/STOP Circuit Wiring Fails Open
PSC01BPW	0	0	45 MG Set 1 Controller Faulty Demand Signal
PST001PW	0	0	46 No. 1 Feedback Signal Circuit Fails With High Signal
PSC01CPW	0	0	47 DCS No. 1 Fails With Low Demand Signal
PWR042PB	0	0	48 Exciter No. 1 Wiring to Generator Fails Open
PWR042PC	0	0	49 Exciter No. 1 Wiring to Generator Short Circuit
UCB125QO	0	0	50 CPS Panel 1 Breaker 25 Fails Open
PWR01BPB	0	0	51 Primary Pump 2 START/STOP Circuit Wiring Fails Open
PSC02BPW	0	0	52 MG Set 2 Controller Faulty Demand Signal
PST002PW	0	0	53 No. 2 Feedback Signal Circuit Fails With High Signal
PSC02CPW	0	0	54 DCS No. 2 Fails With Low Demand Signal
PWR142PB	0	0	55 Exciter No. 2 Wiring to Generator Fails Open
PWR142PC	0	0	56 Exciter No. 2 Wiring to Generator Short Circuit
UCB217QO	0	0	57 CPS Panel 2 Breaker 17 Fails Open
PWR022YB	0	0	58 CPS Feeder to HP Power Supply No. 1 Fails Open
PWR022YC	0	0	59 CPS Feeder to HP Power Supply No. 1 Short Circuit
PREEX1PB	0	0	60 Relay EX1 Coil Fails Open
PCZX15PB	0	0	61 Contacts 5 of Relay EX1 Fail Open
PREPBSPB	0	0	62 Relay PBS Coil Fails Open
PCZPB1PB	0	0	63 Contacts 1 of Relay PBS Fail Open
PCZPB8PB	0	0	64 Contacts 8 of Relay PBS Fail Open
PWR044PB	0	0	65 CPS Feeder to Exciter No. 1 PS Fails Open
PWR044PC	0	0	66 CPS Feeder to Exciter No. 1 PS Short Circuit
PWR122YB	0	0	67 CPS Feeder to HP Power Supply No. 2 Fails Open
PWR122YC	0	0	68 CPS Feeder to HP Power Supply No. 2 Short Circuit
PCZCP7QB	0	0	69 Contacts 7 of Relay CPC-2 Fail Open
PWR03BQB	0	0	70 Wiring to/from CPC-2 Relay Fails Open, Pump 2
PREEX2PB	0	0	71 Relay EX2 Coil Fails Open
PCZX24PB	0	0	72 Contacts 4 of Relay EX2 Fail Open
PCZX25PB	0	0	73 Contacts 5 of Relay EX2 Fail Open
PWR144PB	0	0	74 CPS Feeder to Exciter No. 2 PS Fails Open
PWR144PC	0	0	75 CPS Feeder to Exciter No. 2 PS Short Circuit
PSW01APD	0	0	76 Primary Panel STOP Switch Short, Loss of CPS, Pump 1
PSW02ACD	0	0	77 Control Room STOP Switch Short, Loss of CPS, Pump 1
PWR01APB	0	0	78 Primary Panel STOP Switch Fails Open, Pump 1
PSW02ACB	0	0	79 Control Room STOP Switch Fails Open, Pump 1
PWR02APB	0	0	80 Wiring between STOP Switches Fail, Pump 1

PREP2AQB 0 0 81 Relay CPC-2 Coil Fails Open
 PCZCP6QB 0 0 82 Contacts 6 of Relay CPC-2 Fail Open
 PRK03AQB 0 0 83 Wiring to/from CPC-2 Relay Fails Open, Pump 1
 PSW01BPD 0 0 84 Primary Panel STOP Switch Short, Loss of CPS, Pump 2
 PSW02BCD 0 0 85 Control Room STOP Switch Short, Loss of CPS, Pump 2
 PSW01BPB 0 0 86 Primary Panel STOP Switch Fails Open, Pump 2
 PSW02BCB 0 0 87 Control Room STOP Switch Fails Open, Pump 2
 PRK02BPB 0 0 88 Wiring between STOP Switches Fails Open, Pump 2

0.100E+01
 0.100E+01

1 0 0.100E-09 0.100E+02 0.000E+00

2 2 0.300E-07 0.100E+02

3 2 0.300E-07 0.100E+02

4 2 0.290E-05 0.100E+02

5 2 0.100E-07 0.100E+02

6 2 0.100E-07 0.100E+02

7 2 0.300E-06 0.100E+02

8 2 0.300E-06 0.100E+02

9 2 0.300E-06 0.100E+02

10 2 0.600E-06 0.100E+02

11 2 0.200E-04 0.300E+01

12 1 0.200E-04 0.300E+01

13 2 0.300E-04 0.300E+01

14 2 0.500E-04 0.300E+02

15 2 0.300E-05 0.100E+02

16 2 0.290E-04 0.260E+01

17 2 0.190E-05 0.840E+01

18 1 0.300E-04 0.300E+01

19 1 0.500E-04 0.300E+02

20 1 0.300E-05 0.100E+02

21 1 0.290E-04 0.260E+01

22 1 0.190E-05 0.840E+01

23 2 0.300E-06 0.100E+02

24 2 0.300E-05 0.100E+02

25 2 0.300E-06 0.100E+02

26 2 0.300E-05 0.100E+02

27 2 0.300E-05 0.100E+02

28 2 0.300E-05 0.100E+02

29 2 0.300E-05 0.100E+02

30 2 0.300E-06 0.100E+02

31 2 0.300E-05 0.100E+02

32 2 0.300E-06 0.100E+02

33 1 0.300E-06 0.100E+02

34 1 0.300E-05 0.100E+02

35 1 0.300E-06 0.100E+02

36 1 0.300E-05 0.100E+02

37 1 0.300E-05 0.100E+02

38 1 0.300E-05 0.100E+02

39 1 0.300E-05 0.100E+02

40 1 0.300E-06 0.100E+02

41 1 0.300E-05 0.100E+02

42 1 0.300E-06 0.100E+02

43 2 0.100E-06 0.100E+02

44 2 0.300E-05 0.100E+02

45 2 0.300E-05 0.100E+02

46 2 0.300E-05 0.100E+02

47 2 0.300E-05 0.100E+02

48 2 0.300E-05 0.100E+02

49 2 0.300E-06 0.100E+02

50 2 0.300E-06 0.100E+02

51	1	0.300E-05	0.100E+02
52	1	0.300E-05	0.100E+02
53	1	0.300E-05	0.100E+02
54	1	0.300E-05	0.100E+02
55	1	0.300E-05	0.100E+02
56	1	0.300E-06	0.100E+02
57	1	0.300E-06	0.100E+02
58	2	0.300E-05	0.100E+02
59	2	0.300E-06	0.100E+02
60	2	0.500E-06	0.100E+02
61	2	0.100E-06	0.100E+02
62	2	0.500E-06	0.100E+02
63	2	0.100E-06	0.100E+02
64	2	0.100E-06	0.100E+02
65	2	0.300E-05	0.100E+02
66	2	0.300E-06	0.100E+02
67	1	0.300E-05	0.100E+02
68	1	0.300E-06	0.100E+02
69	2	0.100E-06	0.100E+02
70	2	0.600E-05	0.100E+02
71	1	0.500E-06	0.100E+02
72	1	0.100E-06	0.100E+02
73	1	0.100E-06	0.100E+02
74	1	0.300E-05	0.100E+02
75	1	0.300E-06	0.100E+02
76	2	0.100E-06	0.100E+02
77	2	0.100E-06	0.100E+02
78	2	0.100E-06	0.100E+02
79	2	0.100E-06	0.100E+02
80	2	0.300E-05	0.100E+02
81	2	0.500E-06	0.100E+02
82	2	0.100E-06	0.100E+02
83	2	0.600E-05	0.100E+02
84	2	0.100E-06	0.100E+02
85	2	0.100E-06	0.100E+02
86	2	0.100E-06	0.100E+02
87	2	0.100E-06	0.100E+02
88	2	0.300E-05	0.100E+02

Plant Operating on "Hotel" Load	
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1 1 1
Plant Operating on "Hotel" Load
P7777 PRIMARY PUMPS NOT RUNNING: HOTEL LOAD LOSS OF BOP
+ 0 1 B0099ZUW
B0099ZUW 0 0 1 Operating on Hotel Mode
0.100E+01
0.100E+01
1 0 0.200E-02 0.100E+02 0.000E+00

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B.2-250

Balance of Plant (Sec Na and Steam)

Fri Jun 14 14:03:43 1991

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B.2-251

19 15 12

Balance of Plant (Sec Na and Steam)			NOT AVAILABLE FOR >8 HR	IN FIRST 7 DAYS
B1000L	BOP DKHR	+ 2 0 B2000L	B7000L	
B2000L	STEAM PLANT	+ 2 1 B3000L	B5000L	REMOVAL NOT AVAILABLE >8 HR
B7000L	SECONDARY Na	+ 1 1 B7100L	NSW000CZ	REMOVAL NOT AVAILABLE >8 HR
B3000L	FEEDWATER NOT AVAILABLE DUE TO ELECTRICAL OR WATER SUPPLY	+ 3 0 B3100L	B3200L	B3300L
B5000L	NO STEAM PATH FOR CONDENSATION OR VENTING	+ 2 0 B5100L	B0100L	
B7100L	SECONDARY SODIUM DUMPED AND NOT RECOVERABLE	+ 0 2 NOOOOUE		
B3100L	BFP #1 or #2 NOT AVAILABLE DUE TO ELECTRICAL OR WATER SUP	+ 2 0 B4000L	B0100L	
B3200L	START UP FP NOT AVAILABLE DUE TO ELECTRICAL OR WATER SUP	+ 2 0 B4000L	B0555L	
B3300L	EMERGENCY FP NOT AVAILABLE DUE TO ELECTRICAL OR WATER SUP	+ 2 0 B3110L	B0120L	
B5100L	NO VENT PATH FOR STEAM	+ 0 2 VRV161YP	VRV161YQ	
B0100L	LOSS WITHOUT RECOVERY OF 480 V NORMAL POWER	+ 0 2 J0100L	30010200	
B4000L	CONDENSATE NOT AVAILABLE DUE TO ELECTRICAL OR WATER SUP	+ 2 0 B4100L	B4300L	
B0555L	LOSS WITHOUT RECOVERY OF 480 V NORMAL BUS P-2A	+ 0 2 J0555L	30055200	
B3110L	EMERGENCY FP NOT AVAILABLE DUE TO WATER SUPPLY	+ 2 0 B4000L	B6000L	
B0120L	LOSS WITHOUT RECOVERY OF 480 V EMERGENCY BUS2 P-4E	+ 0 2 E0120L	00012200	
B4100L	DUP COND PUMP ELEC 480E P1-B OR COND WATER SUPPLY ELEC 240	+ 2 0 B0200L	B0100L	
B4300L	COND PUMP #2 ELEC 480 P2-A OR COND WATER SUPPLY ELEC 240	+ 2 0 B0555L	B0100L	
B6000L	NO WATER SUPPLY TO CONDENSATE STORAGE TANK	+ 1 1 B0200L	VDL0000W	
B0200L	LOSS WITHOUT RECOVERY OF 480 V EMERGENCY BUS P-1B	+ 0 2 E0200L	00020200	
J0100L	3 0 1 Loss of 13.8 kV Bus Power			
J0555L	3 0 1 LOSS OF 480 VAC NORMAL BUS P-2A			
E0120L	3 0 1 LOSS OF 480 VAC EMERGENCY BUS P-4E			
E0200L	3 0 1 LOSS OF 480 VAC EMERGENCY BUS P-1B			
VOOOOOW	0 0 2 OTHER BOP FAILURES			
NSW000CZ	0 0 3 INADVERTANT OPERATOR ACTION TO DUMP			
NOOOOUE	0 0 4 SECONDARY SODIUM LEAK AND/OR FIRE			
VRV161YQ	0 0 5 SECONDARY SODIUM BLOCKAGE			
30010200	2 0 7 STEAM HEADER RELIEF DOES NOT OPEN			
30055200	2 0 8 NO RECOVERY OF LONP			
00012200	2 0 9 NO RECOVERY OF 480 V NORMAL BUS P-2A			
VDL0000W	0 0 10 NO RECOVERY OF 480 V EMERGENCY BUS P-4E			
00020200	2 0 11 DIESEL DRIVEN FIRE PUMPS NOT AVAILABLE			
0.336E+03	0 0 12 NO RECOVERY OF 480 V EMERGENCY BUS P-1B			
0.336E+03				
1 0	0.100E-09	0.100E+02	0.000E+00	
2 0	0.171E-01	0.100E+02		
3 0	0.100E-03	0.100E+02	0.000E+00	

4	0	0.202E-03	0.100E+02	
5	0	0.672E-04	0.100E+02	
6	0	0.300E-02	0.500E+01	0.000E+00
7	0	0.168E-02	0.100E+02	
8	T	0.100E+01	0.100E+01	
9	T	0.100E+01	0.100E+01	
10	T	0.100E+01	0.100E+01	
11	0	0.100E-01	0.500E+01	0.000E+00
12	T	0.100E+01	0.100E+01

Secondary Pump Fails to Trip	
Fri Jun 14 14:02:15 1991	
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1 1 1
Secondary Pump Fails to Trip
N7000 SECONDARY PUMP TRIP
      + 0 1 NPMOVCMN
NPMOVCMN 0 0 1 Probability of Secondary Pump Failing to Trip
0.100E+01
0.100E+01
1 0 0.100E-02 0.100E+02 0.000E+00

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B.2-256

Both Shutdown Coolers Undercooling not	
Recovered	
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B.2-257

Both Shutdown Coolers Undercooling not	
Recovered	
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B.2-259

Both Shutdown Coolers Undercooling not	
Recovered	
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Both Shutdown Coolers Undercooling not	
Recovered	
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B.2-261

Both Shutdown Coolers Undercooling not	
Recovered	
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B.2-262

Both Shutdown Coolers Undercooling not	
Recovered	
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Both Shutdown Coolers Undercooling not Recovered	
D1104L	BOTH SHUTDOWN COOLERS UNDERCOOLING NOT RECOVERED
D2103L	* 2 0 D2103L D2203L
D2203L	UNDERCOOLING SHUTDOWN COOLER #1 NOT RECOVERED
D2203L	+ 2 0 D2101L D2104L
D2203L	UNDERCOOLING SHUTDOWN COOLER #2 NOT RECOVERED
D2203L	+ 2 0 D2201L D2204L
D2101L	UNRECOVERABLE LOSS OF SHUTDOWN COOLER #1
D2101L	+ 0 2 9CR001OE 9CR001OF
D2104L	UNRECOVERED RECOVERABLE UNDERCOOLING COOLER #1
D2201L	* 1 1 D2102L 9HU0000L
D2201L	UNRECOVERABLE LOSS OF SHUTDOWN COOLER #2
D2201L	+ 0 2 9CR002OE 9CR002OF
D2204L	UNRECOVERED RECOVERABLE UNDERCOOLING COOLER #2
D2204L	* 1 1 D2202L 9HU0000L
D2102L	RECOVERABLE UNDERCOOLING SHUTDOWN COOLER #1
D2202L	+ 2 1 D2110L D2120L 9TW001OE
D2202L	RECOVERABLE UNDERCOOLING SHUTDOWN COOLER #2
D2202L	+ 2 1 D2210L D2220L 9TW002OE
D2110L	SHUTDOWN COOLER #1 UPPER DAMPERS FAIL TO OPEN
D2110L	+ 1 1 D2105L 9AT01UOP
D2120L	SHUTDOWN COOLER #1 LOWER DAMPERS FAIL TO OPEN
D2120L	+ 1 1 D2105L 9AT01LOP
D2210L	SHUTDOWN COOLER #2 UPPER DAMPERS FAIL TO OPEN
D2210L	+ 1 1 D2205L 9AT02UOP
D2220L	SHUTDOWN COOLER #2 LOWER DAMPERS FAIL TO OPEN
D2220L	+ 1 1 D2205L 9AT02LOP
D2105L	COOLER #1 AIR ACTUATORS REMAIN PRESSURIZED
D2205L	+ 1 1 D6811L 9SV01AON
D2205L	COOLER #2 AIR ACTUATORS REMAIN PRESSURIZED
D2205L	+ 1 1 D6811L 9SV01BON
D6811L	ELECT POWER NOT REMOVED FROM SOLENOIDS R4-VS-681A & B
D6811L	+ 1 1 D0531L 9RE53ACP
D0531L	ELECT POWER NOT REMOVED FROM RELAY ITEM 53A-PR2A
D0531L	+ 1 1 D0821L 9ARX82QP
D0821L	RONAN X82 ALARM PROCESSING ERROR P1-TA-504A
D0821L	+ 1 1 D0810L 9ARX82QW
D0810L	EXPANDED RANGE MV/I FALSE LOW ELEC OUTPUT P1-TMV/I-504B
D0810L	+ 1 1 D5040L 9AM4BLQW
D5040L	AV BULK NA MV/I FALSE LOW ELEC OUTPUT P1-TMV/I-504A
D5040L	+ 1 1 D5400L 9AM4ALAW
D5400L	SENSOR MALFUNCTION THERMOCOUPLE LOW OUTPUT
D5400L	+ 2 0 D5401L D5404L
D5401L	CONTINUING SENSOR MALFUNCTION T/C SHORTED
D5401L	* 2 0 D5402L D5403L
D5404L	CONTINUING T/C REF JUNCT OVEN MALFUNCTION
D5404L	+ 5 0 D0501L D0502L D0504L D0505L D0751L
D5402L	SENSOR MALFUNCTION THERMOCOUPLE SHORTED
D5403L	+ 0 5 9TC40HRD 9TC40KRD 9TC40MRD 9TC40NRD 9TC40PRD
D5403L	NO ALARM OR OPERATOR ACTION FOR SHORTED T/C CONDITION
D5403L	+ 0 2 9AN82LAW 9HU82LCZ
D0501L	CONTINUING T/C REF JUNCT OVEN MALFUNCTION R1-CRJ-50-1
D0501L	* 1 1 D1501L 9HR501AW
D0502L	CONTINUING T/C REF JUNCT OVEN MALFUNCTION R1-CRJ-50-2
D0502L	* 1 1 D1502L 9HR502AW
D0504L	CONTINUING T/C REF JUNCT OVEN MALFUNCTION R1-CRJ-50-4
D0504L	* 1 1 D1504L 9HR504AW
D0505L	CONTINUING T/C REF JUNCT OVEN MALFUNCTION R1-CRJ-50-5
D0505L	* 1 1 D1505L 9HR505AW

D0751L	CONTINUING	T/C REF	JUNCT	OVEN	MALEFUNCTION	R1-CRJ-75-1
D1501L	* 1 1 D1751L	9HR751AW				
D1501L	NO ALARM OR	OPERATOR	ACTION	OVEN	TEMP HIGH	R1-CRJ-50-1
D1502L	+ 0 2 9AN501AW	9HU500CZ				
D1502L	NO ALARM OR	OPERATOR	ACTION	OVEN	TEMP HIGH	R1-CRJ-50-2
D1504L	+ 0 2 9AN502AW	9HU500CZ				
D1504L	NO ALARM OR	OPERATOR	ACTION	OVEN	TEMP HIGH	R1-CRJ-50-4
D1505L	+ 0 2 9AN504AW	9HU500CZ				
D1505L	NO ALARM OR	OPERATOR	ACTION	OVEN	TEMP HIGH	R1-CRJ-50-5
D1751L	+ 0 2 9AN505AW	9HU500CZ				
D1751L	NO ALARM OR	OPERATOR	ACTION	OVEN	TEMP HIGH	R1-CRJ-75-1
9CR001OE	+ 0 2 9AN751AW	9HU500CZ				
9CR001OE	0 0	1 COOLER #1	NAK	BLOCKAGE		
9CR001OF	0 0	2 COOLER #1	NAK	LEAKAGE		
9HU000OL	0 0	3 HUMAN ERROR	-	FAILURE TO RECOVER COOLERS		
9CR002OE	0 0	4 COOLER #2	NAK	BLOCKAGE		
9CR002OF	0 0	5 COOLER #2	NAK	LEAKAGE		
9TW001OE	0 0	6 COOLER #1	CHIMNEY	BLOCKED		
9TW002OE	0 0	7 COOLER #2	CHIMNEY	BLOCKED		
9AT01UOP	0 0	8 COOLER 1	UPPER ACTUATOR OR	LINKAGE FAILS	TO OPEN DAMPER	
9AT01LOP	0 0	9 COOLER 1	LOWER ACTUATOR OR	LINKAGE FAILS	TO OPEN DAMPER	
9AT02UOP	0 0	10 COOLER 2	UPPER ACTUATOR OR	LINKAGE FAILS	TO OPEN DAMPER	
9AT02LOP	0 0	11 COOLER 2	LOWER ACTUATOR OR	LINKAGE FAILS	TO OPEN DAMPER	
9SV01AON	0 0	12 SOLENOID	COOLER #1	FAILS TO CLOSE	R4-VS-681A	
9SV01BON	0 0	13 SOLENOID	COOLER #2	FAILS TO CLOSE	R4-VS-681B	
9RES3ACP	0 0	14 AUX ANN RELAY	PT 2 (CONTACT)	FAILS TO OPEN	ITEM 53A-PR2A	
9ARX82QP	0 0	15 RONAN X82	ALARM RELAY	FAILS TO OPEN	P1-TA-504A	
9ARX82QM	0 0	16 RONAN X82	ALARM CIRCUIT	FAILS LOW	P1-TA-504A	
9AM4ALAW	0 0	17 EXPN RANGE	TEMP RONAN X81MV/I	FAILS LOW ELEC	P1-TMV/I-504B	
9TC40HRD	0 0	18 AV BULK NA	TEMP RONAN X81MV/I	FAILS LOW ELEC	P1-TMV/I-504A	
9TC40HRD	0 0	19 BULK SODIUM	THERMOCOUPLE	SHORTED	RI-TC-540H	
9TC40KRD	0 0	20 BULK SODIUM	THERMOCOUPLE	SHORTED	RI-TC-540K	
9TC40NRD	0 0	21 BULK SODIUM	THERMOCOUPLE	SHORTED	RI-TC-540M	
9TC40PRD	0 0	22 BULK SODIUM	THERMOCOUPLE	SHORTED	RI-TC-540N	
9AN821AW	0 0	23 BULK SODIUM	THERMOCOUPLE	SHORTED	RI-TC-540P	
9HUB21CZ	0 0	24 BULK SODIUM	LOW TEMPERATURE	ANNUN FAILS		
9HR501AW	0 0	25 HUMAN ERROR	BULK SODIUM	LOW TEMP	ALARM IGNORED	
9HR502AW	0 0	26 T/C REFERENCE	JUNCTION OVEN	FAILS HIGH	R1-CRJ-50-1	
9HR504AW	0 0	27 T/C REFERENCE	JUNCTION OVEN	FAILS HIGH	R1-CRJ-50-2	
9HR505AW	0 0	28 T/C REFERENCE	JUNCTION OVEN	FAILS HIGH	R1-CRJ-50-4	
9HR751AW	0 0	29 T/C REFERENCE	JUNCTION OVEN	FAILS HIGH	R1-CRJ-50-5	
9HU500CZ	0 0	30 T/C REFERENCE	JUNCTION OVEN	FAILS HIGH	R1-CRJ-75-1	
9AN502AW	0 0	31 T/C REFERENCE	JUNCTION HIGH	ANNUN FAILS	FOR R1-CRJ-50-1	
9AN504AW	0 0	32 HUMAN ERROR	ANY T/C REF	JUNCT >152F	ALARM IGNORED	
9AN505AW	0 0	33 T/C REFERENCE	JUNCTION HIGH	ANNUN FAILS	FOR R1-CRJ-50-2	
9AN751AW	0 0	34 T/C REFERENCE	JUNCTION HIGH	ANNUN FAILS	FOR R1-CRJ-50-4	
	0 0	35 T/C REFERENCE	JUNCTION HIGH	ANNUN FAILS	FOR R1-CRJ-50-5	
	0 0	36 T/C REFERENCE	JUNCTION HIGH	ANNUN FAILS	FOR R1-CRJ-75-1	
	0.336E+03					
	0.336E+03					
	1 0 0.336E-04	0.100E+02				
	2 0 0.101E-03	0.100E+02				
	3 0 0.400E-05	0.100E+02	0.000E+00			
	4 0 0.336E-04	0.100E+02				
	5 0 0.101E-03	0.100E+02				
	6 0 0.370E-03	0.100E+02				
	7 0 0.370E-03	0.100E+02				
	8 0 0.300E-02	0.500E+01	0.000E+00			
	9 0 0.300E-02	0.500E+01	0.000E+00			
	10 0 0.300E-02	0.500E+01	0.000E+00			

11	0	0.300E-02	0.500E+01	0.000E+00
12	0	0.300E-02	0.500E+01	0.000E+00
13	0	0.300E-02	0.500E+01	0.000E+00
14	0	0.300E-03	0.100E+02	0.000E+00
15	0	0.300E-03	0.100E+02	0.000E+00
16	0	0.101E-02	0.100E+02	
17	0	0.101E-02	0.100E+02	
18	0	0.101E-02	0.100E+02	
19	0	0.336E-03	0.100E+02	
20	0	0.336E-03	0.100E+02	
21	0	0.336E-03	0.100E+02	
22	0	0.336E-03	0.100E+02	
23	0	0.336E-03	0.100E+02	
24	0	0.101E-02	0.100E+02	
25	0	0.100E-01	0.100E+02	0.000E+00
26	0	0.101E-02	0.100E+02	
27	0	0.101E-02	0.100E+02	
28	0	0.101E-02	0.100E+02	
29	0	0.101E-02	0.100E+02	
30	0	0.101E-02	0.100E+02	
31	0	0.101E-02	0.100E+02	
32	0	0.100E-01	0.100E+02	0.000E+00
33	0	0.101E-02	0.100E+02	
34	0	0.101E-02	0.100E+02	
35	0	0.101E-02	0.100E+02	
36	0	0.101E-02	0.100E+02

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Both Shutdown Coolers Undercooling not	
Recovered	
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Both Shutdown Coolers Undercooling not	
Recovered	
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Both Shutdown Coolers Undercooling not	
Recovered	
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Both Shutdown Coolers Undercooling not	
Recovered	
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Both Shutdown Coolers Undercooling not Recovered

D1104	BOTH SHUTDOWN COOLERS UNDERCOOLING NOT RECOVERED	
D2103	* 2 0 D2103 D2203	
D2203	UNDERCOOLING SHUTDOWN COOLER #1 NOT RECOVERED	
D2203	+ 2 0 D2101 D2104	
D2203	UNDERCOOLING SHUTDOWN COOLER #2 NOT RECOVERED	
D2203	+ 2 0 D2201 D2204	
D2101	UNRECOVERABLE LOSS OF	SHUTDOWN COOLER #1
D2104	+ 0 2 DCR0010E DCR0010F	
D2201	UNRECOVERED RECOVERABLE UNDERCOOLING COOLER #1	
D2204	* 1 1 D2102 DHU0000L	
D2102	UNRECOVERABLE LOSS OF	SHUTDOWN COOLER #2
D2204	+ 0 2 DCR0020E DCR0020F	
D2102	UNRECOVERED RECOVERABLE UNDERCOOLING COOLER #2	
D2202	* 1 1 D2202 DHU0000L	
D2110	RECOVERABLE UNDERCOOLING SHUTDOWN COOLER #1	
D2210	+ 2 1 D2110 D2120 DTW0010E	
D2120	RECOVERABLE UNDERCOOLING SHUTDOWN COOLER #2	
D2220	+ 2 1 D2210 D2220 DTW0020E	
D2110	SHUTDOWN COOLER #1	UPPER DAMPERS FAIL TO OPEN
D2120	+ 1 1 D2105 DAT01UOP	
D2210	SHUTDOWN COOLER #1	LOWER DAMPERS FAIL TO OPEN
D2220	+ 1 1 D2105 DAT01LOP	
D2105	SHUTDOWN COOLER #2	UPPER DAMPERS FAIL TO OPEN
D2205	+ 1 1 D2205 DAT02UOP	
D2105	SHUTDOWN COOLER #2	LOWER DAMPERS FAIL TO OPEN
D2205	+ 1 1 D2205 DAT02LOP	
D2105	COOLER #1	AIR ACTUATORS REMAIN PRESSURIZED
D2205	+ 1 1 D6811 DSV01AON	
D6811	COOLER #2	AIR ACTUATORS REMAIN PRESSURIZED
D0531	+ 1 1 D6811 DSV01BON	
D0821	ELECT POWER NOT REMOVED	FROM SOLENOIDS R4-VS-681A & B
D0810	+ 1 1 D0531 DRES3ACP	
D5040	ELECT POWER NOT REMOVED	FROM RELAY ITEM 53A-PR2A
D5400	+ 1 1 D0821 DARY82QP	
D5401	RONAN X82 ALARM PROCESSING ERROR	P1-TA-504A
D5404	+ 1 1 D0810 DARY82QW	
D5400	EXPANDED RANGE MV/I FALSE LOW ELEC OUTPUT	P1-TMV/I-504B
D5401	+ 1 1 D5040 DAM4BLW	
D5404	AV BULK NA MV/I FALSE LOW	ELEC OUTPUT P1-TMV/I-504A
D5400	+ 1 1 D5400 DAM4ALAW	
D5401	SENSOR MALFUNCTION	THERMOCOUPLE LOW OUTPUT
D5404	+ 2 0 D5401 D5404	
D5400	CONTINUING SENSOR	MALFUNCTION T/C SHORTED
D5404	* 2 0 D5402 D5403	
D5402	CONTINUING T/C REF JUNCT OVEN	MALFUNCTION D0751
D5403	+ 5 0 D0501 D0502 D0504 D0505	
D0501	SENSOR MALFUNCTION	THERMOCOUPLE SHORTED
D0502	+ 0 5 DTC40HRD DTC40MRD DTC40NRD DTC40PRD	
D0504	NO ALARM OR OPERATOR ACTION FOR SHORTED	T/C CONDITION
D0505	+ 0 2 DAN82LAW DHU82LCZ	
D0501	CONTINUING T/C REF JUNCT OVEN MALFUNCTION R1-CRJ-50-1	
D0502	* 1 1 D1501 DHR501AW	
D0504	CONTINUING T/C REF JUNCT OVEN MALFUNCTION R1-CRJ-50-2	
D0505	* 1 1 D1502 DHR502AW	
D0501	CONTINUING T/C REF JUNCT OVEN MALFUNCTION R1-CRJ-50-4	
D0502	* 1 1 D1504 DHR504AW	
D0504	CONTINUING T/C REF JUNCT OVEN MALFUNCTION R1-CRJ-50-5	
D0505	* 1 1 D1505 DHR505AW	

D0751	CONTINUING	T/C REF	JUNCT	OVEN	MALEFUNCTION	R1-CRJ-75-1
D1501	* 1 1 D1751	DHR751AW	OPERATOR	ACTION	OVEN TEMP HIGH	R1-CRJ-50-1
D1502	+ 0 2 DAN501AW	DHU500CZ	OPERATOR	ACTION	OVEN TEMP HIGH	R1-CRJ-50-2
D1504	+ 0 2 DAN502AW	DHU500CZ	OPERATOR	ACTION	OVEN TEMP HIGH	R1-CRJ-50-4
D1505	+ 0 2 DAN504AW	DHU500CZ	OPERATOR	ACTION	OVEN TEMP HIGH	R1-CRJ-50-5
D1751	+ 0 2 DAN505AW	DHU500CZ	OPERATOR	ACTION	OVEN TEMP HIGH	R1-CRJ-75-1
DCR0010E	0 0 1 COOLER #1	NAK BLOCKAGE				
DCR0010F	0 0 2 COOLER #1	NAK LEAKAGE				
DHU0000L	0 0 3 HUMAN ERROR	- FAILURE TO RECOVER COOLERS				
DCR0020E	0 0 4 COOLER #2	NAK BLOCKAGE				
DCR0020F	0 0 5 COOLER #2	NAK LEAKAGE				
DTW0010E	0 0 6 COOLER #1	CHIMNEY				
DTW0020E	0 0 7 COOLER #2	CHIMNEY				
DAT01UOP	0 0 8 COOLER 1 UPPER ACTUATOR	OR				
DAT01LOP	0 0 9 COOLER 1 LOWER ACTUATOR	OR				
DAT02UOP	0 0 10 COOLER 2 UPPER ACTUATOR	OR				
DAT02LOP	0 0 11 COOLER 2 LOWER ACTUATOR	OR				
DSV01AON	0 0 12 SOLENOID	COOLER #1				
DSV01BON	0 0 13 SOLENOID	COOLER #2				
DRES3ACP	0 0 14 AUX ANN RELAY	PT 2 (CONTACT)				
DARX82QP	0 0 15 RONAN X82	ALARM RELAY				
DARX82QW	0 0 16 RONAN X82	ALARM CIRCUIT				
DAM4BLQW	0 0 17 EXPN RANGE TEMP	RONAN X81NV/I				
DAM4ALAW	0 0 18 AV BULK NA TEMP	RONAN X81NV/I				
DTC40HRD	0 0 19 BULK SODIUM	THERMOCOUPLE				
DTC40KRD	0 0 20 BULK SODIUM	THERMOCOUPLE				
DTC40MRD	0 0 21 BULK SODIUM	THERMOCOUPLE				
DTC40NRD	0 0 22 BULK SODIUM	THERMOCOUPLE				
DTC40PRD	0 0 23 BULK SODIUM	THERMOCOUPLE				
DAN82LAW	0 0 24 BULK SODIUM	LOW TEMPERATURE				
DHU82LCZ	0 0 25 HUMAN ERROR	BULK SODIUM				
DHR501AW	0 0 26 T/C REFERENCE	JUNCTION OVEN				
DHR502AW	0 0 27 T/C REFERENCE	JUNCTION OVEN				
DHR504AW	0 0 28 T/C REFERENCE	JUNCTION OVEN				
DHR505AW	0 0 29 T/C REFERENCE	JUNCTION OVEN				
DHR751AW	0 0 30 T/C REFERENCE	JUNCTION HIGH				
DAN501AW	0 0 31 T/C REFERENCE	JUNCTION HIGH				
DHU500CZ	0 0 32 HUMAN ERROR	ANY T/C REF				
DAN502AW	0 0 33 T/C REFERENCE	JUNCTION HIGH				
DAN504AW	0 0 34 T/C REFERENCE	JUNCTION HIGH				
DAN505AW	0 0 35 T/C REFERENCE	JUNCTION HIGH				
DAN751AW	0 0 36 T/C REFERENCE	JUNCTION HIGH				

0.744E+03						
0.744E+03						
1 2	0.100E-06	0.100E+02				
2 2	0.300E-06	0.100E+02				
3 0	0.400E-05	0.100E+02	0.000E+00			
4 2	0.100E-06	0.100E+02				
5 2	0.300E-06	0.100E+02				
6 2	0.110E-05	0.100E+02				
7 2	0.110E-05	0.100E+02				
8 0	0.300E-02	0.500E+01	0.000E+00			
9 0	0.300E-02	0.500E+01	0.000E+00			
10 0	0.300E-02	0.500E+01	0.000E+00			

11	0	0.300E-02	0.500E+01	0.000E+00
12	0	0.300E-02	0.500E+01	0.000E+00
13	0	0.300E-02	0.500E+01	0.000E+00
14	0	0.300E-03	0.100E+02	0.000E+00
15	0	0.300E-03	0.100E+02	0.000E+00
16	1	0.300E-05	0.100E+02	
17	1	0.300E-05	0.100E+02	
18	1	0.300E-05	0.100E+02	
19	1	0.100E-05	0.100E+02	
20	1	0.100E-05	0.100E+02	
21	1	0.100E-05	0.100E+02	
22	1	0.100E-05	0.100E+02	
23	1	0.100E-05	0.100E+02	
24	1	0.300E-05	0.100E+02	
25	0	0.100E-01	0.100E+02	0.000E+00
26	1	0.300E-05	0.100E+02	
27	1	0.300E-05	0.100E+02	
28	1	0.300E-05	0.100E+02	
29	1	0.300E-05	0.100E+02	
30	1	0.300E-05	0.100E+02	
31	1	0.300E-05	0.100E+02	
32	0	0.100E-01	0.100E+02	0.000E+00
33	1	0.300E-05	0.100E+02	
34	1	0.300E-05	0.100E+02	
35	1	0.300E-05	0.100E+02	
36	1	0.300E-05	0.100E+02

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Long Term Decay Heat Removal; S/D Coolers;

Shield and Thimble Cooling

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Long Term Decay Heat Removal; S/D Coolers;

Shield and Thimble Cooling

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Long Term Decay Heat Removal; S/D Coolers; Shield and Thimble Cooling
D5000 LONG TERM DECAY HEAT REMOVAL NOT AVAILABLE
* 2 1 D2103 D3020 B0002Z00
D2103 UNDERCOOLING SHUTDOWN COOLER #1 NOT RECOVERED
+ 2 0 D2101 D2104
D3020 SHIELD COOLING OR THIMBLE COOLING NOT AVAILABLE
+ 0 2 H1000 T1000
D2101 UNRECOVERABLE LOSS OF SHUTDOWN COOLER #1
+ 0 2 DCR0010E DCR0010F
D2104 UNRECOVERED RECOVERABLE UNDERCOOLING COOLER #1
* 1 1 D2102 DHU0000L
D2102 RECOVERABLE UNDERCOOLING SHUTDOWN COOLER #1
+ 2 1 D2110 D2120 DTW0010E
D2110 SHUTDOWN COOLER #1 UPPER DAMPERS FAIL TO OPEN
+ 1 1 D2105 DAT01UOP
D2120 SHUTDOWN COOLER #1 LOWER DAMPERS FAIL TO OPEN
+ 1 1 D2105 DAT01LOP
D2105 COOLER #1 AIR ACTUATORS REMAIN PRESSURIZED
+ 1 1 D6811 DSV01AON
D6811 ELECT POWER NOT REMOVED FROM SOLENOIDS R4-VS-681A & B
+ 1 1 D0531 DRE53ACP
D0531 ELECT POWER NOT REMOVED FROM RELAY ITEM 53A-PR2A
+ 1 1 D0821 DARX82QP
D0821 RONAN X82 ALARM PROCESSING ERROR P1-TA-504A
+ 1 1 D0810 DARX82QW
D0810 EXPANDED RANGE MV/I FALSE LOW ELEC OUTPUT P1-TMV/I-504B
+ 1 1 D5040 DAM4BLQW
D5040 AV BULK NA MV/I FALSE LOW ELEC OUTPUT P1-TMV/I-504A
+ 1 1 D5400 DAM4ALAW
D5400 SENSOR MALFUNCTION THERMOCOUPLE LOW OUTPUT
+ 2 0 D5401 D5404
D5401 CONTINUING SENSOR MALFUNCTION T/C SHORTED
* 2 0 D5402 D5403
D5404 CONTINUING T/C REF JUNCT OVEN MALFUNCTION
+ 5 0 D0501 D0502 D0504 D0751
D5402 SENSOR MALFUNCTION THERMOCOUPLE SHORTED
+ 0 5 DTC40HRD DTC40KRD DTC40MRD DTC40NRD DTC40PRD
D5403 NO ALARM OR OPERATOR ACTION FOR SHORTED T/C CONDITION
+ 0 2 DAN82LAW DHU82LCZ
D0501 CONTINUING T/C REF JUNCT OVEN MALFUNCTION R1-CRJ-50-1
* 1 1 D1501 DHR501AW
D0502 CONTINUING T/C REF JUNCT OVEN MALFUNCTION R1-CRJ-50-2
* 1 1 D1502 DHR502AW
D0504 CONTINUING T/C REF JUNCT OVEN MALFUNCTION R1-CRJ-50-4
* 1 1 D1504 DHR504AW
D0505 CONTINUING T/C REF JUNCT OVEN MALFUNCTION R1-CRJ-50-5
* 1 1 D1505 DHR505AW
D0751 CONTINUING T/C REF JUNCT OVEN MALFUNCTION R1-CRJ-75-1
* 1 1 D1751 DHR751AW
D1501 NO ALARM OR OPERATOR ACTION OVEN TEMP HIGH R1-CRJ-50-1
+ 0 2 DAN501AW DHU500CZ
D1502 NO ALARM OR OPERATOR ACTION OVEN TEMP HIGH R1-CRJ-50-2
+ 0 2 DAN502AW DHU500CZ
D1504 NO ALARM OR OPERATOR ACTION OVEN TEMP HIGH R1-CRJ-50-4
+ 0 2 DAN504AW DHU500CZ
D1505 NO ALARM OR OPERATOR ACTION OVEN TEMP HIGH R1-CRJ-50-5
+ 0 2 DAN505AW DHU500CZ
D1751 NO ALARM OR OPERATOR ACTION OVEN TEMP HIGH R1-CRJ-75-1
+ 0 2 DAN751AW DHU500CZ

H1000	3	0	1 FAILED OR	REDUCED HEAT	REMOVAL	Y	SHIELD COOLING
T1000	3	0	1 THIMBLE COOLING DECAY HEAT REMOVAL LESS THAN 31 KW				
DCR0010E	0	0	2 COOLER #1	NAK BLOCKAGE			
DCR0010F	0	0	3 COOLER #1	NAK LEAKAGE			
DHU0000L	0	0	4 HUMAN ERROR - FAILURE TO RECOVER COOLERS				
DTW0010E	0	0	5 COOLER #1	CHIMNEY BLOCKED			
DAT01UOP	0	0	6 COOLER 1 UPPER ACTUATOR OR	LINKAGE FAILS			TO OPEN DAMPER
DAT01UOP	0	0	7 COOLER 1 LOWER ACTUATOR OR	LINKAGE FAILS			TO OPEN DAMPER
DSV01AON	0	0	8 SOLENOID	COOLER #1			FAILS TO CLOSE R4-VS-681A
DRE53ACP	0	0	9 AUX ANN RELAY PT 2 (CONTACT)	FAILS TO OPEN			ITEM 53A-PR2A
DARX82QP	0	0	10 RONAN X82 ALARM RELAY	FAILS TO OPEN			P1-TA-504A
DARX82QP	0	0	11 RONAN X82 ALARM CIRCUIT	FAILS LOW			P1-TA-504A
DAM4BLQW	0	0	12 EXPN RANGE TEMP RONAN X81MV/I	FAILS LOW ELEC			P1-TWV/I-504B
DAM4ALAW	0	0	13 AV BULK NA TEMP RONAN X81MV/I	FAILS LOW ELEC			P1-TWV/I-504A
DTCA40HRD	0	0	14 BULK SODIUM	THERMOCOUPLE SHORTED			RI-TC-540H
DTCA40KRD	0	0	15 BULK SODIUM	THERMOCOUPLE SHORTED			RI-TC-540K
DTCA40MRD	0	0	16 BULK SODIUM	THERMOCOUPLE SHORTED			RI-TC-540M
DTCA40NRD	0	0	17 BULK SODIUM	THERMOCOUPLE SHORTED			RI-TC-540N
DTCA40PRD	0	0	18 BULK SODIUM	THERMOCOUPLE SHORTED			RI-TC-540P
DAN62LAW	0	0	19 BULK SODIUM LOW TEMPERATURE	ANNUN FAILS			
DHU82LCZ	0	0	20 HUMAN ERROR	BULK SODIUM LOW TEMP			ALARM IGNORED
DHR501AW	0	0	21 T/C REFERENCE	JUNCTION OVEN FAILS HIGH			R1-CRJ-50-1
DHR502AW	0	0	22 T/C REFERENCE	JUNCTION OVEN FAILS HIGH			R1-CRJ-50-2
DHR504AW	0	0	23 T/C REFERENCE	JUNCTION OVEN FAILS HIGH			R1-CRJ-50-4
DHR505AW	0	0	24 T/C REFERENCE	JUNCTION OVEN FAILS HIGH			R1-CRJ-50-5
DHR751AW	0	0	25 T/C REFERENCE	JUNCTION OVEN FAILS HIGH			R1-CRJ-75-1
DAN501AW	0	0	26 T/C REFERENCE	JUNCTION HIGH ANNUN FAILS			FOR R1-CRJ-50-1
DHU500CZ	0	0	27 HUMAN ERROR	ANY T/C REF JUNCT >152F			ALARM IGNORED
DAN502AW	0	0	28 T/C REFERENCE	JUNCTION HIGH ANNUN FAILS			FOR R1-CRJ-50-2
DAN504AW	0	0	29 T/C REFERENCE	JUNCTION HIGH ANNUN FAILS			FOR R1-CRJ-50-4
DAN505AW	0	0	30 T/C REFERENCE	JUNCTION HIGH ANNUN FAILS			FOR R1-CRJ-50-5
DAN751AW	0	0	31 T/C REFERENCE	JUNCTION HIGH ANNUN FAILS			FOR R1-CRJ-75-1
BOC02Z00	2	0	32 BALANCE OF PLANT NOT AVAILABLE FOR LONG TERM	HEAT REMOVAL			

0.744E+03

0.744E+03

1 0 0.100E-09 0.100E+02 0.000E+00

2 2 0.100E-06 0.100E+02

3 2 0.300E-06 0.100E+02

4 0 0.400E-05 0.100E+02 0.000E+00

5 2 0.110E-05 0.100E+02

6 0 0.300E-02 0.500E+01 0.000E+00

7 0 0.300E-02 0.500E+01 0.000E+00

8 0 0.300E-02 0.500E+01 0.000E+00

9 0 0.300E-03 0.100E+02 0.000E+00

10 0 0.300E-03 0.100E+02 0.000E+00

11 1 0.300E-05 0.100E+02

12 1 0.300E-05 0.100E+02

13 1 0.300E-05 0.100E+02

14 1 0.100E-05 0.100E+02

15 1 0.100E-05 0.100E+02

16 1 0.100E-05 0.100E+02

17 1 0.100E-05 0.100E+02

18 1 0.100E-05 0.100E+02

19 1 0.300E-05 0.100E+02

20 0 0.100E-01 0.100E+02 0.000E+00

21 1 0.300E-05 0.100E+02

22 1 0.300E-05 0.100E+02

23 1 0.300E-05 0.100E+02

24 1 0.300E-05 0.100E+02

25 1 0.300E-05 0.100E+02

26	1	0.300E-05	0.100E+02	
27	0	0.100E-01	0.100E+02	0.000E+00
28	1	0.300E-05	0.100E+02	
29	1	0.300E-05	0.100E+02	
30	1	0.300E-05	0.100E+02	
31	1	0.300E-05	0.100E+02	
32	T	0.100E+01	0.100E+01

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Long Term Decay Heat Removal; S/D Coolers;	
Shield and Thimble Cooling	
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2	4	2	
Long Term Decay Heat Removal; S/D Coolers; Shield and Thimble Cooling			
D4000	LONG TERM	DECAY HEAT REMOVAL	NOT AVAILABLE
	* 1 2 D4001	D1104 B0002Z00	
D4001	SHIELD OR THIMBLE COOLING NOT AVAILABLE		
	+ 0 2 H1000	T1000	
D1104	3 0	1 BOTH SHUTDOWN COOLERS UNDERCOOLING NOT RECOVERED	
H1000	3 0	1 FAILED OR REDUCED HEAT REMOVAL BY SHIELD COOLING	
T1000	3 0	1 THIMBLE COOLING DECAY HEAT REMOVAL LESS THAN 31 KW	
B0002Z00	2 0	2 BALANCE OF PLANT NOT AVAILABLE FOR LONG TERM HEAT REMOVAL	
		0.744E+03	
		0.744E+03	
	1 0	0.100E-09 0.100E+02 0.000E+00	
	2 F	0.100E+01 0.100E+01

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Short Term Decay Heat Removal; Shutdown	
Coolers	
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Short Term Decay Heat Removal; Shutdown	
Coolers	
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Short Term Decay Heat Removal; Shutdown Coolers		NOT AVAILABLE >8 HR	
D6000L	SHORT TERM	DKHR	
D2103L	+ 1 0 D2103L		
D2103L	UNDERCOOLING SHUTDOWN COOLER #1 NOT RECOVERED		
D2101L	+ 2 0 D2101L D2104L		
D2101L	UNRECOVERABLE LOSS OF	SHUTDOWN	COOLER #1
D2104L	+ 0 2 9CR001OE 9CR001OF		
D2104L	UNRECOVERED RECOVERABLE UNDERCOOLING COOLER #1		
D2102L	* 1 1 D2102L 9HU0000L		
D2102L	RECOVERABLE UNDERCOOLING SHUTDOWN COOLER #1		
D2110L	+ 2 1 D2110L D2120L 9TW001OE		
D2110L	SHUTDOWN COOLER #1	UPPER DAMPERS	FAIL TO OPEN
D2120L	+ 1 1 D2105L 9AT01UOP		
D2120L	SHUTDOWN COOLER #1	LOWER DAMPERS	FAIL TO OPEN
D2105L	+ 1 1 D2105L 9AT01LOP		
D2105L	COOLER #1	AIR ACTUATORS	REMAIN
D6811L	+ 1 1 D6811L 9SV01AON		
D6811L	ELECT POWER NOT REMOVED	FROM SOLENOIDS R4-VS-681A & B	
D0531L	+ 1 1 D0531L 9RE53ACP		
D0531L	ELECT POWER NOT REMOVED	FROM RELAY	ITEM 53A-PR2A
D0821L	+ 1 1 D0821L 9ARX82QP		
D0821L	RONAN X82 ALARM PROCESSING	ERROR	P1-TA-504A
D0810L	+ 1 1 D0810L 9ARX82QW		
D0810L	EXPANDED RANGE MV/I FALSE LOW ELEC OUTPUT	P1-TMV/I-504B	
D5040L	+ 1 1 D5040L 9AM4BLOW		
D5040L	AV BULK NA MV/I FALSE LOW	ELEC OUTPUT	P1-TMV/I-504A
D5400L	+ 1 1 D5400L 9AM4ALAW		
D5400L	SENSOR MALFUNCTION	THERMOCOUPLE	LOW OUTPUT
D5401L	+ 2 0 D5401L D5404L		
D5401L	CONTINUING SENSOR	MALFUNCTION	T/C SHORTED
D5404L	* 2 0 D5402L D5403L		
D5404L	CONTINUING T/C REF JUNCT	OVEN	MALFUNCTION
D5402L	+ 5 0 D0501L D0502L D0504L D0505L D0751L		
D5402L	SENSOR MALFUNCTION	THERMOCOUPLE	SHORTED
D5403L	+ 0 5 9TC40HRD 9TC40MRD 9TC40NRD 9TC40PRD		
D5403L	NO ALARM OR OPERATOR ACTION FOR SHORTED	T/C CONDITION	
D0501L	+ 0 2 9AN82LAW 9HU82LCZ		
D0501L	CONTINUING T/C REF JUNCT	OVEN MALFUNCTION	R1-CRJ-50-1
D0502L	* 1 1 D1501L 9HR501AW		
D0502L	CONTINUING T/C REF JUNCT	OVEN MALFUNCTION	R1-CRJ-50-2
D0504L	* 1 1 D1502L 9HR502AW		
D0504L	CONTINUING T/C REF JUNCT	OVEN MALFUNCTION	R1-CRJ-50-4
D0505L	* 1 1 D1504L 9HR504AW		
D0505L	CONTINUING T/C REF JUNCT	OVEN MALFUNCTION	R1-CRJ-50-5
D0751L	* 1 1 D1505L 9HR505AW		
D0751L	CONTINUING T/C REF JUNCT	OVEN MALFUNCTION	R1-CRJ-75-1
D1501L	* 1 1 D1751L 9HR751AW		
D1501L	NO ALARM OR OPERATOR ACTION	OVEN TEMP HIGH	R1-CRJ-50-1
D1502L	+ 0 2 9AN501AW 9HU500CZ		
D1502L	NO ALARM OR OPERATOR ACTION	OVEN TEMP HIGH	R1-CRJ-50-2
D1504L	+ 0 2 9AN502AW 9HU500CZ		
D1504L	NO ALARM OR OPERATOR ACTION	OVEN TEMP HIGH	R1-CRJ-50-4
D1505L	+ 0 2 9AN504AW 9HU500CZ		
D1505L	NO ALARM OR OPERATOR ACTION	OVEN TEMP HIGH	R1-CRJ-50-5
D1751L	+ 0 2 9AN505AW 9HU500CZ		
D1751L	NO ALARM OR OPERATOR ACTION	OVEN TEMP HIGH	R1-CRJ-75-1
9CR001OE	+ 0 2 9AN751AW 9HU500CZ		
9CR001OE	0 0 1 COOLER #1	NAK	BLOCKAGE
9CR001OF	0 0 2 COOLER #1	NAK	LEAKAGE

2	3	2	
FAULT TREE FOR THE 480 VAC EMERGENCY BUS E-1			
E0100	LOSS OF 480 VAC EMERGENCY BUS E-1		
	+ 1 1 E0101 EBSE11YC		
E0101	SUPPLY FAULT LOSS OF 480 VAC BUS E-1		
	* 0 2 E0102 E0105		
E0102	3 0 1 SUPPLY/OPEN	FAULT LOSS OF	POWER FROM 480 NORMAL BUS No.2
E0105	3 0 1 SUPPLY.OPEN	FAULT LOSS OF	POWER FROM 400 KW GENERATOR
EBSE11YC	0 0 2	Uncleared Overload on Emergency Bus E-1	
	0.744E+03		
	0.744E+03		
	1 0 0.100E-09	0.100E+02	0.000E+00
	2 2 0.100E-06	0.100E+02

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FAULT TREE FOR THE 480 VAC EMERGENCY BUS E-1	
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1 4 4
FAULT TREE FOR THE 480 VAC EMERGENCY BUS E-1
E0102 SUPPLY/OPEN FAULT LOSS OF POWER FROM 480 NORMAL BUS No.2
      + 0 4 J0530 EWR2E1YB JCB05CVO ESW004YO
J0530 3 0 1 LOSS OF 480 VAC NORMAL BUS No.2
EWR2E1YB 0 0 2 480 Vac Normal Bus No. 2 to E-1 Wiring Fails Open
JCB05CVO 0 0 3 Normal Bus No. 2 Breaker 5C Fails Open
ESW004YO 0 0 4 E-1 Bus Disconn. Switch 4 Fails Open
0.744E+03
0.744E+03
1 0 0.100E-09 0.100E+02 0.000E+00
2 2 0.100E-06 0.100E+02
3 2 0.300E-06 0.100E+02
4 2 0.100E-05 0.100E+02

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FAULT TREE FOR THE 480 VAC EMERGENCY BUS E-1
 E0105 SUPPLY.OPEN FAULT LOSS OF POWER FROM 400 KW GENERATOR
 + 3 4 E0106 E0111 E0098 EWR4SWHB EGE4EFHW ECB4OBHO EDL41OHU
 E0106 400 KW DIESEL FAILS TO START
 + 1 1 E0107 EDL4SFHS
 E0111 480 VAC BUS E-1 BREAKER 3D FAILS TO CLOSE
 + 1 4 E0112 ECB3D2YN LWR3D4BB LFU3D4BB L0500
 E0098 EXCESSIVE LOAD ON 400 KW GENERATOR
 + 1 1 E0099 ECB1XYP
 E0107 FAULT IN 400 KW DIESEL START SIGNAL CKT
 + 0 3 EIN4SSHW L0500 LFU400BB
 E0112 E-1 BREAKER 3D CLOSE SIGNAL FAILS
 + 1 1 E0113 EIN3D3YW
 E0099 INTERLOCK FAILURE OVERLOAD ON 400 KW GENERATOR
 * 1 1 E0113 EIN1XYW
 E0113 NORMAL BUS No.2 BREAKER 5C FAILS TO OPEN ON DEMAND
 + 1 1 E0114 JCBN5CVP
 E0114 NORMAL BUS No.2 BREAKER 5C OPEN SIGNAL FAILS
 + 0 3 L0510 JFUN5CVB JINN5CVW
 L0500 3 0 1 LOSS OF 480 VAC EMERGENCY SWITCHGEAR 125 VDC BUS No.1
 L0510 3 0 1 LOSS OF 480 VAC 125 VDC BUS
 EWR4SWHB 0 0 2 400 Kw Generator Supply Wiring Fails Open
 EGE4EFHW 0 0 3 400 Kw Diesel Fails to Run
 ECB4OBHO 0 0 4 400 Kw Generator Output Breaker Fails Open
 EDL41OHU 0 0 5 400/100 KW Diesel Fuel Supply Common Mode Failure
 EDL4SFHS 0 0 6 400 KW Diesel Fails to Start on Command
 ECB3D2YN 0 0 7 E-1 Breaker 3D Failure to Close
 LWR3D4BB 0 0 8 E-1 Breaker 3D 125 Vdc Sup. Wiring Opens
 LFU3D4BB 0 0 9 E-1 Breaker 3D 125 Vdc Sup. Wire/Fuse Opens
 ECB1XYP 0 0 10 Selected Emergency Power Loads are not Shed
 EIN4SSHW 0 0 11 400 Diesel Start Signal Ckt Fails
 LFU400BB 0 0 12 400 Kw Diesel Start Sup. Fuse Open
 EIN3D3YW 0 0 13 E-1 Breaker 3D Close Signal Ckt Fails
 EIN1XYW 0 0 14 Brkr 5C - Brkr 3D Interlock Fails and 3D Closes
 JCBN5CVP 0 0 15 Normal Bus No. 2 Breaker 5C Fails to Open
 JFUN5CVB 0 0 16 Normal Bus No. 2 Breaker 5C Trip Fuse Opens
 JINN5CVW 0 0 17 Normal Bus No. 2 Breaker 5C Trip Signal Ckt Fails
 0.744E+03
 0.744E+03

FAULT TREE FOR THE LOSS OF THE 480 VAC	
EMERGENCY BUS P-4E	
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FAULT TREE FOR THE LOSS OF THE 480 VAC EMERGENCY BUS P-4E
 E0120 LOSS OF 480 VAC EMERGENCY BUS P-4E
 + 1 1 ED121 EBP4EYC
 ED121 SUPPLY/OPEN FAULT LOSS OF 480 VAC BUS P-4E
 + 1 3 ED122 EWP4EYB ESWP4EYW ECBP42YO
 ED122 LOSS OF 480 VAC OUTPUT FROM P-4E MAN TRANSF SWITCH
 * 2 0 ED129 ED125
 ED129 LOSS OF E-1 480 VAC FROM P-4E MAN. TRANSF. SWITCH
 + 0 4 ED100 ESWP4EYB EWRE1AYB ECBE1AYO
 ED125 LOSS OF NORMAL 480 VAC FROM P-4E MAN. TRANSF. SWITCH
 + 1 2 ED126 ESWP4NYB ESWP4NYZ
 ED126 LOSS OF NORMAL 480 VAC TO P-4E MAN. TRANSF. SWITCH
 + 0 4 JD500 JWRN3AVB JCBN3AVN JCBN3AVZ
 ED100 3 0 1 LOSS OF 480 VAC EMERGENCY BUS E-1
 JD500 3 0 1 LOSS OF 480 VAC NORMAL BUS No.1
 EBP4EYC 0 0 2 Uncleared Overload Fault on Bus P-4E
 EWP4EYB 0 0 3 Line From P-4E to 10 KW Rectifier Fails
 ESWP4EYW 0 0 4 Total Failure of P-4E Man Transf Switch
 ECBP42YO 0 0 5 P-4E Supply Breaker 2C Fails Open
 ESWP4EYB 0 0 6 E-1 Side of P-4E Man. Trans. Switch Fails Open
 EWRE1AYB 0 0 7 Emergency Bus E-1 Breaker 1A Wiring Fails Open
 ECBE1AYO 0 0 8 Emergency Bus E-1 Breaker 1A Fails Open
 ESWP4NYB 0 0 9 Normal Pwr Side of P-4E Man. Transf. Switch Fails Open
 ESWP4NYZ 0 0 10 Operator Fails to Switch P-4E Man. Transf. SW to Norm Pwr
 JWRN3AVB 0 0 11 Normal Bus No. 1 Breaker 3A Wiring Fails Open
 JCBN3AVN 0 0 12 Normal Bus No. 1 Breaker 3A Fails Open
 JCBN3AVZ 0 0 13 Operator Fails to Close Norm Bus No. 1 Brkr 3A
 0.744E+03
 0.744E+03

1	0	0.100E-09	0.100E+02	0.000E+00
2	2	0.100E-06	0.100E+02	
3	5	0.100E-06	0.100E+02	0.876E+04
4	2	0.100E-06	0.100E+02	
5	2	0.300E-06	0.100E+02	
6	2	0.100E-05	0.100E+02	
7	2	0.100E-06	0.100E+02	
8	2	0.300E-06	0.100E+02	
9	5	0.100E-05	0.100E+02	0.876E+04
10	0	0.100E-03	0.100E+02	0.100E+01
11	5	0.100E-06	0.100E+02	0.876E+04
12	5	0.300E-06	0.100E+02	0.876E+04
13	0	0.100E-03	0.100E+02	0.100E+01

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FAULT TREE FOR THE LOSS OF THE 480 VAC	
LIGHTING DISTRIBUTION PANEL E-2	
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FAULT TREE FOR THE LOSS OF THE 480 VAC LIGHTING DISTRIBUTION PANEL E-2
 E0132 LOSS OF 480 VAC LIGHTING DISTRIBUTION PANEL E-2
 + 2 0 E0133 E0140
 E0133 SUPPLY FAULT LOSS OF 480 VAC EMER LIGHTING PANEL E-2
 * 2 0 E0135 E0138
 E0140 OVERLOAD FAULT LOSS OF 480 VAC EMERGENCY LIGHT DISTR PAN E-2
 + 0 2 EBSEE2YC EWREE2YC
 E0135 E-2 SUPPLY/OPEN PWR LOSS FROM 480 VAC NORMAL BUS No.1
 + 0 4 JWRNE2YB J0500 JCB03BVO ECBH1AYO
 E0138 E-2 SUPPLY/OPEN PWR LOSS FROM 480 VAC EMERGENCY BUS E-1
 + 1 4 E0139 EWREE2YC EWREE2YB E0100 ESCE2EYW
 E0139 EMERGENCY BUS E-1 BREAKER 2D FAILS TO CLOSE
 + 0 2 ECB2DMYN ECB2DHYZ
 J0500 3 0 1 LOSS OF 480 VAC NORMAL BUS No.1
 E0100 3 0 1 LOSS OF 480 VAC EMERGENCY BUS E-1
 EBSEE2YC 0 0 2 Uncleared Overload on Emergency Bus E-2
 EWREE2YC 0 0 3 Uncleared Overload on Emergency Pwr Line to E-2
 JWRNE2YB 0 0 4 Normal Power Line to E-2 Fails Open
 JCB03BVO 0 0 5 Normal Bus No. 1 Breaker 3B Fails Open
 ECBH1AYO 0 0 6 E-2 Bus Breaker 1A Fails Open
 EWREE2YB 0 0 7 Emergency Pwr Line to E-2 Fails Open
 ESCE2EYW 0 0 8 Emergency Bus E-2 Interlock Fails Gen. On, 3B, 2D closed
 ECB2DMYN 0 0 9 E-2 Breaker 2D Fails to Remain Closed
 ECB2DHYZ 0 0 10 Operator Error Fails to Close E-2 Breaker 2D
 0.744E+03
 0.744E+03
 1 0 0.100E-09 0.100E+02 0.000E+00
 2 2 0.100E-06 0.100E+02
 3 5 0.100E-06 0.100E+02 0.876E+04
 4 2 0.100E-06 0.100E+02
 5 2 0.300E-06 0.100E+02
 6 2 0.300E-06 0.100E+02
 7 5 0.100E-06 0.100E+02 0.876E+04
 8 5 0.300E-05 0.100E+02 0.876E+04
 9 5 0.300E-06 0.100E+02 0.876E+04
 10 0 0.100E-03 0.100E+02 0.100E+01

FAULT TREE FOR THE LOSS OF THE 480 VAC	
EMERGENCY BUS R-1B	
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FAULT TREE FOR THE LOSS OF THE 480 VAC	
EMERGENCY BUS R-1B	
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FAULT TREE FOR THE LOSS OF THE 480 VAC

EMERGENCY BUS R-1B

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FAULT TREE FOR THE LOSS OF THE 480 VAC	
EMERGENCY BUS R-1B	
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FAULT TREE FOR THE LOSS OF THE 480 VAC	
EMERGENCY BUS R-1B	
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FAULT TREE FOR THE LOSS OF THE 480 VAC EMERGENCY BUS R-1B
 E0150 LOSS OF 480 VAC EMERGENCY BUS R-1B
 + 1 1 E0152 EBSR1BYC
 E0152 LOSS OF 480 VAC EMERGENCY BUS R-1B POWER SOURCES
 * 2 0 E0160 E0154
 E0160 LOSS OF R-1B POWER FROM NORMAL BUS No.2
 + 1 2 E0162 EWR2E1YC E0102
 E0154 LOSS OF R-1B POWER FROM R-1A AND 400 KW GENERATOR
 + 2 0 E0156 E0158
 E0162 LOSS OF E-1 TO R-1B LINE
 + 1 2 E0164 EWR11BYC EBS1E1YC
 E0156 LOSS OF R-1B, R-1A PWR FAILS BEFORE OR WITH NORMAL BUS No.2
 * 1 1 E0170 J0550
 E0158 LOSS OF R-1B, R-1A POWER FAILS AFTER NORMAL BUS No.1
 * 2 0 E0172 E0180
 E0164 OPEN FAULT ON E-1 TO R-1B LINE
 + 0 3 EWR11BYB ECB2BBYO ECB16AYO
 E0170 400 KW GEN. POWER TO R-1B FAILS, AUTO CONNECT MODE
 + 1 1 E0162 E0105
 E0172 R-1A PWR FAILS AFTER POWER FAILURE FROM NORMAL BUS No.2
 + 1 1 E0174 J0550
 E0180 400 KW GEN. PWR TO R-1B FAILS, MANUAL CONNECT MODE
 + 1 2 E0162 E0105 ECB11BYZ
 E0174 BUS R-1 BREAKER 8A FAILS TO CLOSE OR TRIPS OPEN
 + 3 1 E0176 E0184 E0188 ECBT8AYO
 E0176 BUS R-1 BREAKER 8A FAILS TO CLOSE
 + 1 1 E0178 ECB18AYN
 E0184 BUS R-1 BREAKER 16A FAILS TO OPEN
 + 1 1 E0186 ECB16AYP
 E0188 BUS E-1 BREAKER 2B FAILS TO OPEN
 + 1 1 E0190 ECB2BAYP
 E0178 BUS R-1 BREAKER 8A CLOSE SIGNAL FAILS
 + 0 4 L0530 EFUD8ABB EIND8AYW EWRD8AYB
 E0186 BUS R-1 BREAKER 16A OPEN SIGNAL FAILS
 + 0 4 L0530 EFU16ABB EIN16AYW EWR16AYB
 E0190 BUS E-1 BREAKER 2B OPEN SIGNAL FAILS
 + 0 4 L0520 EFU12BBB EIN12BYW EWR12BYB
 E0102 3 0 1 SUPPLY/OPEN FAULT LOSS OF POWER FROM 480 NORMAL BUS No.2
 J0550 3 0 1 LOSS OF 480 VAC NORMAL BUS R-1A
 E0105 3 0 1 SUPPLY.OPEN FAULT LOSS OF POWER FROM 400 KW GENERATOR
 L0530 3 0 1 LOSS OF 125 VDC MCC BUS
 L0520 3 0 1 LOSS OF 480 VAC EMERGENCY SWITCHGEAR 125 VDC BUS No. 2
 EBSR1BYC 0 0 2 Uncleared Overload on Emergency Bus R-1B
 EWR2E1YC 0 0 3 Overload on Normal Bus No. 2 to E-1 Line
 EWR11BYC 0 0 4 Overload Fault on E-1 to R-1B Line
 EBS1E1YC 0 0 5 Uncleared Overload on Emergency Bus E-1
 EWR11BYB 0 0 6 E-1 to R-1B Line Fails Open
 ECB2BBYO 0 0 7 Emergency Bus E-1 Breaker 2B Fails Open
 ECB16AYO 0 0 8 Emergency Bus R-1B Breaker 16A Fails Open
 ECB11BYZ 0 0 9 Operator Error in Connecting E-1 to R-1B
 ECBT8AYO 0 0 10 Bus R-1 Breaker 8A Trips Open
 ECB18AYN 0 0 11 Bus R-1 Breaker 8A Fails to Close on Demand
 ECB16AYP 0 0 12 Bus R-1 Breaker 16A Fails to Open on Demand
 ECB2BAYP 0 0 13 Bus E-1 Breaker 2B Fails to Open on Demand
 EFUD8ABB 0 0 14 Bus R-1 Breaker 8A Close Fuse Fails Open
 EIND8AYW 0 0 15 Bus R-1 Breaker 8A Close Signal Ckt Fails
 EWRD8AYB 0 0 16 Bus R-1 Breaker 8A Close Ckt Sup. Wiring Fails Open
 EFU16ABB 0 0 17 Bus R-1 Breaker 16A Trip Fuse Opens
 EIN16AYW 0 0 18 Bus R-1 Breaker 16A Trip Signal Ckt Fails

EWR16AYB 0 0 19 Bus R-1 Breaker 16A Trip Ckt Sup. Wiring Fails Open
 EFU12BBB 0 0 20 Bus E-1 Breaker 2B Trip Fuse Opens
 EIN12BYW 0 0 21 Bus E-1 Breaker Trip Signal Ckt Fails
 EWR12BYB 0 0 22 Bus E-1 Breaker 2B Trip Ckt Sup. Wiring Fails Open
 0.744E+03
 0.744E+03
 1 0 0.100E-09 0.100E+02 0.000E+00
 2 2 0.100E-06 0.100E+02
 3 2 0.100E-06 0.100E+02
 4 2 0.100E-06 0.100E+02
 5 2 0.100E-06 0.100E+02
 6 2 0.100E-06 0.100E+02
 7 2 0.300E-06 0.100E+02
 8 2 0.300E-06 0.100E+02
 9 0 0.100E-03 0.100E+02 0.100E+01
 10 2 0.300E-06 0.100E+02
 11 0 0.500E-03 0.100E+02 0.100E+01
 12 0 0.500E-03 0.100E+02 0.100E+01
 13 0 0.500E-03 0.100E+02 0.100E+01
 14 5 0.100E-06 0.100E+02 0.876E+04
 15 5 0.300E-05 0.100E+02 0.876E+04
 16 5 0.100E-06 0.100E+02 0.876E+04
 17 5 0.100E-06 0.100E+02 0.876E+04
 18 5 0.300E-05 0.100E+02 0.876E+04
 19 5 0.100E-06 0.100E+02 0.876E+04
 20 5 0.100E-06 0.100E+02 0.876E+04
 21 5 0.300E-05 0.100E+02 0.876E+04
 22 2 0.100E-06 0.100E+02

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FAULT TREE FOR THE LOSS OF THE 480 VAC	
EMERGENCY BUS P-1B	
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FAULT TREE FOR THE LOSS OF THE 480 VAC	
EMERGENCY BUS P-1B	
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FAULT TREE FOR THE LOSS OF THE 480 VAC	
EMERGENCY BUS P-1B	
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FAULT TREE FOR THE LOSS OF THE 480 VAC	
EMERGENCY BUS P-1B	
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FAULT TREE FOR THE LOSS OF THE 480 VAC EMERGENCY BUS
 E0200 LOSS OF 480 VAC EMERGENCY BUS P-1B
 + 1 1 E0202 EBSPIBYC
 E0202 LOSS OF 480 VAC EMERGENCY BUS P-1B POWER SOURCES
 * 2 0 E0210 E0204
 E0210 LOSS OF P-1B POWER FROM NORMAL BUS No.2
 + 1 2 E0212 EWR2E1YC E0102
 E0204 LOSS OF P-1B POWER FROM P-1A AND 400 KW GENERATOR
 + 2 0 E0206 E0208
 E0212 LOSS OF E-1 TO P-1B LINE
 + 1 2 E0214 EWRP1BYC EBS1E1YC
 E0206 LOSS OF P-1B, P-1A PWR FAILS BEFORE OR WITH NORMAL BUS No.2
 * 1 1 E0220 J0565
 E0208 LOSS OF P-1B, P-1A POWER FAILS AFTER NORMAL BUS No.1
 * 2 0 E0222 E0230
 E0214 OPEN FAULT ON E-1 TO P-1B LINE
 + 0 3 EWR12BYB ECB2C8YO ECB16YO
 E0220 400 KW GEN. POWER TO P-1B FAILS, AUTO CONNECT MODE
 + 1 1 E0212 E0105
 E0222 P-1A PWR FAILS AFTER POWER FAILURE FROM NORMAL BUS No.2
 + 1 1 E0224 J0565
 E0230 400 KW GEN. PWR TO P-1B FAILS, MANUAL CONNECT MODE
 + 1 2 E0212 E0105 ECB1PBYZ
 E0224 BUS P-1 BREAKER 9A FAILS TO CLOSE OR TRIPS OPEN
 + 3 1 E0226 E0234 E0238 ECHT9AYO
 E0226 BUS P-1 BREAKER 9A FAILS TO CLOSE
 + 1 1 E0228 ECH19AYN
 E0234 BUS P-1 BREAKER 16A FAILS TO OPEN
 + 1 1 E0236 ECH16YP
 E0238 BUS E-1 BREAKER 2C FAILS TO OPEN
 + 1 1 E0240 ECB2CAYP
 E0228 BUS P-1 BREAKER 9A CLOSE SIGNAL FAILS
 + 0 4 L0530 EFUD9ABB EIND9AYW EWRD9AYB
 E0236 BUS P-1 BREAKER 16A OPEN SIGNAL FAILS
 + 0 4 L0530 EFU16PBB EIN16PYW EWR16PYB
 E0240 BUS E-1 BREAKER 2C OPEN SIGNAL FAILS
 + 0 4 L0520 EFU12CBB EIN12CYW EWR12CYB
 E0102 3 0 1 SUPPLY/OPEN FAULT LOSS OF POWER FROM 480 NORMAL BUS No.2
 J0565 3 0 1 LOSS OF 480 VAC NORMAL BUS P-1A
 E0105 3 0 1 SUPPLY.OPEN FAULT LOSS OF POWER FROM 400 KW GENERATOR
 L0530 3 0 1 LOSS OF 125 VDC MCC BUS
 L0520 3 0 1 LOSS OF 480 VAC EMERGENCY SWITCHGEAR 125 VDC BUS No. 2
 EBSPIBYC 0 0 2 Uncleared Overload on Emergency Bus P-1B
 EWR2E1YC 0 0 3 Overload on Normal Bus No. 2 to E-1 Line
 EWRP1BYC 0 0 4 Overload Fault on E-1 to P-1B Line
 EBS1E1YC 0 0 5 Uncleared Overload on Emergency Bus E-1
 EWR12BYB 0 0 6 Bus E-1 Breaker 2B Trip Ckt Sup. Wiring Fails Open
 ECB2C8YO 0 0 7 Emergency Bus E-1 Breaker 2C Fails Open
 ECB16YO 0 0 8 Emergency Bus P-1B Breaker 16A Fails Open
 ECB1PBYZ 0 0 9 Operator Error in Connecting E-1 to P-1B
 ECHT9AYO 0 0 10 Bus P-1 Breaker 9A Trips Open
 ECB19AYN 0 0 11 Bus P-1 Breaker 9A Fails to Close on Demand
 ECB16YP 0 0 12 Bus P-1 Breaker 16A Fails to Open on Demand
 ECB2CAYP 0 0 13 Bus E-1 Breaker 2C Fails to Open on Demand
 EFUD9ABB 0 0 14 Bus P-1 Breaker 9A Close Fuse Fails Open
 EIND9AYW 0 0 15 Bus P-1 Breaker 9A Close Signal Ckt Fails
 EWRD9AYB 0 0 16 Bus P-1 Breaker 9A Close Ckt Sup. Wiring Fails Open
 EFU16PBB 0 0 17 Bus P-1 Breaker 16A Trip Fuse Opens
 EIN16PYW 0 0 18 Bus P-1 Breaker 16A Trip Signal Ckt Fails

EWR16PYB 0 0 19 Bus P-1 Breaker 16A Trip Ckt Sup. Wiring Fails Open
 EFU12CBB 0 0 20 Bus E-1 Breaker 2C Trip Fuse Opens
 EIN12CYW 0 0 21 Bus E-1 Breaker 2C Trip Signal Ckt Fails
 EWR12CYB 0 0 22 Bus E-1 Breaker 2C Trip Ckt Sup. Wiring Fails Open
 0.744E+03
 0.744E+03
 1 0 0.100E-09 0.100E+02 0.000E+00
 2 2 0.100E-06 0.100E+02
 3 2 0.100E-06 0.100E+02
 4 2 0.100E-06 0.100E+02
 5 2 0.100E-06 0.100E+02
 6 2 0.100E-06 0.100E+02
 7 2 0.300E-06 0.100E+02
 8 2 0.300E-06 0.100E+02
 9 0 0.100E-03 0.100E+02 0.100E+01
 10 2 0.300E-06 0.100E+02
 11 0 0.500E-03 0.100E+02 0.100E+01
 12 0 0.500E-03 0.100E+02 0.100E+01
 13 0 0.500E-03 0.100E+02 0.100E+01
 14 5 0.100E-06 0.100E+02 0.876E+04
 15 5 0.300E-05 0.100E+02 0.876E+04
 16 5 0.100E-06 0.100E+02 0.876E+04
 17 5 0.100E-06 0.100E+02 0.876E+04
 18 5 0.300E-05 0.100E+02 0.876E+04
 19 5 0.100E-06 0.100E+02 0.876E+04
 20 5 0.100E-06 0.100E+02 0.876E+04
 21 5 0.300E-05 0.100E+02 0.876E+04
 22 5 0.100E-06 0.100E+02 0.876E+04

FAULT TREE FOR THE LOSS OF THE 480 VAC	
EMERGENCY BUS S-1B	
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FAULT TREE FOR THE LOSS OF THE 480 VAC EMERGENCY BUS S-
 E0250 LOSS OF 480 VAC EMERGENCY BUS S-1B
 + 1 1 E0252 EBS1BYC
 E0252 LOSS OF 480 VAC EMERGENCY BUS S-1B POWER SOURCES
 * 2 0 E0254 E0258
 E0254 LOSS OF S-1B POWER FROM EMERGENCY BUS P-1B
 + 1 1 E0260 E0200
 E0258 LOSS OF S-1B POWER FROM S-1A GIVEN POWER FROM P-1B FAILS
 + 1 2 E0270 J0570 ECB42EYO
 E0260 LOSS OF P-1B TO S-1B LINE
 + 1 1 E0262 EWRPSIYC
 E0270 LOSS OF S-1A TO S-1B LINE
 + 1 1 E0272 EWRSSIYC
 E0262 OPEN FAULT ON P-1B TO S-1B LINE
 + 0 4 EWRPSIYB ECB8EAYO ECB14AYO ECB42EYO
 E0272 OPEN FAULT ON S-1A TO S-1B LINE
 + 1 2 E0276 EWRSSIYB ECB6AAYO
 E0276 EMERGENCY BUS S-1B CONTACTOR 42N FAILS TO CLOSE
 + 0 2 ERE42NYO EIN42NYW
 E0200 3 0 1 LOSS OF 480 VAC EMERGENCY BUS P-1B
 J0570 3 0 1 LOSS OF 480 VAC NORMAL BUS S-1A
 EBS1BYC 0 0 2 Uncleared Overload on Emergency Bus S-1B
 ECB42EYO 0 0 3 Emergency Bus S-1B Contractor 42E Fails Open
 EWRPSIYC 0 0 4 Overload Fault on P-1B to S-1B Line
 EWRSSIYC 0 0 5 Overload Fault on S-1A to S-1B Line
 EWRPSIYB 0 0 6 P-1B to S-1B Line Fails Open
 ECB8EAYO 0 0 7 Emergency Bus S-1B Breaker 8E Fails Open
 ECB14AYO 0 0 8 Emergency Bus P-1B Breaker 14AA Fails Open
 EWRSSIYB 0 0 9 S-1A to S-1B Line Fails Open
 ECB6AAYO 0 0 10 Emergency Bus S-1 Breaker 6A Fails Open
 ERE42NYO 0 0 11 Emergency Bus S-1B Cntcr 42N Fails to Close on Demand
 EIN42NYW 0 0 12 Emergency Bus S-1B Cntcr 42N Close Signal Ckt Fails
 0.744E+03
 0.744E+03
 1 0 0.100E-09 0.100E+02 0.000E+00
 2 2 0.100E-06 0.100E+02
 3 5 0.300E-06 0.100E+02 0.876E+04
 4 2 0.100E-06 0.100E+02
 5 2 0.100E-06 0.100E+02
 6 2 0.100E-06 0.100E+02
 7 2 0.100E-06 0.100E+02
 8 2 0.100E-06 0.100E+02
 9 2 0.100E-06 0.100E+02
 10 2 0.100E-06 0.100E+02
 11 0 0.500E-03 0.100E+02 0.100E+01
 12 5 0.300E-05 0.100E+02 0.876E+04

FAULT TREE FOR THE LOSS OF THE 480 VAC	
EMERGENCY BUS R-2A	
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3	5	4	
FAULT TREE FOR THE LOSS OF THE 480 VAC EMERGENCY BUS R-2A			
E0300	LOSS OF 480 VAC EMERGENCY BUS R-2A		
	+ 1 2 E0302 EBSR2BYC EBSR2AYC		
E0302	LOSS OF 480 VAC BUS R-2A POWER SOURCES		
	* 1 1 E0304 E0308		
E0304	BUS R-2A LOSS OF POWER FROM BUS R-2B		
	+ 0 2 E0306 ECB23BYO		
E0308	3 0 1 BUS R-2A LOSS OF POWER FROM 100 KW GENERATOR		
E0306	3 0 1 LOSS OF R-1B AND E-1 POWER TO EMERGENCY BUS R-2B		
EBSR2BYC	0 0 2 Uncleared Overload on Emergency Bus R-2B		
EBSR2AYC	0 0 3 Uncleared Overload on Emergency Bus R-2A		
ECB23BYO	0 0 4 Emergency Bus R-2 Tie Breaker 3B Fails Open		
	0.744E+03		
	0.744E+03		
1	0 0.100E-09 0.100E+02 0.000E+00		
2	2 0.100E-06 0.100E+02		
3	2 0.100E-06 0.100E+02		
4	2 0.300E-06 0.100E+02

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FAULT TREE FOR THE LOSS OF THE 480 VAC

EMERGENCY BUS R-2A

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FAULT TREE FOR THE LOSS OF THE 480 VAC	
EMERGENCY BUS R-2A	
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FAULT TREE FOR THE LOSS OF THE 480 VAC EMERGENCY BUS R-2A
 E0306 LOSS OF R-1B AND E-1 POWER TO EMERGENCY BUS R-2B
 * 2 0 E0334 E0344
 E0334 LOSS OF EMERG BUS R-1B PWR TO EMERG BUS R-2B (NORMAL OPS.)
 + 1 1 E0336 E0150
 E0344 LOSS OF EMERG BUS E-1 PWR TO EMERG BUS R-2B (ALT. OPS.)
 + 1 1 E0343 ROO99A00
 E0336 LOSS OF R-1B TO R-2B LINE
 + 1 1 E0338 EWRRI2YC
 E0343 LOSS OF SUPPLY FROM E-1 - ALT. OPS.
 * 1 1 E0342 ROO99B00
 E0338 OPEN FAULT ON R-1B TO R-2B LINE
 + 0 3 EWRRI2YB ESWRIBYO ECH15CYO
 E0342 LOSS OF SUPPLY FROM EMERGENCY BUS E-1
 + 1 1 E0346 E0100
 E0346 LOSS OF E-1 TO R-2B LINE
 + 1 1 E0348 EWRRIYC
 E0348 OPEN FAULT ON E-1 TO R-2B LINE
 + 0 4 EWRRI2YB ESWRIYB ECH15CYO ESWRIYB
 E0150 3 0 1 LOSS OF 480 VAC EMERGENCY BUS R-1B
 E0100 3 0 1 LOSS OF 480 VAC EMERGENCY BUS E-1
 ROO99A00 2 0 2 Reactor Operation Permitted
 EWRRI2YC 0 0 3 Overload Fault on R-1B to R-2B Line
 ROO99B00 2 0 4 Reactor Operation Not Permitted
 EWRRI2YB 0 0 5 R-1B to R-2B Line Fails Open
 ESWRIBYO 0 0 6 Bus R-1B Manual Trans. Switch R-1B Side Fails Open
 ECH15CYO 0 0 7 Bus R-1B Breaker 15C Fails Open
 EWRRIYC 0 0 8 Overload Fault on E-1 to R-2B Line
 EWRRI2YB 0 0 9 E-1 to R-2B Line Fails Open
 ESWRIYB 0 0 10 Bus R-1B Manual Trans. Switch E-1 Side Fails Open
 ECH15CYO 0 0 11 Bus E-1 Breaker 1-B Fails Open
 ESWRIYB 0 0 12 Operator Fails to Effect E-1 to R-2B Line on Demand
 0.744E+03
 0.744E+03
 1 0 0.100E-09 0.100E+02 0.000E+00
 2 T 0.100E+01 0.100E+01
 3 2 0.100E-06 0.100E+02
 4 F 0.000E+00 0.100E+01
 5 2 0.100E-06 0.100E+02
 6 2 0.100E-05 0.100E+02
 7 2 0.300E-06 0.100E+02
 8 2 0.100E-06 0.100E+02
 9 2 0.100E-06 0.100E+02
 10 0 0.100E-04 0.100E+02 0.100E+01
 11 2 0.300E-06 0.100E+02
 12 0 0.100E-03 0.100E+02 0.100E+01

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FAULT TREE FOR THE LOSS OF THE 480 VAC

EMERGENCY BUS R-2A

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FAULT TREE FOR THE LOSS OF THE 480 VAC EMERGENCY BUS
 E0308 BUS R-2A LOSS OF POWER FROM 100 KW GENERATOR
 + 3 4 E0310 E0326 E0314 EWR1SWGB ECH1OBGO EGE1EFGW EDL41OHU
 E0310 100 KW DIESEL FAILS TO START
 + 1 1 E0316 EDL1SEGS
 E0326 EMERGENCY BUS R-2A BREAKER 1A FAILS TO CLOSE
 + 1 4 E0328 ECB21AYN LMR21ABB LFU21ABB L0530
 E0314 EXCESSIVE LOAD ON 100 KW GENERATOR
 + 1 1 E0318 ECBER2YP
 E0316 FAULT IN 100 KW DIESEL START SIGNAL CKT
 + 0 3 EIN1SSGW L0530 LFU100BB
 E0328 R-2A BREAKER 1A CLOSE SIGNAL FAILS
 + 1 2 E0322 EIN21AYW ECB15CYP
 E0318 INTERLOCK FAILURE OVERLOAD ON 100 KW GENERATOR
 * 1 2 E0322 EIN151YW ECB15CYP
 E0322 EMERGENCY BUS R-1B BRKR 15C OPEN SIGNAL FAILS
 + 0 4 L0530 EFU15CYB EIN15CYW EWR15CYB
 L0530 3 0 1 LOSS OF 125 VDC MCC BUS
 EWR1SWGB 0 0 2 100 KW Generator Supply Wiring Fails Open
 ECH1OBGO 0 0 3 100 KW Generator Output Breaker Fails Open
 EGE1EFGW 0 0 4 100 KW Diesel Fails to Run
 EDL41OHU 0 0 5 400/100 KW Diesel Fuel Supply Common Mode Failure
 EDL1SEGS 0 0 6 100 KW Diesel Fails to Start on Command
 ECB21AYN 0 0 7 R-2A Breaker 1A Fails to Close
 LMR21ABB 0 0 8 R-2A Breaker 1A 125 VDC Sup. Wiring Opens
 LFU21ABB 0 0 9 R-2A Breaker 1A 125 VDC Sup. Wire/Fuse Opens
 ECBER2YP 0 0 10 Selected 100 KW Emergency Power Loads Are Not Shed
 EIN1SSGW 0 0 11 100 KW Diesel Start Signal Ckt Fails
 LFU100BB 0 0 12 100 KW Diesel Start Sup. Fuse Open
 EIN21AYW 0 0 13 R-2A Breaker 1A Close Signal Ckt Fails
 ECB15CYP 0 0 14 Emergency Bus R-1B Brkr 15C Fails to Open on Demand
 EIN151YW 0 0 15 Brkr 15C - Brkr 1A Interlock Fails and 1A Closes
 EFU15CYB 0 0 16 Emergency Bus R-1B Brkr 15C Trip Fuse Opens
 EIN15CYW 0 0 17 Emergency Bus R-1B Brkr 15C Trip Signal Ckt Fails
 EWR15CYB 0 0 18 Emerg Bus R-1B Brkr 15C Trip Ckt Sup. Wiring Fails Open
 0.744E+03
 0.744E+03

1	0	0.100E-09	0.100E+02	0.000E+00
2	5	0.100E-06	0.100E+02	0.876E+04
3	5	0.300E-06	0.100E+02	0.876E+04
4	2	0.500E-02	0.100E+02	0.168E+03
5	5	0.500E-03	0.100E+02	0.168E+03
6	0	0.100E-01	0.500E+01	0.100E+01
7	0	0.500E-03	0.100E+02	0.100E+01
8	5	0.100E-06	0.100E+02	0.876E+04
9	5	0.100E-06	0.100E+02	0.876E+04
10	5	0.300E-05	0.100E+02	0.876E+04
11	5	0.300E-05	0.100E+02	0.168E+03
12	5	0.100E-06	0.100E+02	0.168E+03
13	5	0.300E-05	0.100E+02	0.876E+04
14	0	0.500E-03	0.100E+02	0.100E+01
15	5	0.300E-05	0.100E+02	0.876E+04
16	5	0.100E-06	0.100E+02	0.876E+04
17	5	0.300E-05	0.100E+02	0.876E+04
18	5	0.100E-06	0.100E+02	0.876E+04

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FAULT TREE FOR THE LOSS OF THE 480 VAC

EMERGENCY BUS R-2B

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4	6	5	
FAULT TREE FOR THE LOSS OF THE 480 VAC EMERGENCY BUS R-2B			
E0360	LOSS OF 480 VAC EMERGENCY BUS R-2B		
	+ 2 1 E0362 E0364 EBSR2BYC		
E0362	OVERLOAD ON EMERGENCY BUS R-2A AFFECTING BUS R-2B		
	* 0 2 EBSR2AYC ECB23PYP		
E0364	LOSS OF 480 VAC BUS R-2B, NORMAL OPS. CONFIGURATION		
	* 1 1 E0366 E0306		
E0366	BUS R-2B LOSS OF POWER FROM BUS R-2A		
	+ 0 2 ECB23BYO E0308		
E0306	3 0 1 LOSS OF R-1B AND E-1 POWER TO EMERGENCY BUS R-2B		
E0308	3 0 1 BUS R-2A LOSS OF POWER FROM 100 KW GENERATOR		
EBSR2BYC	0 0 2 Uncleared Overload on Emergency Bus R-2B		
EBSR2AYC	0 0 3 Uncleared Overload on Emergency Bus R-2A		
ECB23PYP	0 0 4 Bus R-2 Tie Breaker 3B Fails to Open on Overload		
ECB23BYO	0 0 5 Emergency Bus R-2 Tie Breaker 3B Fails Open		
	0.744E+03		
	0.744E+03		
	1 0 0.100E-09 0.100E+02 0.000E+00		
	2 2 0.100E-06 0.100E+02		
	3 2 0.100E-06 0.100E+02		
	4 0 0.500E-03 0.100E+02 0.100E+01		
	5 2 0.300E-06 0.100E+02

1	2	1
Loss of Output from 13.8 kV Bus		
J0100	Loss of 13.8 kV Bus Power	
	* 0 2	J0101 J0201
J0101	3 0	1 Loss of 13.8 kV Bus 1
J0201	3 0	1 Loss of 13.8 kV Bus 2
0.744E+03		
0.744E+03		
1 0	0.100E-09	0.100E+02 0.000E+00

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Loss of Output from 13.8 kV Bus	
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Loss of Output from 13.8 kV Bus	
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J0101	Loss of Output from 13.8 kV Bus		
	Loss of 13.8 kV Bus 1		
J0125	+ 2 1 J0125 J0102 JSG100ZU		
	Loss of 13.8 kV Bus 1 from Bus Overload		
J0102	+ 1 1 J0128 JBS0012D		
	Loss of Input to 13.8 kV Bus 1		
J0128	* 3 0 J0103 J0110 J0120		
	Loss of 13.8 kV Bus 1 from Bus 2 Overload		
J0103	* 0 2 JBS0022D JSG1092P		
	Loss of Output from 13.8 kV Switchgear Unit 5		
J0110	+ 1 1 J0104 JSG10520		
	Loss of 13.8 kV Feed from Bus 2		
J0120	+ 1 1 J0203 JSG10920		
	Loss of Output from 13.8 kV Switchgear Unit 7		
J0104	+ 1 1 J0121 JSG10720		
	Loss of Input to 13.8 kV Switchgear Unit 5		
J0203	+ 1 2 J0105 JWR1050B JWR1000U		
	Loss of Output from 13.8 kV Switchgear Unit 10		
J0121	+ 1 1 J0204 JSG11020		
	Loss of Input to 13.8 kV Switchgear Unit 7		
J0105	+ 1 1 J0122 JWR1070B		
	Loss of Output from 138/13.8 kV Transformer 1		
J0204	+ 1 2 J0106 JTR0010W JTR0120U		
	Loss of Input to 13.8 kV Switchgear Unit 10		
J0122	+ 1 2 J0205 JWR1100B JWR1000U		
	Loss of 25600 kVA Generator Output		
J0106	* 0 2 J0001200 JGE0012W		
	Loss of Input to 138/13.8 kV Transformer 1		
J0205	+ 0 3 JWR0010B JBS0010B JWR0000U		
	Loss of Output from 138/13.8 kV Transformer 2		
J0206	+ 1 2 J0206 JTR0020W JTR0120U		
	Loss of Input to 138/13.8 kV Transformer 2		
JSG100ZU	+ 0 3 JWR0020B JBS0010B JWR0000U		
JBS0012D	0 0 1 13.8 kV Switchgear Common Mode Failure		
JBS00022D	0 0 2 Overload Fault on 13.8 kV Bus 1		
JSG1092P	0 0 3 Overload Fault on 13.8 kV Bus 2		
JSG10520	0 0 4 13.8 kV Switchgear Bus Tie (Unit 9) Fails to Open		
JSG10920	0 0 5 13.8 kV Switchgear Unit 5 Fails Open		
JSG10720	0 0 6 13.8 kV Switchgear Bus Tie (Unit 9) Fails Open		
JWR1050B	0 0 7 13.8 kV Switchgear Unit 7 Fails Open		
JWR1000U	0 0 8 13.8 kV Feed Wire to Switchgear Unit 5 Fails		
JSG11020	0 0 9 13.8 kV Feed Wires to Switchgear Common Mode Failure		
JWR1070B	0 0 10 13.8 kV Switchgear Unit 10 Fails Open		
JTR0010W	0 0 11 Feed Wire to 13.8 kV Switchgear Unit 7 Fails		
JTR0120U	0 0 12 138/13.8 kV Transformers 1 Fails		
JWR1100B	0 0 13 138/13.8 kV Transformers Common Mode Failure		
J0001200	2 0 14 13.8 kV Feed Wire to Switchgear Unit 10 Fails		
JGE0012W	0 0 15 Generator On-Line		
JWR0010B	0 0 16 Generator Electrical or Mechanical Malfunction		
JBS0010B	0 0 17 138 kV Feed Wire to Transformer 1 Fails		
JWR0000U	0 0 18 Loss of 138 kV Bus		
JTR0020W	0 0 19 138 kV Feed Wires Common Mode Failure		
JWR0020B	0 0 20 138/13.8 kV Transformer 2 Fails		
0.744E+03	0 0 21 138 kV Feed Wire to Transformer 2 Fails		
0.744E+03			
1 2 0.300E-07	0.100E+02		
2 2 0.100E-06	0.100E+02		
3 2 0.100E-06	0.100E+02		

4	0	0.500E-03	0.100E+02	0.000E+00
5	2	0.300E-06	0.100E+02	
6	2	0.300E-06	0.100E+02	
7	2	0.300E-06	0.100E+02	
8	2	0.100E-06	0.100E+02	
9	2	0.100E-07	0.100E+02	
10	2	0.300E-06	0.100E+02	
11	2	0.100E-06	0.100E+02	
12	2	0.100E-05	0.100E+02	
13	2	0.100E-06	0.100E+02	
14	2	0.100E-06	0.100E+02	
15	T	0.100E+01	0.100E+01	
16	4	0.300E-03	0.300E+01	0.300E+01
17	2	0.100E-06	0.100E+02	
18	2	0.100E-03	0.100E+02	
19	2	0.100E-07	0.100E+02	
20	2	0.100E-05	0.100E+02	
21	2	0.100E-06	0.100E+02

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Loss of Output from 13.8 kV Bus	
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17	21	21	
	Loss of Output from 13.8 kV Bus		
J0201	Loss of 13.8 kV Bus 2		
	+ 2 1 J0225 J0202 JSG1002U		
J0225	Loss of 13.8 kV Bus 2 from Bus Overload		
	+ 1 1 J0228 JBS0022D		
J0202	Loss of Input to 13.8 kV Bus 2		
	+ 2 0 J0203 J0210		
J0228	Loss of 13.8 kV Bus 2 from Bus 1 Overload		
	+ 0 2 JBS0012D JSG1092P		
J0203	Loss of Output from 13.8 kV Switchgear Unit 10		
	+ 1 1 J0204 JSG11020		
J0210	Loss of 13.8 kV Feed from Bus 1		
	+ 1 1 J0211 JSG10920		
J0204	Loss of Input to 13.8 kV Switchgear Unit 10		
	+ 1 2 J0205 JWR1100B JWR1000U		
J0211	Loss of Input to 13.8 kV Bus 1		
	+ 2 0 J0103 J0120		
J0205	Loss of Output from 138/13.8 kV Transformer 2		
	+ 1 2 J0206 JTR0020W JTR0120U		
J0103	Loss of Output from 13.8 kV Switchgear Unit 5		
	+ 1 1 J0104 JSG10520		
J0120	Loss of Output from 13.8 kV Switchgear Unit 7		
	+ 1 1 J0121 JSG10720		
J0206	Loss of Input to 138/13.8 kV Transformer 2		
	+ 0 3 JWR0020B JBS0010B JWR0000U		
J0104	Loss of Input to 13.8 kV Switchgear Unit 5		
	+ 1 2 J0105 JWR1050B JWR1000U		
J0121	Loss of Input to 13.8 kV Switchgear Unit 7		
	+ 1 1 J0122 JWR1070B		
J0105	Loss of Output from 138/13.8 kV Transformer 1		
	+ 1 2 J0106 JTR0010W JTR0120U		
J0122	Loss of 25600 KVA Generator Output		
	+ 0 2 J0001200 JGE0012W		
J0106	Loss of Input to 138/13.8 kV Transformer 1		
	+ 0 3 JWR0010B JBS0010B JWR0000U		
JSG1002U	0 0 1 13.8 kV Switchgear Common Mode Failure		
JBS0022D	0 0 2 Overload Fault on 13.8 kV Bus 2		
JBS0012D	0 0 3 Overload Fault on 13.8 kV Bus 1		
JSG1092P	0 0 4 13.8 kV Switchgear Bus Tie (Unit 9) Fails to Open		
JSG11020	0 0 5 13.8 kV Switchgear Unit 10 Fails Open		
JSG10920	0 0 6 13.8 kV Switchgear Bus Tie (Unit 9) Fails Open		
JWR1100B	0 0 7 13.8 kV Feed Wire to Switchgear Unit 10 Fails		
JWR1000U	0 0 8 13.8 kV Feed Wires to Switchgear Common Mode Failure		
JTR0020W	0 0 9 138/13.8 kV Transformer 2 Fails		
JTR0120U	0 0 10 138/13.8 kV Transformers Common Mode Failure		
JSG10520	0 0 11 13.8 kV Switchgear Unit 5 Fails Open		
JSG10720	0 0 12 13.8 kV Switchgear Unit 7 Fails Open		
JWR0020B	0 0 13 138 kV Feed Wire to Transformer 2 Fails		
JBS0010B	0 0 14 Loss of 138 kV Bus		
JWR0000U	0 0 15 138 kV Feed Wires Common Mode Failure		
JWR1050B	0 0 16 13.8 kV Feed Wire to Switchgear Unit 5 Fails		
JWR1070B	0 0 17 Feed Wire to 13.8 kV Switchgear Unit 7 Fails		
JTR0010W	0 0 18 138/13.8 kV Transformer 1 Fails		
J0001200	2 0 19 Generator On-Line		
JGE0012W	0 0 20 Generator Electrical or Mechanical Malfunction		
JWR0010B	0 0 21 138 kV Feed Wire to Transformer 1 Fails		
	0.744E+03		
	0.744E+03		
	1 2 0.300E-07 0.100E+02		

2	2	0.100E-06	0.100E+02
3	2	0.100E-06	0.100E+02
4	0	0.500E-03	0.100E+02
5	2	0.300E-06	0.100E+02
6	2	0.300E-06	0.100E+02
7	2	0.100E-06	0.100E+02
8	2	0.100E-07	0.100E+02
9	2	0.100E-05	0.100E+02
10	2	0.100E-06	0.100E+02
11	2	0.300E-06	0.100E+02
12	2	0.300E-06	0.100E+02
13	2	0.100E-06	0.100E+02
14	2	0.100E-03	0.100E+02
15	2	0.100E-07	0.100E+02
16	2	0.100E-06	0.100E+02
17	2	0.100E-06	0.100E+02
18	2	0.100E-05	0.100E+02
19	T	0.100E+01	0.100E+01
20	4	0.300E-03	0.300E+01
21	2	0.100E-06	0.100E+02

0.000E+00

0.300E+01

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Loss of 2400 V Bus Power	
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18	18	17			
			Loss of 2400 V Bus Power		
J0301			Loss of 2400 V Bus Power		
			+ 1 2 J0302 JBS200YD JSG200VU		
J0302			Loss of Input to 2400 V Bus		
			* 2 0 J0310 J0330		
J0310			Loss of Output from 2400 V Switchgear		Unit 3
			+ 1 1 J0311 JSG203VO		
J0330			Loss of Output from 2400 V Switchgear		Unit 9
			+ 1 1 J0331 JSG209VO		
J0311			Loss of Input to 2400 V Switchgear		Unit 3
			+ 1 2 J0312 JWR203VB JWR200VU		
J0331			Loss of Input to 2400 V Switchgear		Unit 9
			+ 1 2 J0332 JWR209VB JWR200VU		
J0312			Loss of Output from 13.8/2.4 kV		Transformer 3
			+ 2 0 J0315 J0320		
J0332			Loss of Output from 13.8/2.4 kV		Transformer 4
			+ 2 0 J0335 J0340		
J0315			Loss of Input to 13.8/2.4 kV Transformer 3		
			+ 2 0 J0317 J0318		
J0320			13.8/2.4 kV Transformer 3 Failure		
			+ 0 2 JTR0340U JTR003OW		
J0335			Loss of Input to 13.8/2.4 kV Transformer 4		
			+ 2 0 J0337 J0338		
J0340			13.8/2.4 kV Transformer 4 Failure		
			+ 0 2 JTR0340U JTR004OW		
J0317			Loss of Output from 13.8 kV Switchgear		Unit 3
			+ 1 1 J0319 J0101		
J0318			Loss of 13.8 kV Feed to Transformer 3		
			+ 0 2 JWR1200U JWR1032B		
J0337			Loss of Output from 13.8 kV Switchgear		Unit 12
			+ 1 1 J0339 J0201		
J0338			Loss of 13.8 kV Feed to Transformer 4		
			+ 0 2 JWR1200U JWR1122B		
J0319			13.8 kV Switchgear Unit 3		Failure
			+ 0 2 JSG1002U JSG1032O		
J0339			13.8 kV Switchgear Unit 12		Failure
			+ 0 2 JSG1002U JSG1122O		
J0101			3 0 1 Loss of 13.8 kV Bus 1		
J0201			3 0 1 Loss of 13.8 kV Bus 2		
JBS200YD			0 0 2 2400 V Bus Overload Fault		
JSG200VU			0 0 3 2400 V Switchgear Common Mode Failure		
JSG203VO			0 0 4 2400 V Switchgear Unit 3 Fails Open		
JSG209VO			0 0 5 2400 V Switchgear Unit 9 Fails Open		
JWR203VB			0 0 6 Feed Wire to 2400 V Switchgear Unit 3 Fails		
JWR200VU			0 0 7 2400 V Feed Wires Common Mode Failure		
JWR209VB			0 0 8 Feed Wire to 2400 V Switchgear Unit 9 Fails		
JTR0340U			0 0 9 13.8/2.4 kV Transformers Common Mode Failure		
JTR003OW			0 0 10 13.8/2.4 kV Transformer 3 Fails		
JTR004OW			0 0 11 13.8/2.4 kV Transformer 4 Fails		
JWR1200U			0 0 12 13.8 kV Feed Wires Common Mode Failure		
JWR1032B			0 0 13 13.8 kV Feed Wire to Transformer 3 Fails		
JWR1122B			0 0 14 13.8 kV Feed Wire to Transformer 4 Fails		
JSG1002U			0 0 15 13.8 kV Switchgear Common Mode Failure		
JSG1032O			0 0 16 13.8 kV Switchgear Unit 3 Fails Open		
JSG1122O			0 0 17 13.8 kV Switchgear Unit 12 Fails Open		
			0.744E+03		
			0.744E+03		
			1 0 0.100E-09 0.100E+02 0.000E+00		
			2 2 0.100E-06 0.100E+02		

3	2	0.300E-07	0.100E+02
4	2	0.300E-06	0.100E+02
5	2	0.300E-06	0.100E+02
6	2	0.100E-06	0.100E+02
7	2	0.100E-07	0.100E+02
8	2	0.100E-06	0.100E+02
9	2	0.100E-06	0.100E+02
10	2	0.100E-05	0.100E+02
11	2	0.100E-05	0.100E+02
12	2	0.100E-07	0.100E+02
13	2	0.100E-06	0.100E+02
14	2	0.100E-06	0.100E+02
15	2	0.300E-07	0.100E+02
16	2	0.300E-06	0.100E+02
17	2	0.300E-06	0.100E+02

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LOSS OF 480 VAC NORMAL BUS No.1	
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LOSS OF 480 VAC NORMAL BUS No.1	
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LOSS OF 480 VAC NORMAL BUS No.1
 J0500 LOSS OF 480 VAC NORMAL BUS No.1
 + 2 2 J0501 J0510 JBS4802U JTRT560U
 J0501 LOSS OF INPUT TO 480 VAC NORMAL BUS No.1
 * 2 0 J0502 J0503
 J0510 OVERLOAD FAULT LOSS OF 480 VAC NORMAL BUS No.1
 + 2 0 J0511 J0512
 J0502 LOSS OF TRANSF. No.5 POWER TO NORMAL BUS No.1
 + 0 5 J0101 JCA1312B JTR0050B JCB00220 JCB1CDVO
 J0503 LOSS OF TRANSF. No.6 POWER TO NORMAL BUS No.1
 + 1 1 J0532 JCB4BCVO
 J0511 UNCLEARED OVERLOAD ON NORMAL BUS No.1
 + 1 2 J0513 JBSN1LVC JCAN1SVC
 J0512 UNCLEARED OVERLOAD ON BUS No.2 AFFECTING NORMAL BUS No.1
 * 2 0 J0517 J0539
 J0532 LOSS OF TRANSF. No.6 POWER TO NORMAL BUS No.2
 + 0 5 J0201 JCA1322B JTR0060B JCB01320 JCB7CDVO
 J0513 UNCLEARED OVERLOAD BEFORE 1CD AFFECTING NORMAL BUS No.1
 * 1 1 J0514 JCAT50VC
 J0517 TIE BREAKER 4BC FAILS TO OPEN AUTOMATICALLY
 + 1 1 J0519 JCB4BCVP
 J0539 UNCLEARED OVERLOAD ON NORMAL BUS No.2
 + 1 2 J0541 JBSN2LVC JCAN2SVC
 J0514 BREAKER 1CD FAILS TO OPEN AUTOMATICALLY
 + 1 1 J0516 JCB1CDVP
 J0519 OVERLOAD FAULT DETECTION CKT 4BC-21 FAILS
 + 0 5 JFU417VB LFU4BCVB LWR210VB L0510 JIN210VW
 J0541 UNCLEARED OVERLOAD BEFORE 7CD AFFECTING NORMAL BUS No.2
 * 1 1 J0544 JCAT60VC
 J0516 OVERLOAD FAULT DETECTION CKT 1CD FAILS
 + 0 5 JFU417VB JIN1CDVW LFU1CDVB LWR1CDVB L0510
 J0544 BREAKER 7CD FAILS TO OPEN AUTOMATICALLY
 + 1 1 J0546 JCB7CDVP
 J0546 OVERLOAD FAULT DETECTION CKT 7CD FAILS
 + 0 5 JFU417VB JIN7CDVW LFU7CDVB LWR7CDVB L0510
 J0101 3 0 1 Loss of 13.8 kV Bus 1
 J0201 3 0 1 Loss of 13.8 kV Bus 2
 L0510 3 0 1 LOSS OF 480 VAC 125 VDC BUS
 JBS4802U 0 0 2 480 Vac Normal Bus No. 1 and Bus No. 2 Common Mode Failure
 JTRT560U 0 0 3 13.8 KV/480 Vac Transf 5 and 6 Common Mode Failure
 JCA1312B 0 0 4 Supply Wiring Fails Open, Transf. No. 5
 JTR0050B 0 0 5 Transf. No. 5 Fails Open
 JCB00220 0 0 6 13.8 KV Breaker No. 2 Fails Open
 JCB1CDVO 0 0 7 480 Vac Breaker 1CD Fails Open
 JCB4BCVO 0 0 8 Tie Breaker 4BC Fails Open
 JBSN1LVC 0 0 9 Uncleared Overload on Normal Bus No. 1 Load Side
 JCAN1SVC 0 0 10 Overload Fault Between Breaker 1CD and Normal Bus No. 1
 JCA1322B 0 0 11 Supply Wiring Fails Open, Transf. No. 6
 JTR0060B 0 0 12 Transf. No. 6 Fails Open
 JCB01320 0 0 13 13.8 KV Breaker No. 13 Fails Open
 JCB7CDVO 0 0 14 480 Vac Breaker 7CD Fails Open
 JCAT50VC 0 0 15 Overload Fault Occurs Before Breaker 1CD
 JCB4BCVP 0 0 16 Breaker 4BC Fails to Open on Demand
 JBSN2LVC 0 0 17 Uncleared Overload on Normal Bus No. 2 Load Side
 JCAN2SVC 0 0 18 Overload Fault Between Breaker 7CD and Normal Bus No. 2
 JCB1CDVP 0 0 19 Breaker 1CD Fails to Open on Demand
 JFU417VB 0 0 20 4BC, 1CD, 7CD Fault Detection Ckts Fuse Fails Open
 LFU4BCVB 0 0 21 Breaker 4BC Trip Ckt Fuse Fails Open
 LWR210VB 0 0 22 4BC-21 Fault Detection Ckt Supply ng Fails Open

JIN21OVW	0	0	23	4BC-21 Fault Detection Ckt Fails	
JCAT6OVC	0	0	24	Overload Fault Occurs Before Breaker	0.000E+00
JIN1CDVW	0	0	25	ICD Fault Detection Ckt Fails	
LFU1CDVB	0	0	26	Breaker ICD Trip Ckt Fuse Fails Open	
LWR1CDVB	0	0	27	ICD Fault Detection Ckt Supply Wiring Fails Open	
JCB7CDVP	0	0	28	Breaker 7CD Fails to Open on Demand	
JIN7CDVW	0	0	29	7CD Fault Detection Ckt Fails	
LFU7CDVB	0	0	30	Breaker 7CD Trip Ckt Fuse Fails Open	
LWR7CDVB	0	0	31	7CD Fault Detection Ckt Supply Wiring Fails Open	
0.744E+03					
0.744E+03					
1	0	0.100E-09	0.100E+02	0.000E+00	
2	2	0.100E-07	0.100E+02		
3	2	0.100E-06	0.100E+02		
4	2	0.100E-06	0.100E+02		
5	2	0.100E-05	0.100E+02		
6	2	0.300E-06	0.100E+02		
7	2	0.300E-06	0.100E+02		
8	5	0.300E-06	0.100E+02	0.876E+04	
9	2	0.100E-06	0.100E+02		
10	2	0.100E-06	0.100E+02		
11	2	0.100E-06	0.100E+02		
12	2	0.100E-05	0.100E+02		
13	2	0.300E-06	0.100E+02		
14	2	0.300E-06	0.100E+02		
15	2	0.100E-06	0.100E+02		
16	0	0.500E-03	0.100E+02	0.100E+01	
17	2	0.100E-06	0.100E+02		
18	2	0.100E-06	0.100E+02		
19	0	0.500E-03	0.100E+02	0.100E+01	
20	5	0.100E-06	0.100E+02	0.876E+04	
21	5	0.100E-06	0.100E+02	0.876E+04	
22	5	0.100E-06	0.100E+02	0.876E+04	
23	5	0.300E-05	0.100E+02	0.876E+04	
24	2	0.100E-06	0.100E+02		
25	5	0.300E-05	0.100E+02	0.876E+04	
26	5	0.100E-06	0.100E+02	0.876E+04	
27	5	0.100E-06	0.100E+02	0.876E+04	
28	0	0.500E-03	0.100E+02	0.100E+01	
29	5	0.300E-05	0.100E+02	0.876E+04	
30	5	0.100E-06	0.100E+02	0.876E+04	
31	5	0.100E-06	0.100E+02	0.876E+04

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LOSS OF 480 VAC NORMAL BUS No. 2	
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LOSS OF 480 VAC NORMAL BUS No. 2	
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PAGE 3 OF 3	

LWR120VB	0	0	23	4BC-12 Fault Detection Ckt Supply Wiring Fails Open	
JCAT50VC	0	0	24	Overload Fault Occurs Before Breaker LCD	
JIN7CDVW	0	0	25	7CD Fault Detection Ckt Fails	
LFU7CDVB	0	0	26	Breaker 7CD Trip Ckt Fuse Fails Open	
LWR7CDVB	0	0	27	7CD Fault Detection Ckt Supply Wiring Fails Open	
JCB1CDVP	0	0	28	Breaker LCD Fails to Open on Demand	
JIN1CDVW	0	0	29	ICD Fault Detection Ckt Fails	
LFU1CDVB	0	0	30	Breaker LCD Trip Ckt Fuse Fails Open	
LWR1CDVB	0	0	31	ICD Fault Detection Ckt Supply Wiring Fails Open	
0.744E+03					
0.744E+03					
1	0	0.100E-09	0.100E+02	0.000E+00	
2	2	0.100E-06	0.100E+02		
3	2	0.100E-07	0.100E+02		
4	2	0.100E-06	0.100E+02		
5	2	0.100E-05	0.100E+02		
6	2	0.300E-06	0.100E+02		
7	2	0.300E-06	0.100E+02		
8	5	0.300E-06	0.100E+02	0.876E+04	
9	2	0.100E-06	0.100E+02		
10	2	0.100E-06	0.100E+02		
11	2	0.100E-06	0.100E+02		
12	2	0.100E-05	0.100E+02		
13	2	0.300E-06	0.100E+02		
14	2	0.300E-06	0.100E+02		
15	2	0.100E-06	0.100E+02		
16	0	0.500E-03	0.100E+02	0.100E+01	
17	2	0.100E-06	0.100E+02		
18	2	0.100E-06	0.100E+02		
19	0	0.500E-03	0.100E+02	0.100E+01	
20	5	0.100E-06	0.100E+02	0.876E+04	
21	5	0.300E-05	0.100E+02	0.876E+04	
22	5	0.100E-06	0.100E+02	0.876E+04	
23	5	0.100E-06	0.100E+02	0.876E+04	
24	2	0.100E-06	0.100E+02		
25	5	0.300E-05	0.100E+02	0.876E+04	
26	5	0.100E-06	0.100E+02	0.876E+04	
27	5	0.100E-06	0.100E+02	0.876E+04	
28	0	0.500E-03	0.100E+02	0.100E+01	
29	5	0.300E-05	0.100E+02	0.876E+04	
30	5	0.100E-06	0.100E+02	0.876E+04	
31	5	0.100E-06	0.100E+02	0.876E+04

FAULT TREE FOR THE LOSS OF THE 480 VAC	
NORMAL BUS R-1A	
Mon Apr 29 09:26:05 1991	PAGE 1 OF 1

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1 5 5
FAULT TREE FOR THE LOSS OF THE 480 VAC NORMAL BUS R-1A
J0550 LOSS OF 480 VAC NORMAL BUS R-1A
      + 0 5 JBSR1AYC JWR1AYB J0500 JCB2DIVO JCB1A1YO
J0500 3 0 1 LOSS OF 480 VAC NORMAL BUS No.1
JBSR1AYC 0 0 2 Uncleared Overload on Normal Bus R-1A
JWR1AYB 0 0 3 Wiring to Normal Bus R-1A Fails Open
JCB2DIVO 0 0 4 Normal Bus No. 1 Breaker 2D Fails Open
JCB1A1YO 0 0 5 Normal Bus R-1A Breaker 1A Fails Open
0.744E+03
0.744E+03
1 0 0.100E-09 0.100E+02 0.000E+00
2 2 0.100E-06 0.100E+02
3 2 0.100E-06 0.100E+02
4 2 0.300E-06 0.100E+02
5 2 0.300E-06 0.100E+02

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FAULT TREE FOR THE LOSS OF THE 480 VAC	
NORMAL BUS P-2A	
Mon Apr 29 14:19:08 1991	PAGE 1 OF 1

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1 5 5
FAULT TREE FOR THE LOSS OF THE 480 VAC NORMAL BUS P-2A
J0555 LOSS OF 480 VAC NORMAL BUS P-2A
      + 0 5 JBSP2AYC JWRP2AYB J0500 JCB2B1VO JCB1A2YO
J0500 3 0 1 LOSS OF 480 VAC NORMAL BUS No.1
JBSP2AYC 0 0 2 Uncleared Overload on Normal Bus P-2A
JWRP2AYB 0 0 3 Wiring to Normal Bus P-2A Fails Open
JCB2B1VO 0 0 4 Normal Bus No. 1 Breaker 2B Fails Open
JCB1A2YO 0 0 5 Normal Bus P-2A Breaker 1A Fails Open
      0.744E+03
      0.744E+03
      1 0 0.100E-09 0.100E+02 0.000E+00
      2 2 0.100E-06 0.100E+02
      3 2 0.100E-06 0.100E+02
      4 2 0.300E-06 0.100E+02
      5 2 0.300E-06 0.100E+02

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B.2-373

FAULT TREE FOR THE LOSS OF THE 480 VAC	
NORMAL BUS P-2B	
Mon Apr 29 14:19:34 1991	PAGE 1 OF 1

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1 5 5
FAULT TREE FOR THE LOSS OF THE 480 VAC NORMAL BUS P-2B
J0560 LOSS OF 480 VAC NORMAL BUS P-2B JCB5A1VO JCB12AYO
      + 0 5 JBSP2BYC JWRP2BYB J0530 JCB5A1VO JCB12AYO
J0530 3 0 1 LOSS OF 480 VAC NORMAL BUS No.2
JBSP2BYC 0 0 2 Uncleared Overload on Normal Bus P-2B
JWRP2BYB 0 0 3 Wiring to Normal Bus P-2B Fails Open
JCB5A1VO 0 0 4 Normal Bus No. 2 Breaker 5A Fails Open
JCB12AYO 0 0 5 Normal Bus P-2B Breaker 12A Fails Open
0.744E+03
0.744E+03
1 0 0.100E-09 0.100E+02 0.000E+00
2 2 0.100E-06 0.100E+02
3 2 0.100E-06 0.100E+02
4 2 0.300E-06 0.100E+02
5 2 0.300E-06 0.100E+02

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FAULT TREE FOR THE LOSS OF THE 480 VAC BUS	
P-1A	
Mon Apr 29 14:20:06 1991	PAGE 1 OF 1

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1 5 5
FAULT TREE FOR THE LOSS OF THE 480 VAC BUS P-1A
J0565 LOSS OF 480 VAC NORMAL BUS P-1A
      + 0 5 JBSPLAYC JWRPLAYB J0500 JCB2CIVO JCB1A3YO
J0500 3 0 1 LOSS OF 480 VAC NORMAL BUS No.1
JBSPLAYC 0 0 2 Uncleared Overload on Normal Bus P-1A
JWRPLAYB 0 0 3 Wiring to Normal Bus P-1A Fails Open
JCB2CIVO 0 0 4 Normal Bus No. 1 Breaker 2C Fails Open
JCB1A3YO 0 0 5 Normal Bus P-1A Breaker 1A Fails Open
0.744E+03
0.744E+03
1 0 0.100E-09 0.100E+02 0.000E+00
2 2 0.100E-06 0.100E+02
3 2 0.100E-06 0.100E+02
4 2 0.300E-06 0.100E+02
5 2 0.300E-06 0.100E+02

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FAULT TREE FOR THE LOSS OF THE 480 VAC	
NORMAL BUS S-1A	
Mon Apr 29 14:20:37 1991	PAGE 1 OF 1

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1 5 5
FAULT TREE FOR THE LOSS OF THE 480 VAC NORMAL BUS S-1A
J0570 LOSS OF 480 VAC NORMAL BUS S-1A
      + 0 5 JBSS1AYC JWRS1AYB J0500 JCB3C1VO JCB1D1YO
J0500 3 0 1 LOSS OF 480 VAC NORMAL BUS No.1
JBSS1AYC 0 0 2 Uncleared Overload on Normal Bus. S-1A
JWRS1AYB 0 0 3 Wiring to Normal Bus S-1A Fails Open
JCB3C1VO 0 0 4 Normal Bus No. 1 Breaker 3C Fails Open
JCB1D1YO 0 0 5 Normal Bus S-1A Breaker 1D Fails Open
      0.744E+03
      0.744E+03
      1 0 0.100E-09 0.100E+02 0.000E+00
      2 2 0.100E-06 0.100E+02
      3 2 0.100E-06 0.100E+02
      4 2 0.300E-06 0.100E+02
      5 2 0.300E-06 0.100E+02

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FAULT TREE FOR THE LOSS OF THE 480 VAC	
NORMAL BUS R-3	
Mon Apr 29 14:21:11 1991	PAGE 1 OF 1

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1 5 5
FAULT TREE FOR THE LOSS OF THE 480 VAC NORMAL BUS R-3
J0575 LOSS OF 480 VAC NORMAL BUS R-3
      + 0 5 JBSR3XYC JWR3XYB J0530 JCB5B1VO JCBLA4YO
J0530 3 0 1 LOSS OF 480 VAC NORMAL BUS No.2
JBSR3XYC 0 0 2 Uncleared Overload on Normal Bus R-3
JWR3XYB 0 0 3 Wiring to Normal Bus R-3 Fails Open
JCB5B1VO 0 0 4 Normal Bus No. 2 Breaker 5B Fails Open
JCBLA4YO 0 0 5 Normal Bus R-3 Breaker 1A Fails Open
0.744E+03
0.744E+03
1 0 0.100E-09 0.100E+02 0.000E+00
2 2 0.100E-06 0.100E+02
3 2 0.100E-06 0.100E+02
4 2 0.300E-06 0.100E+02
5 2 0.300E-06 0.100E+02

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FAULT TREE FOR THE LOSS OF THE 480 VAC	
NORMAL BUS P-3	
Mon Apr 29 14:21:35 1991	PAGE 1 OF 1

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1 5 5
FAULT TREE FOR THE LOSS OF THE 480 VAC NORMAL BUS P-3
J0580 LOSS OF 480 VAC NORMAL BUS P-3
      + 0 5 JBSP3XYC JWRP3XYB J0530 JCB6C1VO JCB2C1YO
J0530 3 0 1 LOSS OF 480 VAC NORMAL BUS No.2
JBSP3XYC 0 0 2 Uncleared Overload on Normal Bus P-3
JWRP3XYB 0 0 3 Wiring to Normal Bus P-3 Fails Open
JCB6C1VO 0 0 4 Normal Bus No. 2 Breaker 6C Fails Open
JCB2C1YO 0 0 5 Normal Bus P-3 Breaker 2C Fails Open
0.744E+03
0.744E+03
1 0 0.100E-09 0.100E+02 0.000E+00
2 2 0.100E-06 0.100E+02
3 2 0.100E-06 0.100E+02
4 2 0.300E-06 0.100E+02
5 2 0.300E-06 0.100E+02

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B.2-384

FAULT TREE FOR THE LOSS OF THE 125 VDC	
DISTRIBUTION CENTER BUS	
Mon Apr 29 14:22:06 1991	PAGE 2 OF 3

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FAULT TREE FOR THE LOSS OF THE 125 VDC	
DISTRIBUTION CENTER BUS	
Mon Apr 29 14:22:06 1991	PAGE 3 OF 3

FAULT TREE FOR THE LOSS OF THE 125 VDC DISTRIBUTION CENTER BUS
 L0100 LOSS OF 125 VDC DISTRIBUTION CENTER BUS
 + 2 2 L0102 L0104 LWR12MBB LBS001BU
 L0102 LOSS OF 125 VDC TO DISTR CENT. BUS, NORM AND DELAYED SOURCE
 * 2 0 L0112 U0100
 L0104 UNCLEARED OVERLOAD ON 125 VDC DISTR CENTER BUS
 + 1 1 L0106 LBS12MBA
 L0112 LOSS OF 125 VDC TO DISTR CENTR BUS FROM STATION BAT
 + 1 3 L0114 LBA12MBW LWR12TBB LSW12TBB
 U0100 LOSS OF 135 VDC TO DISTR CENTR BUS FROM UNINT PWR SUP 1
 + 1 1 U0099 ROO99A00
 L0106 DOUBLE GROUND FAULT ON 125 VDC DISTR CENTER BUS
 * 1 1 L0108 LBS122BC
 L0114 125 VDC STATION BATTERY HAS UNDETECTED LOW CHARGE
 + 2 0 L0116 L0118
 U0099 LOSS OF SUPPLY FROM UPS No.1 (ALT. OPS.)
 * 1 1 U0098 ROO99B00
 L0108 FIRST 125 VDC GROUND FAULT OCCURS W/O CORRECT. ACTION
 * 1 1 L0110 LBS121BC
 L0116 125 VDC STATION BATTERY HAS UNDETECTED LOW CAPACITY
 * 0 2 LBA12CBW LIN12CBL
 L0118 125 VDC STATION BATTERY HAS UNDETECTED LOW CHARGE RATE
 * 0 2 LBY12CBW LIN12CBW
 U0098 LOSS OF SUPPLY FROM UPS No.1
 + 1 3 U0102 LMRUPTYB LSWUPTBB LSWUPTBZ
 L0110 FIRST 125 VDC GROUND FAULT CORRECTIVE ACT NOT TAKEN
 + 0 2 LIN12MBW LIN12MBZ
 U0102 POWER FROM UPS 1 BATTERY AND CHARGER FAILS
 * 2 0 U0104 U0112
 U0104 POWER FROM UPS 1 BATTERY FAILS
 + 1 2 U0106 UBAUP1BW UCBUL1YO
 U0112 POWER FROM UPS 1 CHARGER FAILS
 + 1 1 U0114 UBYUPL1YW
 U0106 135 VDC UPS 1 BATTERY HAS UNDETECTED LOW CHARGE
 + 2 0 U0108 U0110
 U0114 LOSS OF 480 VAC TO UPS 1 CHARGER
 + 0 3 E0120 ECB3EYO EWRP4UYB
 U0108 135 VDC UPS 1 BATTERY HAS UNDETECTED LOW CAPACITY
 * 0 2 UINVIYL UBAUPIBW
 U0110 135 VDC UPS 1 BATTERY HAS UNDETECTED LOW CHARGE RATE
 * 0 2 UBYUPL1BW UINVI1BYW
 E0120 3 0 1 LOSS OF 480 VAC EMERGENCY BUS P-4E
 LWR12MBB 0 0 2 Line From 125V Throwover SW to Distr Cen. Bus Fails Open
 LBS001BU 0 0 3 125 VDC Distr. Center Bus Pwr Sources Common Mode Failure
 LBS12MBA 0 0 4 Line-to-Line Short on 125 VDC Distr Center Bus
 LBA12MBW 0 0 5 Charged 125 VDC Station Battery Fails Suddenly
 LWR12TBB 0 0 6 Line From Sta. Battery to 125 VDC Throwover Switch Fails
 LSW12TBB 0 0 7 125 VDC Throwover Sw. Sta. Battery Side Fails
 ROO99A00 2 0 8 Reactor Operation Permitted
 LBS122BC 0 0 9 Second 125 VDC Ground Fault Occurs
 ROO99B00 2 0 10 Reactor Operation Not Permitted
 LBS121BC 0 0 11 First 125 VDC Ground Fault Occurs
 LBA12CBW 0 0 12 125 VDC Station Battery Loses Capacity
 LIN12CBL 0 0 13 125 VDC Station Battery Capacity Loss Not Detected
 LBY12CBW 0 0 14 125 VDC Station Battery Has Failed Charger
 LWRUPTYB 0 0 15 125 VDC Station Battery Low Charge Rate Not Detected
 LSWUPTBB 0 0 16 Line From UPS 1 to 125 VDC Throwover Switch Fails
 LSWUPTBZ 0 0 17 125 VDC Throwover SW UPS 1 Bat Side Fails
 1 Bat
 18 Operator Error Fails to Switch to 1 Bat

LIN12MBW 0 0 19 First 125 VDC Ground Fault Alarm CK ills
 LIN12MBZ 0 0 20 Operator Error Ignores First 125 VDC Ground Fault Alarm
 URAUP1BW 0 0 21 Charged 135 VDC UPS 1 Battery Fails Suddenly
 UCBU11YO 0 0 22 135 VDC UPS 1 Battery Output Ckt Brkr Fails Open
 UBYUP1YW 0 0 23 UPS 1 Charger Circuit Fails
 ECB3EEYO 0 0 24 Bus P-4E Breaker 3EE Fails Open
 EWRP4UYB 0 0 25 Line From Bus P-4E to UPS 1 Fails
 UINV1BYL 0 0 26 135 VDC UPS 1 Battery Capacity Loss Not Detected
 URAUP1BW 0 0 27 135 VDC UPS 1 Battery Loses Capacity
 UBYUP1BW 0 0 28 135 VDC UPS 1 Battery Has Low Charge Rate
 UINV1BYW 0 0 29 135 VDC UPS 1 Battery Low Charge Rate Not Detected
 0.744E+03
 0.744E+03
 1 0 0.100E-09 0.100E+02 0.000E+00
 2 2 0.100E-06 0.100E+02
 3 2 0.100E-06 0.100E+02
 4 2 0.100E-06 0.100E+02
 5 2 0.100E-04 0.500E+01
 6 2 0.100E-06 0.100E+02
 7 2 0.100E-05 0.100E+02
 8 T 0.100E+01 0.100E+01
 9 2 0.100E-06 0.100E+02
 10 F 0.000E+00 0.100E+01
 11 2 0.100E-06 0.100E+02
 12 2 0.100E-04 0.500E+01
 13 0 0.100E-02 0.300E+02 0.100E+01
 14 2 0.100E-04 0.500E+01
 15 2 0.300E-05 0.100E+02
 16 5 0.100E-06 0.100E+02 0.876E+04
 17 5 0.100E-05 0.100E+02 0.876E+04
 18 0 0.100E-03 0.100E+02 0.100E+01
 19 2 0.300E-05 0.100E+02
 20 0 0.100E-03 0.100E+02 0.100E+01
 21 5 0.100E-04 0.500E+01 0.240E+04
 22 2 0.100E-06 0.100E+02
 23 2 0.100E-04 0.500E+01
 24 2 0.300E-06 0.100E+02
 25 2 0.100E-06 0.100E+02
 26 0 0.100E-02 0.300E+02 0.100E+01
 27 5 0.100E-04 0.500E+01 0.240E+04
 28 2 0.100E-04 0.500E+01
 29 2 0.300E-05 0.100E+02

B.2-388

FAULT TREE FOR THE LOSS OF THE 125 VDC	
EMERGENCY BUS	
Mon Apr 29 14:23:05 1991	PAGE 1 OF 1

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1 5 5
FAULT TREE FOR THE LOSS OF THE 125 VDC EMERGENCY BUS
L0200 LOSS OF 125 VDC EMERGENCY BUS
+ 0 5 E0120 ECH2FFYO EWRP4EYB LDC12EBW LBS12EBC
E0120 3 0 1 LOSS OF 480 VAC EMERGENCY BUS P-4E
ECH2FFYO 0 0 2 Bus P-4E Breaker 2F Fails Open
EWRP4EYB 0 0 3 Line From P-4E to 10 KW Rectifier Fails
LDC12EBW 0 0 4 10 KW Rectifier Fails
LBS12EBC 0 0 5 Uncleared Overload on 125 VDC Emergency Bus
0.744E+03
0.744E+03
1 0 0.100E-09 0.100E+02 0.000E+00
2 5 0.300E-06 0.100E+02 0.876E+04
3 5 0.100E-06 0.100E+02 0.876E+04
4 5 0.100E-04 0.500E+01 0.876E+04
5 5 0.100E-06 0.100E+02 0.876E+04

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FAULT TREE FOR THE LOSS OF THE 13.8 KV SWITCHGEAR 125 VDC BUS
 L0300 LOSS OF 13.8 KV SWITCHGEAR 125 VDC BUS
 + 1 3 L0302 LWR133BB LBS133BC LBS005BU
 L0302 LOSS OF 125 VDC TO 125 VDC/13.8 KV SWITCHGEAR CONTROL BUS
 + 2 0 L0304 L0306
 L0304 LOSS OF 125 VDC TO 125 VDC/13.8 KV BUS FROM DISTR CENT BUS
 + 0 4 L0100 LCB07BO LSWMD3BO LWRMD3BB
 L0306 LOSS OF 125 VDC TO 125/13.8 KV BUS FROM EMERG 125 VDC
 + 1 1 L0305 ROO99A00
 L0305 LOSS OF EMERG 125 VDC TO 13.8 KV BUS - ALT. OPS.
 + 1 1 L0303 ROO99B00
 L0303 LOSS OF EMERG 125 VDC TO 13.8 KV BUS
 + 0 5 L0200 LCBEL13BO LSWME3BO LWRME3BB LSWME3BZ
 L0100 3 0 1 LOSS OF 125 VDC DISTRIBUTION CENTER BUS
 L0200 3 0 1 LOSS OF 125 VDC EMERGENCY BUS
 LWR133BB 0 0 2 Wiring From 125 VDC MTS 3 to Bus Fails Open
 LBS133BC 0 0 3 Uncleared Overload on 125 VDC/13.8 KV Bus
 LBS005BU 0 0 4 125 VDC/13.8 KVAC Bus Common Mode Loss of Power Sources
 LCB07BO 0 0 5 125 VDC Distr Center Bus Breaker 7 Fails Open
 LSWMD3BO 0 0 6 125 VDC MTS 3 Distr Cent Bus Side Fails Open
 LWRMD3BB 0 0 7 Wiring From 125 VDC Distr Cent Bus to MTS 3 Fails
 ROO99A00 2 0 8 Reactor Operation Permitted
 ROO99B00 2 0 9 Reactor Operation Not Permitted
 LCBEL13BO 0 0 10 125 VDC Emergency Bus Breaker 3 Fails Open
 LSWME3BO 0 0 11 125 VDC MTS 3 Emergency Bus Side Fails Open
 LWRME3BB 0 0 12 Wiring From 125 VDC Emergency Bus to MTS 3 Fails
 LSWME3BZ 0 0 13 Operator Fails to Close 125 VDC MTS 3 Emerg Bus Side
 0.744E+03
 0.744E+03
 1 0 0.100E-09 0.100E+02 0.000E+00
 2 2 0.100E-06 0.100E+02
 3 2 0.100E-06 0.100E+02
 4 2 0.100E-06 0.100E+02
 5 2 0.300E-06 0.100E+02
 6 2 0.100E-05 0.100E+02
 7 2 0.100E-06 0.100E+02
 8 T 0.100E+01 0.100E+01
 9 F 0.000E+00 0.100E+01
 10 5 0.300E-06 0.100E+02 0.876E+04
 11 5 0.100E-05 0.100E+02 0.876E+04
 12 5 0.100E-06 0.100E+02 0.876E+04
 13 0 0.100E-03 0.100E+02 0.100E+01

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FAULT TREE FOR THE LOSS OF THE 2.4 KV	
SWITCHGEAR 125 VDC BUS	
Mon Apr 29 14:23:46 1991	PAGE 1 OF 1

FAULT TREE FOR THE LOSS OF THE 2.4 KV SWITCHGEAR 125 BUS
 L0400 LOSS OF 2.4 KV SWITCHGEAR 125 VDC BUS
 + 1 3 L0402 LMR244BB LBS244BC LBS004BU
 L0402 LOSS OF 125 VDC TO 125 VDC/2.4 KV SWITCHGEAR CONTROL BUS
 * 2 0 L0404 L0406
 L0404 LOSS OF 125 VDC TO 125/2.4 KV BUS FROM DISTR CENT BUS
 + 0 4 L0100 LCB006BO LWRMD4BB LSWMD4BO
 L0406 LOSS OF 125 VDC TO 125/2.4 KV BUS FROM EMERG 125 VDC
 + 1 1 L0405 ROO99A00
 L0405 LOSS OF EMERG 125 VDC TO 2.4 KV BUS - ALT. OPS.
 * 1 1 L0403 ROO99B00
 L0403 LOSS OF EMERG 125 VDC TO 2.4 KV BUS
 + 0 5 L0200 LCBE14BO LSWME4BO LMRME4BB LSWME4BZ
 L0100 3 0 1 LOSS OF 125 VDC DISTRIBUTION CENTER BUS
 L0200 3 0 1 LOSS OF 125 VDC EMERGENCY BUS
 LMR244BB 0 0 2 Wiring From 125 VDC MTS 4 to Bus Fails Open
 LBS244BC 0 0 3 Uncleared Overload on 125 VDC/2.4 KV Bus
 LBS004BU 0 0 4 125 VDC KVAC Bus Common Mode Loss of Power Sources
 LCB006BO 0 0 5 125 VDC Distr Center Bus Breaker 6 Fails Open
 LWRMD4BB 0 0 6 Wiring From 125 VDC Distr Cent Bus to MTS 4 Fails Open
 LSWMD4BO 0 0 7 125 VDC MTS 4 Distr Center Bus Side Fails Open
 ROO99A00 2 0 8 Reactor Operation Permitted
 ROO99B00 2 0 9 Reactor Operation Not Permitted
 LCBE14BO 0 0 10 125 VDC Emergency Bus Breaker 4 Fails Open
 LSWME4BO 0 0 11 125 VDC MTS 4 Emergency Bus Side Fails Open
 LMRME4BB 0 0 12 Wiring From 125 VDC Emergency Bus to MTS 4 Fails
 LSWME4BZ 0 0 13 Operator Fails to Close 125 VDC MTS 4 Emerg Bus Side
 0.744E+03
 0.744E+03
 1 0 0.100E-09 0.100E+02 0.000E+00
 2 2 0.100E-06 0.100E+02
 3 2 0.100E-06 0.100E+02
 4 2 0.100E-06 0.100E+02
 5 2 0.300E-06 0.100E+02
 6 2 0.100E-06 0.100E+02
 7 2 0.100E-05 0.100E+02
 8 T 0.100E+01 0.100E+01
 9 F 0.000E+00 0.100E+01
 10 5 0.300E-06 0.100E+02 0.876E+04
 11 5 0.100E-05 0.100E+02 0.876E+04
 12 5 0.100E-06 0.100E+02 0.876E+04
 13 0 0.100E-03 0.100E+02 0.100E+01

LOSS OF THE 480 VAC EMERGENCY SWITCHGEAR 125 VDC BUS No.1
 L0500 LOSS OF 480 VAC EMERGENCY SWITCHGEAR 125 VDC BUS No.1
 + 1 3 L0502 LWRMT2BB LBSEB1BC LBS003BU
 L0502 LOSS OF 125 VDC TO 125 VDC/480 VAC EMERG BUS 1
 * 2 0 L0504 L0506
 L0504 LOSS OF 125 VDC TO 125/480 EMERG BUS 1 FROM DISTR CT
 + 0 4 L0100 LCB10BO LSWMD2BO LWRMD2BB
 L0506 LOSS OF 125 VDC TO 125/480 EMERG BUS 1 FROM EMER 125 VDC
 + 1 1 L0505 ROO99A00
 L0505 LOSS OF EMERG 125 VDC TO 480 VAC EMERG BUSES (ALT. OPS.)
 * 1 1 L0503 ROO99B00
 L0503 LOSS OF EMERG 125 VDC TO 480 VAC EMERG BUSES
 + 0 5 L0200 LCB12BO LSWME2BO LWRME2BB LSWME2BZ
 L0100 3 0 1 LOSS OF 125 VDC DISTRIBUTION CENTER BUS
 L0200 3 0 1 LOSS OF 125 VDC EMERGENCY BUS
 LWRMT2BB 0 0 2 Wiring From 125 VDC MTS 2 to Bus Fails Open
 LBSEB1BC 0 0 3 Uncleared Overload on 125 VDC/480 Vac Emerg Bus No. 1
 LBS003BU 0 0 4 125 VDC/480 Vac Emerg Bus Common Mode Loss of Power Sources
 LCB10BO 0 0 5 125 VDC Distr Center Bus Breaker 10 Fails Open
 LSWMD2BO 0 0 6 125 VDC MTS 2 Distr Cent Bus Side Fails Open
 LWRMD2BB 0 0 7 Wiring From 125 VDC Distr Cent Bus to MTS 2 Fails
 ROO99A00 2 0 8 Reactor Operation Permitted
 ROO99B00 2 0 9 Reactor Operation Not Permitted
 LCB12BO 0 0 10 125 VDC Emergency Bus Breaker 2 Fails Open
 LSWME2BO 0 0 11 125 VDC MTS 2 Emergency Bus Side Fails Open
 LWRME2BB 0 0 12 Wiring From 125 VDC Emergency Bus to MTS 2 Fails
 LSWME2BZ 0 0 13 Operator Fails to Close 125 VDC MTS 2 Emerg Bus Side
 0.744E+03
 0.744E+03
 1 0 0.100E-09 0.100E+02 0.000E+00
 2 2 0.100E-06 0.100E+02
 3 2 0.100E-06 0.100E+02
 4 2 0.100E-06 0.100E+02
 5 2 0.300E-06 0.100E+02
 6 2 0.100E-05 0.100E+02
 7 2 0.100E-06 0.100E+02
 8 T 0.100E+01 0.100E+01
 9 F 0.000E+00 0.100E+01
 10 5 0.300E-06 0.100E+02 0.876E+04
 11 5 0.100E-05 0.100E+02 0.876E+04
 12 5 0.100E-06 0.100E+02 0.876E+04
 13 0 0.100E-03 0.100E+02 0.100E+01

FAULT TREE FOR THE LOSS OF THE 480 VAC NORMAL POWER 125 VDC BUS

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L0510  LOSS OF 480 VAC 125 VDC BUS
      + 1 3 L0512  LWRMT1BB LBS124BC LBS002BU
L0512  LOSS OF 125 VDC TO 125 VDC/480 VAC BUS
      * 2 0 L0514  L0516
L0514  LOSS OF 125 VDC TO 125 VDC/480 VAC BUS FROM DISTR CENT BUS
      + 0 4 L0100  LCB011BO LSWMD1BO LWRMD1BB
L0516  LOSS OF 125 VDC TO 125 VDC/480 VAC BUS FROM EMERG 125 VDC
      + 1 1 L0515  ROO99A00
L0515  LOSS OF EMERG 125 VDC TO 480 VAC NORM BUS (ALT. OPS.)
      * 1 1 L0513  ROO99B00
L0513  LOSS OF EMERG 125 VDC TO 480 VAC NORM BUS
      + 0 5 L0200  LCB011BO LSWME1BO LWRME1BB LSWME1BZ
L0100  3 0 1 LOSS OF 125 VDC DISTRIBUTION CENTER BUS
L0200  3 0 1 LOSS OF 125 VDC EMERGENCY BUS
LWRMT1BB 0 0 2 Wiring From 125 VDC MTS 1 to Bus Fails Open
LBS124BC 0 0 3 Uncleared Overload on 125 VDC/480 Vac Bus
LBS002BU 0 0 4 125 VDC/480 Vac Norm Bus Common Mode Loss of Power Sources
LCBD11BO 0 0 5 125 VDC Distr Center Bus Breaker 11 Fails Open
LSWMD1BO 0 0 6 125 VDC MTS 1 Distr Cent Bus Side Fails Open
LWRMD1BB 0 0 7 Wiring From 125 VDC Distr Cent Bus to MTS 1 Fails
ROO99A00 2 0 8 Reactor Operation Permitted
ROO99B00 2 0 9 Reactor Operation Not Permitted
LCBE11BO 0 0 10 125 VDC Emergency Bus Breaker 1 Fails Open
LSWME1BO 0 0 11 125 VDC MTS 1 Emergency Bus Side Fails Open
LWRME1BB 0 0 12 Wiring from 125 VDC Emergency Bus to MTS 1 Fails
LSWME1BZ 0 0 13 Operator Fails to Close 125 VDC MTS 1 Emerg Bus Side
0.744E+03
0.744E+03
1 0 0.100E-09 0.100E+02 0.000E+00
2 2 0.100E-06 0.100E+02
3 2 0.100E-06 0.100E+02
4 2 0.100E-06 0.100E+02
5 2 0.300E-06 0.100E+02
6 2 0.100E-05 0.100E+02
7 2 0.100E-06 0.100E+02
8 T 0.100E+01 0.100E+01
9 F 0.000E+00 0.100E+01
10 5 0.300E-06 0.100E+02 0.876E+04
11 5 0.100E-05 0.100E+02 0.876E+04
12 5 0.100E-06 0.100E+02 0.876E+04
13 0 0.100E-03 0.100E+02 0.100E+01

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FAULT TREE FOR THE LOSS OF THE EMERGENCY	
SWITCHGEAR 125 VDC BUS No.2	
Mon Apr 29 14:24:56 1991	PAGE 1 OF 1

1 4 4
 FAULT TREE FOR THE LOSS OF THE EMERGENCY SWITCHGEAR 125 VDC BUS No. 2
 L0520 LOSS OF 480 VAC EMERGENCY SWITCHGEAR 125 VDC BUS No. 2
 + 0 4 EFUES2BO L0500 LWRSE2BB LBSEB2BC
 L0500 3 0 1 LOSS OF 480 VAC EMERGENCY SWITCHGEAR 125 VDC BUS No. 1
 EFUES2BO 0 0 2 480 Vac Emerg Switchgear 125 Vdc Bus 1 to Bus 2 Fuse Opens
 LWRSE2BB 0 0 3 Wiring From Emerg Swgear DC Bus 1 to Bus 2 Fails Open
 LBSEB2BC 0 0 4 Uncleared Overload on 125 VDC/480 Vac Emerg Bus No. 2
 0.744E+03
 0.744E+03
 1 0 0.100E-09 0.100E+02 0.000E+00
 2 2 0.100E-06 0.100E+02
 3 2 0.100E-06 0.100E+02
 4 2 0.100E-06 0.100E+02

FAULT TREE FOR THE LOSS OF THE 125 VDC MCC BUS	
Mon Apr 29 14:25:13 1991	
PAGE 1 OF 1	

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1 4 4
FAULT TREE FOR THE LOSS OF THE 125 VDC MCC BUS
L0530 LOSS OF 125 VDC MCC BUS
      + 0 4 L0100 LCBMCCBO LWRMCCBB LCBMCCBC
L0100 3 0 1 LOSS OF 125 VDC DISTRIBUTION CENTER BUS
LCBMCCBO 0 0 2 125 VDC Distr Center Bus Breaker 5 Fails Open
LWRMCCBB 0 0 3 125 VDC MCC Bus Supply Wiring Fails Open
LCBMCCBC 0 0 4 Uncleared Overload on 125 VDC MCC Bus
0.744E+03
0.744E+03
1 0 0.100E-09 0.100E+02 0.000E+00
2 2 0.300E-06 0.100E+02
3 2 0.100E-06 0.100E+02
4 2 0.100E-06 0.100E+02

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B.2-402

FAULT TREE FOR THE LOSS OF 120 VAC	
CONTINUOUS POWER PANEL NO. 1	
Mon Apr 29 14:25:35 1991	PAGE 1 OF 1

5 13 10
 FAULT TREE FOR THE LOSS OF 120 VAC CONTINUOUS POWER P. NO. 1
 U0050 LOSS OF 120 VAC CONTINUOUS POWER PANEL No.1
 + 2 3 U0054 U0056 UBSCP1QC USWSS1YW UINU12YU
 U0054 LOSS OF LINE FROM UPS No.1 TO CPP No.1
 + 0 3 UCB1BYO UWRUC1YB USWUP1YO
 U0056 POWER SOURCES TO UPS No.1 STATIC BYPASS SWITCH FAIL
 * 1 2 U0070 U0101 U0084
 U0070 LOSS OF UPS No.2 INVERTER AND BYPASS PWR TO UPS No.1 SBS
 + 1 3 U0072 UWS21YB UCBS21YO USWSS2YW
 U0072 LOSS OF UPS No.2 INVERTER AND BYPASS PWR TO UPS No.2 SBS
 * 0 2 U0201 U0184
 U0101 3 0 1 LOSS OF UPS No.1 INVERTER POWER TO SBS No.1
 U0084 3 0 1 LOSS OF POWER FROM UPS BYPASS No.1
 U0201 3 0 1 LOSS OF UPS No.2 INVERTER POWER TO SBS No.2
 U0184 3 0 1 LOSS OF POWER FROM UPS BYPASS No.2
 UBSCP1QC 0 0 2 Uncleared Overload on Cont. Power Panel No. 1
 USWSS1YW 0 0 3 Ups No. 1 Static Bypass Switch or SBS Control Fails
 UINU12YU 0 0 4 Ups No. 1 and Ups No. 2 Common Mode Failure
 UCB1BYO 0 0 5 Ups No. 1 Isolation Breaker Fails Open
 UWRUC1YB 0 0 6 Ups No. 1 to CPP No. 1 Wiring Fails
 USWUP1YO 0 0 7 Ups No. 1 Manual Bypass Switch Fails Open
 UWS21YB 0 0 8 SBS No. 2 to SBS No.1 Wiring Fails
 UCBS21YO 0 0 9 SBS No. 2 to SBS No. 1 Circuit Breaker Fails Open
 USWSS2YW 0 0 10 Ups No. 2 Static Bypass Switch or SBS Control Fails
 0.744E+03
 0.744E+03
 1 0 0.100E-09 0.100E+02 0.000E+00
 2 2 0.100E-06 0.100E+02
 3 2 0.300E-05 0.100E+02
 4 2 0.300E-05 0.100E+02
 5 2 0.300E-06 0.100E+02
 6 2 0.100E-06 0.100E+02
 7 2 0.100E-05 0.100E+02
 8 5 0.100E-06 0.100E+02 0.876E+04
 9 5 0.300E-06 0.100E+02 0.876E+04
 10 2 0.300E-05 0.100E+02

FAULT TREE FOR THE LOSS OF 120 VAC	
CONTINUOUS POWER PANEL NO. 1	
Mon Apr 29 14:26:05 1991	PAGE 1 OF 1

2 7 7
 FAULT TREE FOR THE LOSS OF 120 VAC CONTINUOUS POWER FA NO. 1
 U0084 LOSS OF POWER FROM UPS BYPASS No.1
 + 1 2 U0086 UCB11YO UCB12YO
 U0086 LOSS OF 120 VAC FROM EMERGENCY BUS R-1B
 + 0 5 E0150 ECBU15YO UTRUBTYW UWRUBWYB UVRBSSYW
 E0150 3 0 1 LOSS OF 480 VAC EMERGENCY BUS R-1B
 UCB11YO 0 0 2 Ups No. 1 Bypass Ckt Brkr No. 1 Fails
 UCB12YO 0 0 3 Bypass Power Input Ckt Breaker to Sbs No.1 Fails
 ECBU15YO 0 0 4 Bus R-1B Breaker 15 A Fails Open
 UTRUBTYW 0 0 5 Ups Bypass 480 V / 120 V Transformer Fails
 UWRUBWYB 0 0 6 Ups Bypass Supply Wiring Fails
 UVRBSSYW 0 0 7 Ups Bypass Surge Suppressor Fails
 0.744E+03
 0.744E+03
 1 0 0.100E-09 0.100E+02 0.000E+00
 2 5 0.300E-06 0.100E+02 0.876E+04
 3 5 0.300E-06 0.100E+02 0.876E+04
 4 2 0.300E-06 0.100E+02
 5 2 0.100E-05 0.100E+02
 6 2 0.100E-06 0.100E+02
 7 2 0.300E-05 0.100E+02

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8 13 13
FAULT TREE FOR THE LOSS OF 120 VAC CONTINUOUS POWER PA NO. 1
U0101 LOSS OF UPS No.1 INVERTER POWER TO SBS No.1
+ 1 3 U0102 UIVUPIYW UCUBUPIO LMRUPTYC
U0102 POWER FROM UPS 1 BATTERY AND CHARGER FAILS
* 2 0 U0104 U0112
U0104 POWER FROM UPS 1 BATTERY FAILS
+ 1 2 U0106 UBAUPIBW UCUBU11YO
U0112 POWER FROM UPS 1 CHARGER FAILS
+ 1 1 U0114 UBYUPIYW
U0106 135 VDC UPS 1 BATTERY HAS UNDETECTED LOW CHARGE
+ 2 0 U0108 U0110
U0114 LOSS OF 480 VAC TO UPS 1 CHARGER
+ 0 3 E0120 ECB3EEO EWRP4UYB
U0108 135 VDC UPS 1 BATTERY HAS UNDETECTED LOW CAPACITY
* 0 2 UINVLBYL UBAUPIBW
U0110 135 VDC UPS 1 BATTERY HAS UNDETECTED LOW CHARGE RATE
* 0 2 UBYUPIBW UINVLBYW
E0120 3 0 1 LOSS OF 480 VAC EMERGENCY BUS P-4E
UIVUPIYW 0 0 2 Ups No. 1 Inverter Fails
UCBUPIO 0 0 3 Ups No. 1 Inverter Output Breaker Fails Open
LMRUPTYC 0 0 4 Overload on Ups 1 Line to 125 VDC Distr Center Bus
UBAUPIBW 0 0 5 Charged 135 VDC UPS 1 Battery Fails Suddenly
UCBU11YO 0 0 6 135 VDC UPS 1 Battery Output Ckt Brkr Fails Open
UBYUPIYW 0 0 7 UPS 1 Charger Circuit Fails
ECB3EEO 0 0 8 Bus P-4E Breaker 3EE Fails Open
EWRP4UYB 0 0 9 Line From Bus P-4E to UPS 1 Fails
UINVLBYL 0 0 10 135 VDC UPS 1 Battery Capacity Loss Not Detected
UBAUPIBW 0 0 11 135 VDC UPS 1 Battery Loses Capacity
UBYUPIBW 0 0 12 135 VDC UPS 1 Battery Has Low Charge Rate
UINVLBYW 0 0 13 135 VDC UPS 1 Battery Low Charge Rate Not Detected
0.744E+03
0.744E+03
1 0 0.100E-09 0.100E+02 0.000E+00
2 2 0.500E-05 0.500E+01
3 2 0.300E-06 0.100E+02
4 2 0.100E-06 0.100E+02
5 5 0.100E-04 0.500E+01 0.240E+04
6 2 0.100E-06 0.100E+02
7 2 0.100E-04 0.500E+01
8 2 0.300E-06 0.100E+02
9 2 0.100E-06 0.100E+02
10 0 0.100E-02 0.300E+02 0.100E+01
11 5 0.100E-04 0.500E+01 0.240E+04
12 2 0.100E-04 0.500E+01
13 2 0.300E-05 0.100E+02

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B.2-408

FAULT TREE FOR THE LOSS OF 120 VAC	
CONTINUOUS POWER PANEL NO. 2	
Mon Apr 29 14:27:12 1991	PAGE 1 OF 1

5 13 10
 FAULT TREE FOR THE LOSS OF 120 VAC CONTINUOUS POWER PANEL J. 2
 U0150 LOSS OF 120 VAC CONTINUOUS POWER PANEL No.2
 + 2 3 U0154 U0156 UBSCP2QC USWSS2YW UINU12YU
 U0154 LOSS OF LINE FROM UPS No.2 TO CPP No.2
 + 0 3 UCIB2YO UWRUC2YB USWUP2YO
 U0156 POWER SOURCES TO UPS No.2 STATIC BYPASS SWITCH FAIL
 * 1 2 U0170 U0201 U0184
 U0170 LOSS OF UPS No.1 INVERTER AND BYPASS PWR TO UPS No.2 SBS
 + 1 3 U0172 UWRS12YB UCBS12YO USWSS1YW
 U0172 LOSS OF UPS No.1 INVERTER AND BYPASS PWR TO UPS No. 1 SBS
 * 0 2 U0101 U0084
 U0201 3 0 1 LOSS OF UPS No.2 INVERTER POWER TO SBS No.2
 U0184 3 0 1 LOSS OF POWER FROM UPS BYPASS No.2
 U0101 3 0 1 LOSS OF UPS No.1 INVERTER POWER TO SBS No.1
 U0084 3 0 1 LOSS OF POWER FROM UPS BYPASS No.1
 UBSCP2QC 0 0 2 Uncleared Overload on Cont. Power Panel No. 2
 USWSS2YW 0 0 3 Ups No. 2 Static Bypass Switch or SBS Control Fails
 UINU12YU 0 0 4 Ups No. 1 and Ups No. 2 Common Mode Failure
 UCIB2YO 0 0 5 Ups No. 2 Isolation Breaker Fails Open
 UWRUC2YB 0 0 6 Ups No. 2 to CPP No. 2 Wiring Fails
 USWUP2YO 0 0 7 Ups No. 2 Manual Bypass Switch Fails Open
 UWRS12YB 0 0 8 SBS No. 1 to SBS No. 2 Wiring Fails
 UCBS12YO 0 0 9 SBS No. 1 to SBS No. 2 Circuit Breaker Fails Open
 USWSS1YW 0 0 10 Ups No. 1 Static Bypass Switch or SBS Control Fails
 0.744E+03
 0.744E+03
 1 0 0.100E-09 0.100E+02 0.000E+00
 2 2 0.100E-06 0.100E+02
 3 2 0.300E-05 0.100E+02
 4 2 0.300E-05 0.100E+02
 5 2 0.300E-06 0.100E+02
 6 2 0.100E-06 0.100E+02
 7 2 0.100E-05 0.100E+02
 8 5 0.100E-06 0.100E+02 0.876E+04
 9 5 0.300E-06 0.100E+02 0.876E+04
 10 2 0.300E-05 0.100E+02

FAULT TREE FOR THE LOSS OF 120 VAC	
CONTINUOUS POWER PANEL NO. 2	
Mon Apr 29 14:27:34 1991	PAGE 1 OF 1

2 7 7
 FAULT TREE FOR THE LOSS OF 120 VAC CONTINUOUS POWER PL. NO. 2
 U0184 LOSS OF POWER FROM UPS BYPASS No.2
 + 1 2 U0086 UCBE22YO UCBB21YO
 U0086 LOSS OF 120 VAC FROM EMERGENCY BUS R-1B
 + 0 5 E0150 ECBU15YO UTRUBTYW UWRUBWYB UVRBSSYW
 E0150 3 0 1 LOSS OF 480 VAC EMERGENCY BUS R-1B
 UCBB22YO 0 0 2 Ups No. 2 Bypass Ckt Brkr No. 2 Fails
 UCBB21YO 0 0 3 Bypass Power Input Ckt Breaker to SBS No. 2 Fails
 ECBU15YO 0 0 4 Bus R-1B Breaker 15 A Fails Open
 UTRUBTYW 0 0 5 Ups Bypass 480 V / 120 V Transformer Fails
 UWRUBWYB 0 0 6 Ups Bypass Supply Wiring Fails
 UVRBSSYW 0 0 7 Ups Bypass Surge Suppressor Fails
 0.744E+03
 0.744E+03
 1 0 0.100E-09 0.100E+02 0.000E+00
 2 5 0.300E-06 0.100E+02 0.876E+04
 3 5 0.300E-06 0.100E+02 0.876E+04
 4 2 0.300E-06 0.100E+02
 5 2 0.100E-05 0.100E+02
 6 2 0.100E-06 0.100E+02
 7 2 0.300E-05 0.100E+02

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FAULT TREE FOR THE LOSS OF 120 VAC

CONTINUOUS POWER PANEL NO. 2

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8 12 12
 FAULT TREE FOR THE LOSS OF 120 VAC CONTINUOUS POWER PANEL NO. 2
 U0201 LOSS OF UPS No. 2 INVERTER POWER TO SBS No. 2
 + 1 2 U0202 UIVUP2YW UCBUP2YO
 U0202 POWER FROM UPS No. 2 BATTERY AND CHARGER FAILS
 * 2 0 U0204 U0212
 U0204 POWER FORM UPS No. 2 BAT FAILS
 + 1 2 U0206 UBAUP2BW UCBU22YO
 U0212 POWER FROM UPS 2 CHARGER FAILS
 + 1 1 U0214 UBYUP2YW
 U0206 135 VDC UPS 2 BATTERY HAS UNDETECTED LOW CHARGE
 + 2 0 U0208 U0210
 U0214 LOSS OF 480 VAC TO UPS 2 CHARGER
 + 0 3 E0200 ECBD15YO EWRP12YB
 U0208 135 VDC UPS 2 BATTERY HAS UNDETECTED LOW CAPACITY
 * 0 2 UBAU21BW UINV2BYL
 U0210 135 VDC UPS 2 BATTERY HAS UNDETECTED LOW CHARGE RATE
 * 0 2 UBYUP2BW UINU2BYW
 E0200 3 0 1 LOSS OF 480 VAC EMERGENCY BUS P-1B
 UIVUP2YW 0 0 2 Ups No. 2 Inverter Fails
 UCBUP2YO 0 0 3 Ups No. 2 Inverter Output Breaker Fails Open
 UBAUP2BW 0 0 4 Charged 135 VDC Ups No. 2 Battery Fails Suddenly
 UCBU22YO 0 0 5 135 VDC Ups 2 Battery Output Ckt Brkr Fails Open
 UBYUP2YW 0 0 6 Ups 2 Charger Circuit Fails
 ECBD15YO 0 0 7 Bus P-1B Breaker 15D Fails Open
 EWRP12YB 0 0 8 Line From Bus P-1B to Ups 2 Fails
 UBAU21BW 0 0 9 135 VDC Ups 2 Battery Loses Capacity
 UINV2BYL 0 0 10 135 VDC Ups 2 Battery Capacity Loss Not Detected
 UBYUP2BW 0 0 11 135 VDC Ups 2 Battery Has Low Charge Rate
 UINU2BYW 0 0 12 135 VDC Ups 2 Battery Low Charge Rate Undetected
 0.744E+03
 0.744E+03
 1 0 0.100E-09 0.100E+02 0.000E+00
 2 2 0.500E-05 0.500E+01
 3 2 0.300E-06 0.100E+02
 4 5 0.100E-04 0.500E+01 0.240E+04
 5 2 0.300E-06 0.100E+02
 6 2 0.100E-04 0.500E+01
 7 2 0.300E-06 0.100E+02
 8 2 0.100E-06 0.100E+02
 9 5 0.100E-04 0.500E+01 0.240E+04
 10 0 0.100E-02 0.300E+02 0.100E+01
 11 2 0.100E-04 0.500E+01
 12 2 0.300E-05 0.100E+02

2	3	2	
FAULT TREE FOR THE 480 VAC EMERGENCY BUS E-1			
ED100	LOSS OF 480 VAC EMERGENCY BUS E-1		
	+ 1 1 ED101 EBSE11YC		
ED101	SUPPLY FAULT LOSS OF 480 VAC BUS E-1		
	* 0 2 ED102 ED105		
ED102	3 0 1 SUPPLY/OPEN	FAULT LOSS OF	POWER FROM 480 NORMAL BUS No. 2
ED105	3 0 1 SUPPLY.OPEN	FAULT LOSS OF	POWER FROM 400 KW GENERATOR
EBSE11YC	0 0 2 Uncleared Overload on Emergency Bus E-1		
	0.744E+03		
	0.744E+03		
	1 0 0.100E-09	0.100E+02	0.000E+00
	2 2 0.100E-06	0.100E+02

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1 4 4
FAULT TREE FOR THE 480 VAC EMERGENCY BUS E-1
ED102 SUPPLY/OPEN FAULT LOSS OF POWER FROM 480 NORMAL BUS No.2
      + 0 4 JD530 EWR2E1YB JCB05CVO ESW004YO
JD530 3 0 1 LOSS OF 480 VAC NORMAL BUS No.2
EWR2E1YB 0 0 2 480 Vac Normal Bus No. 2 to E-1 Wiring Fails Open
JCB05CVO 0 0 3 Normal Bus No. 2 Breaker 5C Fails Open
ESW004YO 0 0 4 E-1 Bus Disconn. Switch 4 Fails Open
0.744E+03
0.744E+03
1 0 0.100E-09 0.100E+02 0.000E+00
2 2 0.100E-06 0.100E+02
3 2 0.300E-06 0.100E+02
4 2 0.100E-05 0.100E+02

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FAULT TREE FOR THE 480 VAC EMERGENCY BUS E-1
 ED105 SUPPLY.OPEN FAULT LOSS OF POWER FROM 400 KW GENERATOR
 + 3 4 ED106 ED111 ED098 EWR4SWHB EGE4EFHW ECB4OBHO EDL41OHU
 ED106 400 KW DIESEL FAILS TO START
 + 1 1 ED107 EDL4SEHS
 ED111 480 VAC BUS E-1 BREAKER 3D FAILS TO CLOSE
 + 1 4 ED112 ECB3D2YN LWR3D4BB LFU3D4BB LD500
 ED098 EXCESSIVE LOAD ON 400 KW GENERATOR
 + 1 1 ED099 ECB1XYP
 ED107 FAULT IN 400 KW DIESEL START SIGNAL CKT
 + 0 3 EIN4SSH LD500 LFU400BB
 ED112 E-1 BREAKER 3D CLOSE SIGNAL FAILS
 + 1 1 ED113 EIN3D3YW
 ED099 INTERLOCK FAILURE OVERLOAD ON 400 KW GENERATOR
 * 1 1 ED113 EIN1XYW
 ED113 NORMAL BUS NO.2 BREAKER 5C FAILS TO OPEN ON DEMAND
 + 1 1 ED114 JCBN5CVP
 ED114 NORMAL BUS NO.2 BREAKER 5C OPEN SIGNAL FAILS
 + 0 3 LD510 JFVN5CVB JINN5CVW
 LD500 3 0 1 LOSS OF 480 VAC EMERGENCY SWITCHGEAR 125 VDC BUS No.1
 LD510 3 0 1 LOSS OF 480 VAC 125 VDC BUS
 EWR4SWHB 0 0 2 400 Kw Generator Supply Wiring Fails Open
 EGE4EFHW 0 0 3 400 Kw Diesel Fails to Run
 ECB4OBHO 0 0 4 400 Kw Generator Output Breaker Fails Open
 EDL41OHU 0 0 5 400/100 Kw Diesel Fuel Supply Common Mode Failure
 EDL4SEHS 0 0 6 400 Kw Diesel Fails to Start on Command
 ECB3D2YN 0 0 7 E-1 Breaker 3D Failure to Close
 LWR3D4BB 0 0 8 E-1 Breaker 3D 125 Vdc Sup. Wiring Opens
 LFU3D4BB 0 0 9 E-1 Breaker 3D 125 Vdc Sup. Wire/Fuse Opens
 ECB1XYP 0 0 10 Selected Emergency Power Loads are not Shed
 EIN4SSH 0 0 11 400 Diesel Start Signal Ckt Fails
 LFU400BB 0 0 12 400 Kw Diesel Start Sup. Fuse Open
 EIN3D3YW 0 0 13 E-1 Breaker 3D Close Signal Ckt Fails
 EIN1XYW 0 0 14 Brkr 5C - Brkr 3D Interlock Fails and 3D Closes
 JCBN5CVP 0 0 15 Normal Bus No. 2 Breaker 5C Fails to Open
 JFVN5CVB 0 0 16 Normal Bus No. 2 Breaker 5C Trip Fuse Opens
 JINN5CVW 0 0 17 Normal Bus No. 2 Breaker 5C Trip Signal Ckt Fails
 0.744E+03
 0.744E+03

LOSS OF 480 VAC NORMAL BUS No.1	
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LOSS OF 480 VAC NORMAL BUS No.1	
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LOSS OF 480 VAC NORMAL BUS No.1
 JD500 LOSS OF 480 VAC NORMAL BUS No.1
 + 2 2 JD501 JD510 JBS4802U JTRT560U
 JD501 LOSS OF INPUT TO 480 VAC NORMAL BUS No.1
 * 2 0 JD502 JD503
 JD510 OVERLOAD FAULT LOSS OF 480 VAC NORMAL BUS No.1
 + 2 0 JD511 JD512
 JD502 LOSS OF TRANSF. No.5 POWER TO NORMAL BUS No.1
 + 0 5 J0101 JCA1312B JTR0050B JCB00220 JCB1CDVO
 JD503 LOSS OF TRANSF. No.6 POWER TO NORMAL BUS No.1
 + 1 1 JD532 JCB4BCVO
 JD511 UNCLEARED OVERLOAD ON NORMAL BUS No.1
 + 1 2 JD513 JBSN1LVC JCAN1SVC
 JD512 UNCLEARED OVERLOAD ON BUS No.2 AFFECTING NORMAL BUS No.1
 * 2 0 JD517 JD539
 JD532 LOSS OF TRANSF. No.6 POWER TO NORMAL BUS No.2
 + 0 5 J0201 JCA1322B JTR0060B JCB01320 JCB7CDVO
 JD513 UNCLEARED OVERLOAD BEFORE 1CD AFFECTING NORMAL BUS No.1
 * 1 1 JD514 JCAT50VC
 JD517 TIE BREAKER 4BC FAILS TO OPEN AUTOMATICALLY
 + 1 1 JD519 JCB4BCVP
 JD539 UNCLEARED OVERLOAD ON NORMAL BUS No.2
 + 1 2 JD541 JBSN2LVC JCAN2SVC
 JD514 BREAKER 1CD FAILS TO OPEN AUTOMATICALLY
 + 1 1 JD516 JCB1CDVP
 JD519 OVERLOAD FAULT DETECTION CKT 4BC-21 FAILS
 + 0 5 JFU417VB LFU4BCVB LWR210VB LD510 JIN210VW
 JD541 UNCLEARED OVERLOAD BEFORE 7CD AFFECTING NORMAL BUS No.2
 * 1 1 JD544 JCAT60VC
 JD516 OVERLOAD FAULT DETECTION CKT 1CD FAILS
 + 0 5 JFU417VB JIN1CDVW LFU1CDVB LMR1CDVB LD510
 JD544 BREAKER 7CD FAILS TO OPEN AUTOMATICALLY
 + 1 1 JD546 JCB7CDVP
 JD546 OVERLOAD FAULT DETECTION CKT 7CD FAILS
 + 0 5 JFU417VB JIN7CDVW LFU7CDVB LMR7CDVB LD510
 J0101 3 0 1 Loss of 13.8 kV Bus 1
 J0201 3 0 1 Loss of 13.8 kV Bus 2
 LD510 3 0 1 LOSS OF 480 VAC 125 VDC BUS
 JBS4802U 0 0 2 480 Vac Normal Bus No. 1 and Bus No. 2 Common Mode Failure
 JTRT560U 0 0 3 13.8 KV/480 Vac Transf 5 and 6 Common Mode Failure
 JCA1312B 0 0 4 Supply Wiring Fails Open, Transf. No. 5
 JTR0050B 0 0 5 Transf. No. 5 Fails Open
 JCB00220 0 0 6 13.8 KV Breaker No. 2 Fails Open
 JCB1CDVO 0 0 7 480 Vac Breaker 1CD Fails Open
 JCB4BCVO 0 0 8 Tie Breaker 4BC Fails Open
 JBSN1LVC 0 0 9 Uncleared Overload on Normal Bus No. 1 Load Side
 JCAN1SVC 0 0 10 Overload Fault Between Breaker 1CD and Normal Bus No. 1
 JCA1322B 0 0 11 Supply Wiring Fails Open, Transf. No. 6
 JTR0060B 0 0 12 Transf. No. 6 Fails Open
 JCB01320 0 0 13 13.8 KV Breaker No. 13 Fails Open
 JCB7CDVO 0 0 14 480 Vac Breaker 7CD Fails Open
 JCAT50VC 0 0 15 Overload Fault Occurs Before Breaker 1CD
 JCB4BCVP 0 0 16 Breaker 4BC Fails to Open on Demand
 JBSN2LVC 0 0 17 Uncleared Overload on Normal Bus No. 2 Load Side
 JCAN2SVC 0 0 18 Overload Fault Between Breaker 7CD and Normal Bus No. 2
 JCB1CDVP 0 0 19 Breaker 1CD Fails to Open on Demand
 JFU417VB 0 0 20 4BC, 1CD, 7CD Fault Detection Ckts Fuse Fails Open
 LFU4BCVB 0 0 21 Breaker 4BC Trip Ckt Fuse Fails Open
 LWR210VB 0 0 22 4BC-21 Fault Detection Ckt Supply W g Fails Open

JIN210VW	0	0	23	4BC-21 Fault Detection Ckt Fails	
JCAT60VC	0	0	24	Overload Fault Occurs Before Breaker 7CD	
JIN1CDVW	0	0	25	ICD Fault Detection CKT Fails	
LFU1CDVB	0	0	26	Breaker 1CD Trip Ckt Fuse Fails Open	
LWR1CDVB	0	0	27	ICD Fault Detection Ckt Supply Wiring Fails Open	
JCB7CDVP	0	0	28	Breaker 7CD Fails to Open on Demand	
JIN7CDVW	0	0	29	7CD Fault Detection Ckt Fails	
LFU7CDVB	0	0	30	Breaker 7CD Trip Ckt Fuse Fails Open	
LWR7CDVB	0	0	31	7CD Fault Detection Ckt Supply Wiring Fails Open	
0.744E+03					
0.744E+03					
1	0	0.100E-09	0.100E+02	0.000E+00	
2	2	0.100E-07	0.100E+02		
3	2	0.100E-06	0.100E+02		
4	2	0.100E-06	0.100E+02		
5	2	0.100E-05	0.100E+02		
6	2	0.300E-06	0.100E+02		
7	2	0.300E-06	0.100E+02		
8	5	0.300E-06	0.100E+02	0.876E+04	
9	2	0.100E-06	0.100E+02		
10	2	0.100E-06	0.100E+02		
11	2	0.100E-06	0.100E+02		
12	2	0.100E-05	0.100E+02		
13	2	0.300E-06	0.100E+02		
14	2	0.300E-06	0.100E+02		
15	2	0.100E-06	0.100E+02		
16	0	0.500E-03	0.100E+02	0.100E+01	
17	2	0.100E-06	0.100E+02		
18	2	0.100E-06	0.100E+02		
19	0	0.500E-03	0.100E+02	0.100E+01	
20	5	0.100E-06	0.100E+02	0.876E+04	
21	5	0.100E-06	0.100E+02	0.876E+04	
22	5	0.100E-06	0.100E+02	0.876E+04	
23	5	0.300E-05	0.100E+02	0.876E+04	
24	2	0.100E-06	0.100E+02		
25	5	0.300E-05	0.100E+02	0.876E+04	
26	5	0.100E-06	0.100E+02	0.876E+04	
27	5	0.100E-06	0.100E+02	0.876E+04	
28	0	0.500E-03	0.100E+02	0.100E+01	
29	5	0.300E-05	0.100E+02	0.876E+04	
30	5	0.100E-06	0.100E+02	0.876E+04	
31	5	0.100E-06	0.100E+02	0.876E+04

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LOSS OF 480 VAC NORMAL BUS No. 2	
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LOSS OF 480 VAC NORMAL BUS No. 2

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LOSS OF 480 VAC NORMAL BUS No. 2
 JD530 LOSS OF 480 VAC NORMAL BUS No. 2
 + 2 2 JD531 JD538 JTRT560U JBS480ZU
 JD531 LOSS OF INPUT TO 480 VAC NORMAL BUS No. 2
 * 2 0 JD532 JD533
 JD538 OVERLOAD FAULT LOSS OF 480 VAC NORMAL BUS No. 2
 + 2 0 JD539 JD540
 JD532 LOSS OF TRANSF. No. 6 POWER TO NORMAL BUS No. 2
 + 0 5 J0201 JCA1322B JTR0060B JCB01320 JCB7CDVO
 JD533 LOSS OF TRANSF. No. 5 POWER TO NORMAL BUS No. 2
 + 1 1 JD502 JCB4BCVO
 JD539 UNCLEARED OVERLOAD ON NORMAL BUS No. 2
 + 1 2 JD541 JBSN2LVC JCAN2SVC
 JD540 UNCLEARED OVERLOAD ON BUS No. 1 AFFECTING NORMAL BUS No. 2
 * 2 0 JD542 JD511
 JD502 LOSS OF TRANSF. No. 5 POWER TO NORMAL BUS No. 1
 + 0 5 J0101 JCA1312B JTR0050B JCB00220 JCB1CDVO
 JD541 UNCLEARED OVERLOAD BEFORE 7CD AFFECTING NORMAL BUS No. 2
 * 1 1 JD544 JCAT60VC
 JD542 TIE BREAKER 4BC FAILS TO OPEN AUTOMATICALLY
 + 1 1 JD543 JCB4BCVP
 JD511 UNCLEARED OVERLOAD ON NORMAL BUS No. 1
 + 1 2 JD513 JBSN1LVC JCAN1SVC
 JD544 BREAKER 7CD FAILS TO OPEN AUTOMATICALLY
 + 1 1 JD546 JCB7CDVP
 JD543 OVERLOAD FAULT DETECTION CKT 4BC-12 FAILS
 + 0 5 JFU417VB JIN120VW LFU4BCVB LWR120VB LD510
 JD513 UNCLEARED OVERLOAD BEFORE 1CD AFFECTING NORMAL BUS No. 1
 * 1 1 JD514 JCAT50VC
 JD546 OVERLOAD FAULT DETECTION CKT 7CD FAILS
 + 0 5 JFU417VB JIN7CDVW LFU7CDVB LWR7CDVB LD510
 JD514 BREAKER 1CD FAILS TO OPEN AUTOMATICALLY
 + 1 1 JD516 JCB1CDVP
 JD516 OVERLOAD FAULT DETECTION CKT 1CD FAILS
 + 0 5 JFU417VB JIN1CDVW LFU1CDVB LWR1CDVB LD510
 J0201 3 0 1 Loss of 13.8 kV Bus 2
 J0101 3 0 1 Loss of 13.8 kV Bus 1
 LD510 3 0 1 LOSS OF 480 VAC 125 VDC BUS
 JTRT560U 0 0 2 13.8 KV/480 Vac Transf 5 and 6 Common Mode Failure
 JBS480ZU 0 0 3 480 Vac Normal Bus No. 1 and Bus No. 2 Common Mode Failure
 JCA1322B 0 0 4 Supply Wiring Fails Open, Transf. No. 6
 JTR0060B 0 0 5 Transf. No. 6 Fails Open
 JCB01320 0 0 6 13.8 KV Breaker No. 13 Fails Open
 JCB7CDVO 0 0 7 480 Vac Breaker 7CD Fails Open
 JCB4BCVO 0 0 8 Tie Breaker 4BC Fails Open
 JBSN2LVC 0 0 9 Uncleared Overload on Normal Bus No. 2 Load Side
 JCAN2SVC 0 0 10 Overload Fault Between Breaker 7CD and Normal Bus No. 2
 JCA1312B 0 0 11 Supply Wiring Fails Open, Transf. No. 5
 JTR0050B 0 0 12 Transf. No. 5 Fails Open
 JCB00220 0 0 13 13.8 KV Breaker No. 2 Fails Open
 JCB1CDVO 0 0 14 480 Vac Breaker 1CD Fails Open
 JCAT60VC 0 0 15 Overload Fault Occurs Before Breaker 7CD
 JCB4BCVP 0 0 16 Breaker 4BC Fails to Open on Demand
 JBSN1LVC 0 0 17 Uncleared Overload on Normal Bus No. 1 Load Side
 JCAN1SVC 0 0 18 Overload Fault Between Breaker 1CD and Normal Bus No. 1
 JCB7CDVP 0 0 19 Breaker 7CD Fails to Open on Demand
 JFU417VB 0 0 20 4BC, 1CD, 7CD Fault Detection Ckts Fuse Fails Open
 JIN120VW 0 0 21 4BC-12 Fault Detection Ckt Fails
 LFU4BCVB 0 0 22 Breaker 4BC Trip Ckt Fuse Fails Open

LWR120VB	0	0	23	4BC-12 Fault Detection Ckt Supply Wiring Fails Open	
JCA150VC	0	0	24	Overload Fault Occurs Before Breaker LCD	
JIN7CDVW	0	0	25	7CD Fault Detection Ckt Fails	
LFU7CDVB	0	0	26	Breaker 7CD Trip Ckt Fuse Fails Open	
LWR7CDVB	0	0	27	7CD Fault Detection Ckt Supply Wiring Fails Open	
JCB1CDVP	0	0	28	Breaker LCD Fails to Open on Demand	
JIN1CDVW	0	0	29	ICD Fault Detection Ckt Fails	
LFU1CDVB	0	0	30	Breaker LCD Trip Ckt Fuse Fails Open	
LWR1CDVB	0	0	31	ICD Fault Detection Ckt Supply Wiring Fails Open	
					0.744E+03
					0.744E+03
1	0	0.100E-09	0.100E+02	0.000E+00	
2	2	0.100E-06	0.100E+02		
3	2	0.100E-07	0.100E+02		
4	2	0.100E-06	0.100E+02		
5	2	0.100E-05	0.100E+02		
6	2	0.300E-06	0.100E+02		
7	2	0.300E-06	0.100E+02		
8	5	0.300E-06	0.100E+02	0.876E+04	
9	2	0.100E-06	0.100E+02		
10	2	0.100E-06	0.100E+02		
11	2	0.100E-06	0.100E+02		
12	2	0.100E-05	0.100E+02		
13	2	0.300E-06	0.100E+02		
14	2	0.300E-06	0.100E+02		
15	2	0.100E-06	0.100E+02		
16	0	0.500E-03	0.100E+02	0.100E+01	
17	2	0.100E-06	0.100E+02		
18	2	0.100E-06	0.100E+02		
19	0	0.500E-03	0.100E+02	0.100E+01	
20	5	0.100E-06	0.100E+02	0.876E+04	
21	5	0.300E-05	0.100E+02	0.876E+04	
22	5	0.100E-06	0.100E+02	0.876E+04	
23	5	0.100E-06	0.100E+02	0.876E+04	
24	2	0.100E-06	0.100E+02		
25	5	0.300E-05	0.100E+02	0.876E+04	
26	5	0.100E-06	0.100E+02	0.876E+04	
27	5	0.100E-06	0.100E+02	0.876E+04	
28	0	0.500E-03	0.100E+02	0.100E+01	
29	5	0.300E-05	0.100E+02	0.876E+04	
30	5	0.100E-06	0.100E+02	0.876E+04	
31	5	0.100E-06	0.100E+02	0.876E+04	

LIN12MBW	0	0	19	First 125 VDC Ground Fault Alarm CK	11s
LIN12MBZ	0	0	20	Operator Error Ignores First 125 VDC Ground Fault Alarm	
URAUP1BW	0	0	21	Charged 135 VDC UPS 1 Battery Fails Suddenly	
UCBU11YO	0	0	22	135 VDC UPS 1 Battery Output Ckt Brkr Fails Open	
UBYUP1YW	0	0	23	UPS 1 Charger Circuit Fails	
ECB3EEYO	0	0	24	Bus P-4E Breaker 3EE Fails Open	
EMRP4UYB	0	0	25	Line From Bus P-4E to UPS 1 Fails	
UINV1BYL	0	0	26	135 VDC UPS 1 Battery Capacity Loss Not Detected	
URAUP1BW	0	0	27	135 VDC UPS 1 Battery Loses Capacity	
UBYUP1BW	0	0	28	135 VDC UPS 1 Battery Has Low Charge Rate	
UINV1BYW	0	0	29	135 VDC UPS 1 Battery Low Charge Rate Not Detected	
				0.744E+03	
				0.744E+03	
	1	0	0.100E-09	0.100E+02	0.000E+00
	2	2	0.100E-06	0.100E+02	
	3	2	0.100E-06	0.100E+02	
	4	2	0.100E-06	0.100E+02	
	5	2	0.100E-04	0.500E+01	
	6	2	0.100E-06	0.100E+02	
	7	2	0.100E-05	0.100E+02	
	8	T	0.100E+01	0.100E+01	
	9	2	0.100E-06	0.100E+02	
	10	F	0.000E+00	0.100E+01	
	11	2	0.100E-06	0.100E+02	
	12	2	0.100E-04	0.500E+01	
	13	0	0.100E-02	0.300E+02	0.100E+01
	14	2	0.100E-04	0.500E+01	
	15	2	0.300E-05	0.100E+02	
	16	5	0.100E-06	0.100E+02	0.876E+04
	17	5	0.100E-05	0.100E+02	0.876E+04
	18	0	0.100E-03	0.100E+02	0.100E+01
	19	2	0.300E-05	0.100E+02	
	20	0	0.100E-03	0.100E+02	0.100E+01
	21	5	0.100E-04	0.500E+01	0.240E+04
	22	2	0.100E-06	0.100E+02	
	23	2	0.100E-04	0.500E+01	
	24	2	0.300E-06	0.100E+02	
	25	2	0.100E-06	0.100E+02	
	26	0	0.100E-02	0.300E+02	0.100E+01
	27	5	0.100E-04	0.500E+01	0.240E+04
	28	2	0.100E-04	0.500E+01	
	29	2	0.300E-05	0.100E+02	
	30	0	0.100E-09	0.100E+01	0.000E+00

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FAULT TREE FOR THE LOSS OF THE 125 VDC	
EMERGENCY BUS	
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1 5 6
FAULT TREE FOR THE LOSS OF THE 125 VDC EMERGENCY BUS
LD200 LOSS OF 125 VDC EMERGENCY BUS
+ 0 5 E0120DUM ECB2FFYO EWRP4EYB LDC12EBW LBS12EBC
E0120DUM 0 0 6 LOSS OF 480 VAC EMERGENCY BUS P-4E
ECB2FFYO 0 0 2 Bus P-4E Breaker 2F Fails Open
EWRP4EYB 0 0 3 Line From P-4E to 10 KW Rectifier Fails
LDC12EBW 0 0 4 10 KW Rectifier Fails
LBS12EBC 0 0 5 Uncleared Overload on 125 VDC Emergency Bus
0.744E+03
0.744E+03
1 0 0.100E-09 0.100E+02 0.000E+00
2 5 0.300E-06 0.100E+02 0.876E+04
3 5 0.100E-06 0.100E+02 0.876E+04
4 5 0.100E-04 0.500E+01 0.876E+04
5 5 0.100E-06 0.100E+02 0.876E+04
6 0 0.100E-09 0.100E+01 0.000E+00

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LOSS OF THE 480 VAC EMERGENCY SWITCHGEAR	
125 VDC BUS No.1	
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LOSS OF THE 480 VAC EMERGENCY SWITCHGEAR 125 VDC BUS No. 1
 LD500 LOSS OF 480 VAC EMERGENCY SWITCHGEAR 125 VDC BUS No. 1
 + 1 3 LD502 LMRMT2BB LBSB1BC LBS003BU
 LD502 LOSS OF 125 VDC TO 125 VDC/480 VAC EMERG BUS 1
 * 2 0 LD504 LD506
 LD504 LOSS OF 125 VDC TO 125/480 EMERG BUS 1 FROM DISTR CT
 + 0 4 LD100 LCB10BO LSWMD2BO LWRMD2BB
 LD506 LOSS OF 125 VDC TO 125/480 EMER BUS 1 FROM EMER 125 VDC
 + 1 1 LD505 RO099A00
 LD505 LOSS OF EMERG 125 VDC TO 480 VAC EMERG BUSES (ALT. OPS.)
 * 1 1 LD503 RO099B00
 LD503 LOSS OF EMERG 125 VDC TO 480 VAC EMERG BUSES
 + 0 5 LD200 LCB12BO LSWME2BO LWRME2BB LSWME2BZ
 LD100 3 0 1 LOSS OF 125 VDC DISTRIBUTION CENTER BUS
 LD200 3 0 1 LOSS OF 125 VDC EMERGENCY BUS
 LMRMT2BB 0 0 2 Wiring From 125 VDC MTS 2 to Bus Falls Open
 LBSB1BC 0 0 3 Uncleared Overload on 125 VDC/480 Vac Emerg Bus No. 1
 LBS003BU 0 0 4 125 VDC/480 Vac Emerg Bus Common Mode Loss of Power Sources
 LCB10BO 0 0 5 125 VDC Distr Center Bus Breaker 10 Falls Open
 LSWMD2BO 0 0 6 125 VDC MTS 2 Distr Cent Bus Side Falls Open
 LWRMD2BB 0 0 7 Wiring From 125 VDC Distr Cent Bus to MTS 2 Falls
 RO099A00 2 0 8 Reactor Operation Permitted
 RO099B00 2 0 9 Reactor Operation Not Permitted
 LCB12BO 0 0 10 125 VDC Emergency Bus Breaker 2 Falls Open
 LSWME2BO 0 0 11 125 VDC MTS 2 Emergency Bus Side Falls Open
 LWRME2BB 0 0 12 Wiring From 125 VDC Emergency Bus to MTS 2 Falls
 LSWME2BZ 0 0 13 Operator Falls to Close 125 VDC MTS 2 Emerg Bus Side
 0.744E+03
 0.744E+03
 1 0 0.100E-09 0.100E+02 0.000E+00
 2 2 0.100E-06 0.100E+02
 3 2 0.100E-06 0.100E+02
 4 2 0.100E-06 0.100E+02
 5 2 0.300E-06 0.100E+02
 6 2 0.100E-05 0.100E+02
 7 2 0.100E-06 0.100E+02
 8 T 0.100E+01 0.100E+01
 9 F 0.000E+00 0.100E+01
 10 5 0.300E-06 0.100E+02 0.876E+04
 11 5 0.100E-05 0.100E+02 0.876E+04
 12 5 0.100E-06 0.100E+02 0.876E+04
 13 0 0.100E-03 0.100E+02 0.100E+01

FAULT TREE FOR THE LOSS OF THE 480 VAC	
NORMAL POWER 125 VDC BUS	
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6 14 13
 FAULT TREE FOR THE LOSS OF THE 480 VAC NORMAL POWER 125 VDC BUS
 LD510 LOSS OF 480 VAC 125 VDC BUS
 + 1 3 LD512 LWRMT1BB LBS124BC LBS002BU
 LD512 LOSS OF 125 VDC TO 125 VDC/480 VAC BUS
 * 2 0 LD514 LD516
 LD514 LOSS OF 125 VDC TO 125 VDC/480 VAC BUS FROM DISTR CENT BUS
 + 0 4 LD100 LCB11BO LSWMD1BO LWRMD1BB
 LD516 LOSS OF 125 VDC TO 125 VDC/480 VAC BUS FROM EMERG 125 VDC
 + 1 1 LD515 RO099A00
 LD515 LOSS OF EMERG 125 VDC TO 480 VAC NORM BUS (ALT. OPS.)
 * 1 1 LD513 RO099B00
 LD513 LOSS OF EMERG 125 VDC TO 480 VAC NORM BUS
 + 0 5 LD200 LCB11BO LSWME1BO LWRME1BB LSWME1BZ
 LD100 3 0 1 LOSS OF 125 VDC EMERGENCY BUS
 LD200 3 0 1 LOSS OF 125 VDC EMERGENCY BUS
 LWRMT1BB 0 0 2 Wiring From 125 VDC MTS 1 to Bus Fails Open
 LBS124BC 0 0 3 Uncleared Overload on 125 VDC/480 Vac Bus
 LBS002BU 0 0 4 125 VDC/480 Vac Norm Bus Common Mode Loss of Power Sources
 LCB11BO 0 0 5 125 VDC Distr Center Bus Breaker 11 Fails Open
 LSWMD1BO 0 0 6 125 VDC MTS 1 Distr Cent Bus Side Fails Open
 LWRMD1BB 0 0 7 Wiring From 125 VDC Distr Cent Bus to MTS 1 Fails
 RO099A00 2 0 8 Reactor Operation Permitted
 RO099B00 2 0 9 Reactor Operation Not Permitted
 LCB11BO 0 0 10 125 VDC Emergency Bus Breaker 1 Fails Open
 LSWME1BO 0 0 11 125 VDC MTS 1 Emergency Bus Side Fails Open
 LWRME1BB 0 0 12 Wiring from 125 VDC Emergency Bus to MTS 1 Fails
 LSWME1BZ 0 0 13 Operator Fails to Close 125 VDC MTS 1 Emerg Bus Side
 0.744E+03
 0.744E+03
 1 0 0.100E-09 0.100E+02 0.000E+00
 2 2 0.100E-06 0.100E+02
 3 2 0.100E-06 0.100E+02
 4 2 0.100E-06 0.100E+02
 5 2 0.300E-06 0.100E+02
 6 2 0.100E-05 0.100E+02
 7 2 0.100E-06 0.100E+02
 8 T 0.100E+01 0.100E+01
 9 F 0.000E+00 0.100E+01
 10 5 0.300E-06 0.100E+02 0.876E+04
 11 5 0.100E-05 0.100E+02 0.876E+04
 12 5 0.100E-06 0.100E+02 0.876E+04
 13 0 0.100E-03 0.100E+02 0.100E+01

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FAULT TREE FOR THE LOSS OF THE 480 VAC	
EMERGENCY BUS P-1B	
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FAULT TREE FOR THE LOSS OF THE 480 VAC	
EMERGENCY BUS P-1B	
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FAULT TREE FOR THE LOSS OF THE 480 VAC	
EMERGENCY BUS P-1B	
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FAULT TREE FOR THE LOSS OF THE 480 VAC EMERGENCY BUS P-1B
 E0200L LOSS OF 480 VAC EMERGENCY BUS P-1B
 + 1 1 E0202L OBSPIBYC
 E0202L LOSS OF 480 VAC EMERGENCY BUS P-1B POWER SOURCES
 * 2 0 E0210L E0204L
 E0210L LOSS OF P-1B POWER FROM NORMAL BUS No.2
 + 1 2 E0212L OWR2E1YC E0102L
 E0204L LOSS OF P-1B POWER FROM P-1A AND 400 KW GENERATOR
 + 2 0 E0206L E0208L
 E0212L LOSS OF E-1 TO P-1B LINE
 + 1 2 E0214L OWRP1BYC OBSE11YC
 E0206L LOSS OF P-1B, P-1A PWR FAILS BEFORE OR WITH NORMAL BUS No.2
 * 1 1 E0220L J0565L
 E0208L LOSS OF P-1B, P-1A POWER FAILS AFTER NORMAL BUS No.1
 * 2 0 E0222L E0230L
 E0214L OPEN FAULT ON E-1 TO P-1B LINE
 + 0 3 OWR12BYB OCB2C8YO OCB16YO
 E0220L 400 KW GEN. POWER TO P-1B FAILS, AUTO CONNECT MODE
 + 1 1 E0212L E0105L
 E0222L P-1A PWR FAILS AFTER POWER FAILURE FROM NORMAL BUS No.2
 + 1 1 E0224L J0565L
 E0230L 400 KW GEN. PWR TO P-1B FAILS, MANUAL CONNECT MODE
 + 1 2 E0212L E0105L OCB1PBYZ
 E0224L BUS P-1 BREAKER 9A FAILS TO CLOSE OR TRIPS OPEN
 + 3 1 E0226L E0234L E0238L OCBT9AYO
 E0226L BUS P-1 BREAKER 9A FAILS TO CLOSE
 + 1 1 E0228L OCB19AYN
 E0234L BUS P-1 BREAKER 16A FAILS TO OPEN
 + 1 1 E0236L OCB16YP
 E0238L BUS E-1 BREAKER 2C FAILS TO OPEN
 + 1 1 E0240L OCB2CAYP
 E0228L BUS P-1 BREAKER 9A CLOSE SIGNAL FAILS
 + 0 4 L0530L OFUD9ABB OIND9AYW OWRD9AYB
 E0236L BUS P-1 BREAKER 16A OPEN SIGNAL FAILS
 + 0 4 L0530L OFU16PBB OINI6PYW OWR16PYB
 E0240L BUS E-1 BREAKER 2C OPEN SIGNAL FAILS
 + 0 4 L0520L OFU12CBB OINI2CYW OWR12CYB
 E0102L 3 0 1 SUPPLY/OPEN FAULT LOSS OF POWER FROM 480 NORMAL BUS No.2
 J0565L 3 0 1 LOSS OF 480 VAC NORMAL BUS P-1A
 E0105L 3 0 1 SUPPLY.OPEN FAULT LOSS OF POWER FROM 400 KW GENERATOR
 L0530L 3 0 1 LOSS OF 125 VDC MCC BUS
 L0520L 3 0 1 LOSS OF 480 VAC EMERGENCY SWITCHGEAR 125 VDC BUS No. 2
 OBSPIBYC 0 0 2 Uncleared Overload on Emergency Bus P-1B
 OWR2E1YC 0 0 3 Overload on Normal Bus No. 2 to E-1 Line
 OWRP1BYC 0 0 4 Overload Fault on E-1 to P-1B Line
 OBSE11YC 0 0 5 Uncleared Overload on Emergency Bus E-1
 OWR12BYB 0 0 6 Bus E-1 Breaker 2B Trip Ckt Sup. Wiring Fails Open
 OCB2C8YO 0 0 7 Emergency Bus E-1 Breaker 2C Fails Open
 OCB16YO 0 0 8 Emergency Bus P-1B Breaker 16A Fails Open
 OCB1PBYZ 0 0 9 Operator Error in Connecting E-1 to P-1B
 OCBT9AYO 0 0 10 Bus P-1 Breaker 9A Trips Open
 OCB19AYN 0 0 11 Bus P-1 Breaker 9A Fails to Close on Demand
 OCB16PY 0 0 12 Bus P-1 Breaker 16A Fails to Open on Demand
 OCB2CAYP 0 0 13 Bus E-1 Breaker 2C Fails to Open on Demand
 OFUD9ABB 0 0 14 Bus P-1 Breaker 9A Close Fuse Fails Open
 OFUD9AYW 0 0 15 Bus P-1 Breaker 9A Close Signal Ckt Fails
 OWRD9AYB 0 0 16 Bus P-1 Breaker 9A Close Ckt Sup. Wiring Fails Open
 OFU16PBB 0 0 17 Bus P-1 Breaker 16A Trip Fuse Opens
 OINI6PYW 0 0 18 Bus P-1 Breaker 16A Trip Signal Fails

OWR16PYB 0 0	19 Bus P-1 Breaker 16A Trip Ckt Sup.	ng Fails Open
OFU12CBB 0 0	20 Bus E-1 Breaker 2C Trip Fuse Opens	
OIN12CYW 0 0	21 Bus E-1 Breaker 2C Trip Signal Ckt Fails	
OWR12CYB 0 0	22 Bus E-1 Breaker 2C Trip Ckt Sup. Wiring Fails Open	
0.336E+03		
0.336E+03		
1 0 0.100E-09	0.100E+02	0.000E+00
2 0 0.336E-04	0.100E+02	
3 0 0.336E-04	0.100E+02	
4 0 0.336E-04	0.100E+02	
5 0 0.336E-04	0.100E+02	
6 0 0.336E-04	0.100E+02	
7 0 0.101E-03	0.100E+02	
8 0 0.101E-03	0.100E+02	
9 0 0.100E-03	0.100E+02	0.100E+01
10 0 0.101E-03	0.100E+02	
11 0 0.500E-03	0.100E+02	0.100E+01
12 0 0.500E-03	0.100E+02	0.100E+01
13 0 0.500E-03	0.100E+02	0.100E+01
14 0 0.438E-03	0.100E+02	
15 0 0.130E-01	0.100E+02	
16 0 0.438E-03	0.100E+02	
17 0 0.438E-03	0.100E+02	
18 0 0.130E-01	0.100E+02	
19 0 0.438E-03	0.100E+02	
20 0 0.438E-03	0.100E+02	
21 0 0.130E-01	0.100E+02	
22 0 0.438E-03	0.100E+02

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Instrument Air Distribution System	
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Instrument Air Distribution System		
I0001	Loss of Instrument Air System Output	
	+ 1 2 I0102 IPP001YF IPP201YF	
I0102	Loss of Input to Inst. Air Distribution Line	
	+ 1 1 I0110 IPP132YF	
I0110	Loss of Output from Inst. Air Secondary Filter System	
	+ 1 1 I0131 IPP122YF	
I0131	Loss of Output from Instrument Air Dryer System	
	+ 1 1 I0160 IRV001YN	
I0160	Loss of Input to Instrument Air Dryer System	
	+ 1 1 I0161 IPP117YF	
I0161	Loss of Output from Main Instrument Air Filter System	
	+ 2 1 I0182 I0184 IPP113YF	
I0182	Loss of Instrument Air Supply, Emergency Crossover Open	
	* 1 3 I0190 I0310 IOO01A00 IOO02Z00	
I0184	Loss of Instrument Air Supply, Emergency Crossover Closed	
	* 1 1 I0190 IOO01B00	
I0190	Loss of Output from Instrument Air Receiver	
	+ 0 3 IO200 IRV003YN IRR719YF	
I0310	3 0 1 Loss of Output from Plant Air Filter System	
I0200	3 0 1 Loss of Input to Instrument Air Receiver	
IPP001YF	0 0 2 Main Instrument Air System Distribution Line Failure	
IPP201YF	0 0 3 Secondary Instrument Air Distribution Line Leakage	
IPP132YF	0 0 4 Sec. Filter to Instrument Air Distribution Line Failure	
IPP122YF	0 0 5 Instrument Air Dryer to Filter Line Failure	
IRV001YN	0 0 6 Instrument Air Dryer Relief Valve Fails to Close	
IPP117YF	0 0 7 Instrument Air Main Filter to Dryer Line Failure	
IPP113YF	0 0 8 Instrument Air Receiver to Main Filter Line Failure	
IOO01A00	2 0 9 Plant/Instrument Air Emergency Crossover Open	
IOO02Z00	2 0 10 Instrument Air System Partially Isolated	
IOO01B00	2 0 11 Plant/Instrument Air Emergency Crossover Closed	
IRV003YN	0 0 12 Instrument Air Receiver Relief Valve Fails to Close	
IRR719YF	0 0 13 Instrument Air Receiver P3-T-719 Tank Failure	
0.744E+03		
0.744E+03		
1 0	0.100E-09	0.100E+02 0.000E+00
2 2	0.800E-08	0.300E+02
3 2	0.800E-08	0.300E+02
4 2	0.800E-08	0.300E+02
5 2	0.800E-08	0.300E+02
6 0	0.300E-02	0.500E+01 0.000E+00
7 2	0.800E-08	0.300E+02
8 2	0.800E-08	0.300E+02
9 T	0.100E+01	0.100E+01
10 T	0.100E+01	0.100E+01
11 F	0.000E+00	0.100E+01
12 0	0.300E-02	0.500E+01 0.000E+00
13 2	0.100E-09	0.300E+02

Loss of Input to Instrument Air Receiver	
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Loss of Input to Instrument Air Receiver	
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Loss of Input to Instrument Air Receiver	
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Loss of Input to Instrument Air Receiver	
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Loss of Input to Instrument Air Receiver
 I0200 Loss of Input to Instrument Air Receiver
 * 2 0 I0201 I0231
 I0201 Loss of Output from Primary Compressor System
 + 1 1 I0203 IPP111YF
 I0231 Loss of Output from Standby Compressor System
 + 1 1 I0233 IPP110YF
 I0203 Loss of Flow through Primary Inst. Air Aftercooler
 + 1 1 I0205 IRV004YN
 I0233 Loss of Flow through Standby Inst. Air Aftercooler
 + 1 1 I0235 IRV006YN
 I0205 Loss of Input to Primary Inst. Air Aftercooler
 + 1 2 I0207 IPP718YF IRV005YN
 I0235 Loss of Input to Standby Inst. Air Aftercooler
 + 1 2 I0237 IPP720YF IRV007YN
 I0207 Loss of Primary Inst. Air Compressor Output
 + 3 1 I0209 I0210 I0211 ICM001YW
 I0237 Loss of Standby Inst. Air Compressor Output
 + 3 2 I0238 I0239 I0240 ICM002YW ICM002YX
 I0209 Loss of 480 V Power to Primary Inst. Air Compressor
 + 0 2 E0120 ECB02EYO
 I0210 Primary Inst. Air Compressor Controller Failure
 + 1 4 I0212 IXT001YW ICZ011YO ICZ012YO ICZ190YO
 I0211 Loss of Cooling Water to Primary Inst. Air Compressor
 + 0 2 W0001 WPP211YF
 I0238 Loss of Cooling Water to Standby Inst. Air Compressor
 + 0 2 W0001 WPP238YF
 I0239 Loss of 480 V Power to Standby Inst. Air Compressor
 + 0 2 E0120 ECB03CYO
 I0240 Standby Inst. Air Compressor Controller Failure
 + 1 4 I0242 IXT002YW ICZ021YO ICZ022YO ICZ290YO
 I0212 Primary Inst. Air Controller RUN Failure
 + 0 4 IPR011YB IPR012YB ITE011YB ITE012YB
 I0242 Standby Inst. Air Controller START/RUN Failure
 + 0 5 IMS002YS IPR021YB IPR022YB ITE021YB ITE022YB
 E0120 3 0 1 LOSS OF 480 VAC EMERGENCY BUS P-4E
 W0001 3 0 1 Loss of Reactor Auxiliaries Cooling Water
 IPP111YF 0 0 2 Primary Aftercooler to Receiver Line Failure
 IPP110YF 0 0 3 Standby Aftercooler to Receiver Line Failure
 IRV004YN 0 0 4 Primary Separator Relief Valve Fails to Close
 IRV006YN 0 0 5 Plant Air Receiver Relief Valve Fails to Close
 IPP718YF 0 0 6 Primary Compressor to Aftercooler Line Failure
 IRV005YN 0 0 7 Primary Compressor Relief Valve Fails to Close
 IPP720YF 0 0 8 Standby Compressor to Aftercooler Line Failure
 IRV007YN 0 0 9 Standby Compressor Relief Valve Fails to Close
 ICM001YW 0 0 10 Primary Inst. Air Compressor, Aftercooler or Motor Failure
 ICM002YW 0 0 11 Standby Inst. Air Compressor, Aftercooler or Motor Failure
 ICM002YX 0 0 12 Standby Inst. Air Compressor Down for Maintenance
 ECB02EYO 0 0 13 Breaker 2E of MCC P4E Fails Open
 IXT001YW 0 0 14 Primary Comp. 480/120 V Transformer Fails
 ICZ011YO 0 0 15 Primary Cont. Main Switch MAN Contacts Fail Open
 ICZ012YO 0 0 16 Primary Cont. Overload Contacts (1/3) Fail Open
 ICZ190YO 0 0 17 Primary Cont. Comp. Motor Contacts Fail Open
 WPP211YF 0 0 18 Primary Air Compressor Cooling Water Line Failure
 WPP238YF 0 0 19 Standby Air Compressor Cooling Water Line Failure
 ECB03CYO 0 0 20 Breaker 3C of MCC P4E Fails Open
 IXT002YW 0 0 21 Standby Comp. 480/120 V Transformer Fails
 ICZ021YO 0 0 22 Standby Cont. Main Switch AUTO Contacts Fail Open
 ICZ022YO 0 0 23 Standby Cont. Overload Contacts (1/3) Fail Open

ICZ290YO	0	0	24 Standby Cont. Comp. Motor Contacts Fail Open	
IPR011YB	0	0	25 Primary Compressor Oil Pressure Sensor Fails Low	
IPR012YB	0	0	26 Primary Comp. Controller Pressure Regulator Fails	
ITE011YB	0	0	27 Primary Comp. Water Temp. Sensor Fails High	
ITE012YB	0	0	28 Primary Compressor Air Temp. Sensor Fails High	
IMS002YS	0	0	29 Standby Compressor Motor Fails to Start	
IPR021YB	0	0	30 Standby Compressor Oil Pressure Sensor Fails Low	
IPR022YB	0	0	31 Standby Comp. Controller Pressure Regulator Fails	
ITE021YB	0	0	32 Standby Comp. Water Temp. Sensor Fails High	
ITE022YB	0	0	33 Standby Compressor Air Temp. Sensor Fails High	
0.744E+03				
0.744E+03				
1	0	0.100E-09	0.100E+02	0.000E+00
2	2	0.800E-08	0.300E+02	
3	1	0.800E-08	0.300E+02	0.168E+03
4	0	0.300E-02	0.500E+01	0.000E+00
5	0	0.300E-02	0.500E+01	0.000E+00
6	2	0.800E-08	0.300E+02	
7	0	0.300E-02	0.500E+01	0.000E+00
8	1	0.800E-08	0.300E+02	0.168E+03
9	0	0.300E-02	0.500E+01	0.000E+00
10	2	0.300E-04	0.100E+02	
11	1	0.300E-04	0.100E+02	0.168E+03
12	0	0.600E-02	0.100E+02	0.000E+00
13	2	0.300E-06	0.100E+02	
14	2	0.100E-05	0.100E+02	
15	2	0.100E-06	0.100E+02	
16	2	0.300E-06	0.100E+02	
17	2	0.100E-06	0.100E+02	
18	2	0.800E-08	0.300E+02	
19	1	0.800E-08	0.300E+02	0.168E+03
20	1	0.300E-06	0.100E+02	0.168E+03
21	1	0.100E-05	0.100E+02	0.168E+03
22	1	0.100E-06	0.100E+02	0.168E+03
23	1	0.300E-06	0.100E+02	0.168E+03
24	1	0.100E-06	0.100E+02	0.168E+03
25	2	0.100E-05	0.100E+02	
26	2	0.100E-05	0.100E+02	
27	2	0.300E-05	0.100E+02	
28	2	0.300E-05	0.100E+02	
29	0	0.300E-04	0.100E+02	0.000E+00
30	1	0.100E-05	0.100E+02	0.168E+03
31	1	0.100E-05	0.100E+02	0.168E+03
32	1	0.300E-05	0.100E+02	0.168E+03
33	1	0.300E-05	0.100E+02	0.168E+03

Plant Air Distribution System	
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2	3	3	
			Plant Air Distribution System
I0301			Loss of Plant Air Distribution
	+ 1	I0304	IPP301YF
I0304			Loss of Input to Plant Air Distribution Line
	+ 0	2 I0310	IPP012YF
I0310	3	0	1 Loss of Output from Plant Air Filter System
IPP301YF	0	0	2 Main Plant Air Distribution Line Failure
IPP012YF	0	0	3 Plant Air Filter to Distribution Line Failure
			0.744E+03
			0.744E+03
	1	0	0.100E-09 0.100E+02 0.000E+00
	2	2	0.800E-08 0.300E+02
	3	2	0.800E-08 0.300E+02

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Plant Air Distribution System		
I0310	Loss of Output from Plant Air Filter System	
	+ 1 I0331 IPF731YF	
I0331	Loss of Output from Plant Air Dryer	
	+ 1 I0381 IPP005YF	
I0381	Loss of Output from Plant Air Receiver System	
	+ 2 I0382 I0400	
I0382	Loss of Flow through Plant Air Receiver P3-T-717	
	+ 0 2 IRV006YN IRR717YF	
I0400	Loss of Input to Plant Air Receiver	
	+ 1 2 I0403 I0200 IPP004YF	
I0403	Loss of Output from Plant Air Aftercooler P3-HE-716	
	+ 1 I0405 IRV009YN	
I0405	Loss of Input to Plant Air Aftercooler	
	+ 1 I0407 IPF716YF	
I0407	Loss of Plant Air Compressor Output	
	+ 3 2 I0410 I0420 ICM003YW ICM003YX	
I0410	Loss of Cooling Water to Plant Air Compressor	
	+ 0 2 W0001 WEP410YF	
I0420	Loss of 480 Vac Normal Electrical Power	
	+ 0 2 J0560 JCB11AYO	
I0421	Plant Air Compressor Controller Failure	
	+ 1 4 I0422 IXT003YW ICZ031YO ICZ032YO ICZ092YO	
I0422	Plant Air Controller START/RUN Failure	
	+ 0 4 ITE031YB IPR031YB IPR032YB IMS003YS	
I0200	3 0 1 Loss of Input to Instrument Air Receiver	
W0001	3 0 1 Loss of Reactor Auxiliaries Cooling Water	
J0560	3 0 1 LOSS OF 480 VAC NORMAL BUS P-2B	
IPF731YF	0 0 2 Plant Air Dryer to Filter Line Failure	
IPP005YF	0 0 3 Plant Air Receiver to Dryer Line Failure	
IRV006YN	0 0 4 Plant Air Receiver Relief Valve Fails to Close	
IRR717YF	0 0 5 Plant Air Receiver P3-T-717 Tank Failure	
IPF716YF	0 0 6 Plant Air Aftercooler to Receive Line Failure	
IRV009YN	0 0 7 Plant Air Moisture Separator Relief Valve Fails to Close	
IPF716YF	0 0 8 Plant Air Compressor to Aftercooler Line Failure	
ICM003YW	0 0 9 Plant Air Compressor or Motor Failure	
ICM003YX	0 0 10 Plant Air Compressor Down for Maintenance	
WEP410YF	0 0 11 Plant Air Compressor Cooling Water Line Failure	
JCB11AYO	0 0 12 Breaker 11A of MCC P2B Fails Open	
IXT003YW	0 0 13 Controller 480/120 V Transformer Fails	
ICZ031YO	0 0 14 Controller Main Switch Contacts Fail Open	
ICZ032YO	0 0 15 Controller Overload Contacts (1/2) Fail Open	
ICZ092YO	0 0 16 Controller Compressor Motor Contacts Fail Open	
ITE031YB	0 0 17 Compressor Water Temp. Sensor Fails High	
IPR031YB	0 0 18 Compressor Oil Pressure Sensor Fails Low	
IPR032YB	0 0 19 Plant Air Controller Pressure Regulator Fails	
IMS003YS	0 0 20 Plant Air Compressor Motor Fails to Start	
	0.744E+03	
	0.744E+03	
	1 0 0.100E-09 0.100E+02 0.000E+00	
	2 2 0.800E-08 0.300E+02	
	3 2 0.800E-08 0.300E+02	
	4 0 0.300E-02 0.500E+01 0.000E+00	
	5 2 0.100E-09 0.300E+02	
	6 2 0.800E-08 0.300E+02	
	7 0 0.300E-02 0.500E+01 0.000E+00	
	8 2 0.800E-08 0.300E+02	
	9 2 0.600E-05 0.100E+02	
	10 0 0.600E-02 0.100E+02 0.000E+00	

11	2	0.800E-08	0.300E+02
12	2	0.300E-06	0.100E+02
13	2	0.100E-05	0.300E+01
14	2	0.100E-06	0.100E+02
15	2	0.200E-06	0.100E+02
16	2	0.100E-06	0.100E+02
17	2	0.300E-05	0.100E+02
18	2	0.100E-05	0.100E+02
19	2	0.100E-05	0.100E+02
20	0	0.300E-04	0.100E+02
			0.000E+00

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Emergency Air Distribution System			
I0501	Loss of Emergency Air System Output		
	+ 1 1 I0510 IPP501LF		
I0510	Loss of Air Input to Distribution Line		
	+ 1 1 I0525 IPP125LF		
I0525	Loss of Output from Emergency Air System Receiver		
	+ 1 2 I0530 IRV013LN IRR635LF		
I0530	Loss of Input to Emergency Air Receiver		
	+ 1 1 I0535 IPP634LF		
I0535	Loss of Emergency Air Compressor Output		
	+ 2 2 I0540 I0545 ICM004LW ICM004LX		
I0540	Loss of 480 V Emergency Power		
	+ 0 2 E0300 ECB3RAYO		
I0545	Emergency Air Compressor Controller Failure		
	+ 1 4 I0550 IXT004LW ICZ042LO ICZ041LO ICZ094LO		
I0550	Emergency Air Controller START/RUN Failure		
	+ 0 3 IPR004LB IPR042LB IMS004LS		
E0300	3 0 1 LOSS OF 480 VAC EMERGENCY BUS R-2A		
IPP501LF	0 0 2 Emergency Air System Distribution Line Failure		
IPP125LF	0 0 3 Emergency Air Filter/Dryer Bypass Line Failure		
IRV013LN	0 0 4 Emergency Air Receiver Relief Valve Fails to Close		
IRR635LF	0 0 5 Emergency Air Receiver R1-T-1635 Tank Failure		
IPP634LF	0 0 6 Emergency Air Compressor to Receiver Line Failure		
ICM004LW	0 0 7 Emergency Air Compressor or Motor Failure		
ICM004LX	0 0 8 Emergency Air Compressor Down for Maintenance		
ECB3RAYO	0 0 9 Breaker 3AA of MCC-R2A Fails Open		
IXT004LW	0 0 10 Controller 480/120 V Transformer Fails		
ICZ042LO	0 0 11 Controller Overload Contacts (1/2) Fail Open		
ICZ041LO	0 0 12 Controller Main Switch STANDBY Contacts Fail Open		
ICZ094LO	0 0 13 Controller Compressor Motor Contacts Fail Open		
IPR004LB	0 0 14 Emergency Air Controller Pressure Regulator Fails		
IPR042LB	0 0 15 Emergency Air Oil Pressure Sensor Fails Low		
IMS004LS	0 0 16 Emergency Air Compressor Motor Fails to Start		
0.744E+03			
0.744E+03			
1 0	0.100E-09	0.100E+02	0.000E+00
2 1	0.800E-08	0.300E+02	0.240E+02
3 1	0.800E-08	0.300E+02	0.240E+02
4 0	0.300E-02	0.500E+01	0.000E+00
5 1	0.100E-09	0.300E+02	0.240E+02
6 1	0.800E-08	0.300E+02	0.240E+02
7 1	0.300E-04	0.100E+02	0.240E+02
8 0	0.600E-02	0.100E+02	0.000E+00
9 1	0.300E-06	0.100E+02	0.240E+02
10 1	0.100E-05	0.100E+02	0.240E+02
11 1	0.200E-06	0.100E+02	0.240E+02
12 1	0.100E-06	0.100E+02	0.240E+02
13 1	0.100E-06	0.100E+02	0.240E+02
14 1	0.100E-05	0.100E+02	0.240E+02
15 1	0.100E-05	0.100E+02	0.240E+02
16 0	0.300E-04	0.100E+02	0.000E+00

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5	8	7	
			Loss of Air Supply to Reactor Building
I0900			Loss of Instrument Air to Reactor Building
	* 2 0	I0901 I0902	
I0901			Loss of Instrument Air Pressure
	+ 1 1	I0903 I0001	
I0902			Loss of Emergency Air Pressure
	+ 1 1	I0904 I0501	
I0903			Instrument Air Valves to Reactor Building Closed
	+ 0 4	IXV134YQ IXV149YQ IXV309YQ IAV695LQ	
I0904			Emergency Air Output Valves Closed
	+ 0 2	IXV120LQ IXV121LQ	
I0001	3 0		1 Loss of Instrument Air System Output
I0501	3 0		1 Loss of Emergency Air System Output
IXV134YQ	0 0		2 Instrument Air Valve PA-134 Fails to Remain Open
IXV149YQ	0 0		3 Instrument Air Valve PA-149 Fails to Remain Open
IXV309YQ	0 0		4 Instrument Air Valve PA-309 Fails to Remain Open
IAV695LQ	0 0		5 Isolation Valve RI-VC-695 Fails Closed
IXV120LQ	0 0		6 Instrument Air Valve RA-120 Fails to Remain Open
IXV121LQ	0 0		7 Instrument Air Valve RA-121 Fails to Remain Open
	0.744E+03		
	0.744E+03		
	1 0	0.100E-09	0.100E+02 0.000E+00
	2 2	0.100E-06	0.300E+01
	3 2	0.100E-06	0.300E+01
	4 2	0.100E-06	0.300E+01
	5 2	0.200E-05	0.300E+01
	6 2	0.100E-06	0.300E+01
	7 2	0.100E-06	0.300E+01

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W0001	Loss of Reactor Auxiliaries Cooling Water	
	Loss of Reactor Auxiliaries Cooling Water	
W0005	+ 1 1 W0005 WFL095YE	
	Loss of Output from Reactor Auxiliaries Pumps	
W0006	+ 2 0 W0006 W0008	
	Loss of Supply to Reactor Auxiliaries Pump System	
W0008	+ 1 1 W0007 WPP006OF	
	Reactor Auxiliaries Dual Pump System Failure	
W0007	* 2 0 W0010 W0020	
	Lack of Supply Water in Tower Basin	
W0010	* 0 2 WPP07AOF WPP07BOF	
	Reactor Aux. Primary Pump System Failure	
W0020	+ 1 1 W0011 WCV001YQ	
	Reactor Aux. Standby Pump System Failure	
W0011	+ 1 1 W0021 WCV002YQ	
	Loss of Output from Reactor Aux. Primary Pump	
W0021	+ 1 1 W0012 WPM001YW	
	Loss of Output from Reactor Aux. Standby Pump	
W0012	+ 1 2 W0022 WPM002YW WPM002YX	
	Loss of Power to Primary Pump Motor	
W0022	+ 2 0 W0013 W0014	
	Loss of Power to Standby Pump Motor	
W0013	+ 2 0 W0023 W0024	
	Loss of 480 V Emergency Power	
W0014	+ 0 2 E0150 ECB10CYO	
	Reactor Aux. Primary Pump Controller Failure	
W0023	+ 0 5 WCZ101YO WCZ102YO WCZ115YO WXT101YW WPR001YO	
	Loss of 480 V Emergency Power	
W0024	+ 0 2 E0150 ECB10DYO	
	Reactor Aux. Standby Pump Controller Failure	
W0025	+ 1 4 W0025 WCZ201YO WCZ202YO WCZ215YO WXT201YW	
	Standby Controller START/RUN Failure	
E0150	+ 0 3 WPR002YN WPR002YO WMS002YW	
WFL095YE	3 0 1 LOSS OF 480 VAC EMERGENCY BUS R-1B	
WPP006OF	0 0 2 Reactor Aux. Water CUNO Filter Blocked	
WPP07AOF	0 0 3 Break in Supply Line from Cooling Tower	
WPP07BOF	0 0 4 Raw Water Supply Line Failure	
WCV001YQ	0 0 5 Insufficient Water Return to Cooling Tower	
WCV002YQ	0 0 6 Reactor Aux. Primary Pump Check Valve Fails Closed	
WPM001YW	0 0 7 Reactor Aux. Standby Pump Check Valve Fails Closed	
WPM002YW	0 0 8 Reactor Aux. Primary Pump Fails	
WPM002YX	0 0 9 Reactor Aux. Standby Pump Fails	
ECB10CYO	0 0 10 Reactor Aux. Standby Pump Down for Maintenance	
WCZ101YO	0 0 11 MCC-R1B Circuit Breaker 10C Fails Open	
WCZ102YO	0 0 12 Primary Controller RUN Contacts Fail Open	
WCZ115YO	0 0 13 Primary Controller OL Contacts (1/2) Fail Open	
WXT101YW	0 0 14 Primary Controller Pump Motor Contacts Fail Open	
WPR001YO	0 0 15 Primary Cont. 480/120 V Transformer Fails	
ECB10DYO	0 0 16 Water Pressure Low Switch Fails to Remain Closed	
WCZ201YO	0 0 17 MCC-R1B Circuit Breaker 10D Fails Open	
WCZ202YO	0 0 18 Standby Cont. STANDBY Contacts Fail Open	
WCZ215YO	0 0 19 Standby Controller OL Contacts (1/2) Fail Open	
WXT201YW	0 0 20 Standby Controller Pump Motor Contacts Fail Open	
WPR002YN	0 0 21 Standby Cont. 480/120 V Transformer Fails	
WPR002YO	0 0 22 Water Pressure Low Switch Fails to Close	
WMS002YW	0 0 23 Water Pressure Low Switch Fails to Remain Closed	
	0 24 Standby Pump Motor Starter Failure	
0.744E+03		
0.744E+03		

1	0	0.100E-09	0.100E+02	0.000E+00
2	2	0.300E-04	0.100E+02	
3	2	0.800E-08	0.300E+02	
4	2	0.800E-08	0.300E+02	
5	2	0.800E-08	0.300E+02	
6	2	0.500E-08	0.100E+02	
7	1	0.500E-08	0.100E+02	0.168E+03
8	2	0.300E-04	0.100E+02	
9	1	0.300E-04	0.100E+02	0.168E+03
10	7	0.100E+01	0.100E+02	0.168E+03
11	2	0.300E-06	0.100E+02	
12	2	0.100E-06	0.100E+02	
13	2	0.200E-06	0.100E+02	
14	2	0.100E-06	0.100E+02	
15	2	0.100E-05	0.300E+01	
16	2	0.100E-05	0.100E+02	
17	1	0.300E-06	0.100E+02	0.168E+03
18	1	0.100E-06	0.100E+02	0.168E+03
19	1	0.200E-06	0.100E+02	0.168E+03
20	1	0.100E-06	0.100E+02	0.168E+03
21	1	0.100E-05	0.100E+02	0.168E+03
22	1	0.300E-06	0.100E+02	0.168E+03
23	1	0.100E-05	0.100E+02	0.168E+03
24	0	0.300E-04	0.100E+02	0.000E+00

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Shield Cooling System

	REDUCED HEAT	REMOVAL BY	SHIELD COOLING
H1000	FAILED OR		
	+ 2 0 H8000		
H8000	NO REACTOR	ROTATING PLUG OR NOZZLE AREA	
	+ 1 1 H8100		
H3000	NO	RECIRCULATION HEAT REMOVAL	
	+ 2 0 H3100		
H8100	NO EXHAUST		
	+ 2 1 H8110		
H3100	NO FLOW IN	RECIRCULATION FAN SECTION	
	+ 2 0 H3110		
H3200	NO FLOW OR	HEAT REMOVAL IN REFRIG SECTION	
	+ 2 0 H4000		
H8110	NO FLOW IN	EXHAUST FILTER SECTION	
	+ 2 0 H8101		
H8120	NO FLOW IN	EXHAUST FAN SECTION	
	+ 2 0 H8121		
H3110	REDUCED OR NO	RECIRCULATION LEG 1	
	+ 2 2 H7070	HDM722SO HDM764SQ	
H3120	REDUCED OR NO	RECIRCULATION LEG 2	
	+ 4 0 H7080	H7220 H7235 H7650	
H4000	NO FLOW OR	HEAT REMOVAL IN 40 TON SECTION	
	+ 2 1 H4100	HDM741SQ	
H2000	NO FLOW OR	HEAT REMOVAL IN 20 TON SECTION	
	+ 4 0 H2100	H2200 H7250 H0742	
H8101	NO FLOW IN EX	FILTER LEG 1 & NO RECOVERY ACTION	
	+ 1 1 H8111	HHU01BSZ	
H8102	NO FLOW IN EX	FILTER LEG 1 & RECOVERY ACTION (LEG 2)	
	+ 2 0 H8111	H8112	
H8121	REDUCED OR NO	FLOW IN EXHAUST LEG 1	
	+ 4 1 H7050	H7057 H7260 H7275 HDM745SQ	
H8122	REDUCED OR NO	FLOW IN EXHAUST LEG 2	
	+ 5 0 H7060	H7065 H7270 H7265 H7460	
H7070	FAN 707 IN	RECIRC LEG 1 NOT RUNNING R8-F-707	
	+ 2 3 H0480	H0134 HFN707SR HWR707TB HWR707TC	
H7230	AUTO DAMPER 723	DOES NOT REMAIN OPEN R8-DM-723	
	+ 1 1 H0099	HDM723SQ	
H7080	FAN 708 IN	RECIRC LEG 2 NOT RUNNING	
	+ 0 5 E0150	HFN708SR HFN708SS HWR708TB HWR708TC	
H7220	AUTOMATIC DAMPER 722	CLOSED AND REMAINS CLOSED	
	+ 1 1 H7223	HDM722SW	
H7235	AUTOMATIC DAMPER 723	OPEN AND REMAINS OPEN	
	+ 1 1 H7238	HDM723SW	
H7650	ISOLATION	DAMPER 765 NOT OPEN R8-DM-765	
	+ 0 2 HDM765SL	HDM765SQ	
H4100	40 TON COOLER	NO HEAT REMOVAL	
	+ 3 0 H4110	H4120 H4130	
H7240	AUTO DAMPER 724	DOES NOT REMAIN OPEN R8-DM-724	
	+ 1 1 H0101	HDM724SQ	
H2100	20 TON	COOLER #1 NO HEAT REMOVAL	
	+ 4 0 H2110	H2120 H2130 H2140	
H2200	20 TON	COOLER #2 NO HEAT REMOVAL	
	+ 4 0 H2210	H2220 H2230 H2240	
H7250	AUTOMATIC DAMPER 725	CLOSED AND REMAINS CLOSED	
	+ 1 1 H7253	HDM725SW	
H0742	ISOLATION	DAMPER 742 NOT OPEN	
	+ 0 2 HDM742SL	HDM742SQ	
H8111	NO FLOW IN	EXHAUST FILTER LEG 1	
	+ 0 2 HDM747SQ	HFL717SE	

	NO FLOW IN	EXHAUST	FILTER LEG 2
H8112	+ 0 3 HDM748SP	HDM748SQ	HFL718SE
H7050	FAN 705 IN EXHAUST LEG 1	NOT RUNNING	
H7057	+ 2 4 H5100	H0041	HFN705SR HRE051SO HWR705TB HWR705TC
	VORTEX DAMPER FAN 705	CLOSES AND REMAINS CLOSED	
H7260	+ 1 1 H7058	HDM705SW	
	AUTO DAMPER 726	DOES NOT	REMAIN OPEN R8-DM-726
H7275	+ 1 2 H0096	I0900	HDM726SQ
	AUTO DAMPER 727	DOES NOT	REMAIN CLOSED R8-DM-727
H7060	+ 0 2 H5V095SO	HAV727SO	
	FAN 706 IN EXHAUST LEG 2	NOT RUNNING	
H7065	+ 1 4 H0021	HFN706SR HFN706SS	HMR706TB HMR706TC
	VORTEX DAMPER ON FAN 706	CLOSED AND REMAINS CLOSED	
H7270	+ 1 1 H7066	HDM706SW	
	AUTOMATIC DAMPER 727	CLOSED AND REMAINS CLOSED	
H7265	+ 1 1 H7273	HDM727SW	
	AUTOMATIC DAMPER 726	OPEN AND REMAINS OPEN	
H7460	+ 1 1 H7268	HDM726SW	
	ISOLATION DAMPER 746	CLOSED	R8-DM-746
H0480	+ 0 2 HDM746SL	HDM746SQ	
	MOTOR RELAY RM48	NOT ENERGIZED R8-F-707	
H0134	+ 1 1 H4812	HTZ099SW	
	NO ELECTRICAL POWER @ OUTPUT OF BREAKER MCC R1B-13D		
H0099	+ 1 1 H0934	E0150	
	CONTROL RELAY CR-99	FALSE	CLOSING SIGNAL DM-723/CR-99
H7223	+ 0 2 HRE99CSO	HXT099SW	
	AUTOMATIC DAMPER 722	CLOSED AND IS NOT RECOVERED	
H7238	+ 1 1 H7224	HHU003SL	
	AUTOMATIC DAMPER 723	OPEN AND NOT RECOVERED	
H4110	+ 1 1 H7239	HHU003SL	
	EXTERNAL HEAT REJECTION	NOT AVAILABLE	40 TON SYSTEM
H4120	+ 3 0 H0700	H7020	H7030
	NO FREON FLOWPATH TO	COOLER COILS	R8-HE-713/714
H4130	+ 4 1 H7131	H7132	H7141
	NORMAL 40 TON COMPRESSOR	NOT RUNNING	H7142 HXV953SQ
H0101	+ 1 1 H0094	HCM040SR	
	CONTROL RELAY CR-101	FALSE	CLOSING SIGNAL DM-724/CR-101
H2110	+ 1 2 H5412	HTZ741SW HRE01CSW	
	EXTERNAL HEAT REJECTION	NOT AVAILABLE	20 TON #1
H2120	+ 3 0 H0335	H3470	H3490
	IMPROPER SHIFT FROM SPACE	COOLER TO	SHIELD COOLING
H2130	+ 2 0 H2121	H2125	
	SPACE COOLER #1	SHIFT TO SHIELD COOLING	NOT MAINTAINED
H2140	+ 2 0 H2131	H2135	
	STANDBY 20 TON COMPRESSOR #1	NOT RUNNING	
H2210	+ 1 2 H0092	HCM201SR HSW201ST	
	EXTERNAL HEAT REJECTION	NOT AVAILABLE	20 TON #2
H2220	+ 3 0 H0334	H3460	H3480
	IMPROPER SHIFT FROM SPACE	COOLER TO	SHIELD COOLING
H2230	+ 2 0 H2221	H2225	
	SPACE COOLER #2	SHIFT TO SHIELD COOLING	NOT MAINTAINED
H2240	+ 2 0 H2231	H2235	
	STANDBY 20 TON COMPRESSOR #2	NOT RUNNING	
H7253	+ 1 2 H0093	HCM202SR HSW202ST	
	AUTOMATIC DAMPER 725	CLOSED AND IS NOT RECOVERED	
H5100	+ 1 1 H7254	HHU003SL	
	MOTOR RELAY RM51	NOT ENERGIZED	R8-F-705/RM-51
H0041	+ 1 1 H5112	HTZ096SW	
	NO ELECTRICAL POWER @ OUTPUT OF BREAKER MCC	2-4A	
	+ 1 1 H0941	E0300	

H7058	VORTEX DAMPER FAN 705 CLOSING AND IS NOT RECOVERED	
H0096	* 1 1 H7059 HHU004SL	DM-726/RSV-96
H0021	SOLENOID VALVE RSV 96	CLOSED
H7066	+ 2 1 H5112 H7459 HSV096SQ	
H7273	NO ELECTRICAL POWER & OUTPUT OF BREAKER MCC R2-2A	
H7268	+ 1 1 H0921 E0300	
H4812	VORTEX DAMPER ON FAN 706 CLOSED AND IS NOT RECOVERED	
H0934	* 1 1 H7067 HHU004SL	
H7224	AUTOMATIC DAMPER 727 CLOSED AND IS NOT RECOVERED	
H7239	* 1 1 H7274 HHU003SL	
H0700	AUTOMATIC DAMPER 726 OPEN AND NOT RECOVERED	
H7141	* 1 1 H7269 HHU003SL	
H7142	NOT OUTPUT FROM 80/120 V RM-48 STARTER TRANSFORMER	
H0094	+ 1 1 H0134 HXT048SW	
H5412	BREAKER NOT RESET AFTER SPURIOUS TRIP MCC R1B-13D	
H0335	* 0 2 ECB13DYB HHU002YL	
H3470	AUTOMATIC DAMPER 722 NOT OPEN R8-DM-722	
H3490	+ 2 0 H7221 H7222	
H2121	AUTOMATIC DAMPER 723 NOT SHUT R8-DM-723	
H2125	+ 1 1 H7236 HDM723SO	
H2131	WATER COOLED 40 TON COND NOT AVAILABLE P8-C-700	
H2135	+ 0 2 WGV701PQ W0001	
H0092	RX BLD FREON ISOLATION DOES NOT REMAIN OPEN P17-VC-702B	
H0334	+ 1 1 H7021 HAV702PQ	
H3460	RX BLD FREON ISOLATION DOES NOT REMAIN OPEN P17-VC-703B	
H3480	+ 1 1 H7031 HAV703PQ	
	NO FREON FLOWPATH TO COOLER COIL R8-HE-713 #1	
	+ 0 3 HXV955SQ HSV769SQ HGV754SQ	
	NO FREON FLOWPATH TO COOLER COIL R8-HE-713 #2	
	+ 0 3 HXV954SQ HSV756SQ HGV754SQ	
	NO FREON FLOWPATH TO COOLER COIL R8-HE-714 #1	
	+ 0 3 HXV957SQ HSV755SQ HGV754SQ	
	NO FREON FLOWPATH TO COOLER COIL R8-HE-714 #2	
	+ 0 3 HXV956SQ HSV768SQ HGV754SQ	
	NO ELECTRICAL POWER & OUTPUT OF BREAKER MCC R1B-9D	
	+ 1 1 H0994 E0150	
	NO OUTPUT FROM 480/120 V RM-54 STARTER TRANSFORMER	
	+ 1 1 H0094 HXT054SW	
	WATER COOLED 20 TON #1 COND NOT AVAILABLE P8-C-335	
	+ 0 2 W0001 WGV342PQ	
	RX BLD FREON ISOLATION DOES NOT REMAIN OPEN P17-VC-347B	
	+ 1 1 H3471 HAV347PQ	
	RX BLD FREON ISOLATION DOES NOT REMAIN OPEN P17-VC-349B	
	+ 1 1 H3491 HAV349PQ	
	NO FREON FLOWPATH TO COOLER COILS R8-HE-715	
	+ 2 2 H2122 H2123 HSV357SP HXV963SL	
	SPACE COOLING UNIT LEFT ON LINE R17-AU-309	
	* 1 1 H2126 HSV356SN	
	LOSS OF FREON FLOWPATH TO COOLER COILS R8-HE-715	
	+ 2 2 H2132 H2133 HSV357SQ HXV963SQ	
	SPACE COOLING UNIT LEFT ON LINE R17-AU-309	
	* 1 1 H2136 HSV356SO	
	NO ELECTRICAL POWER & OUTPUT OF BREAKER MCC R1B-9B	
	+ 1 1 H0992 E0150	
	WATER COOLED 20 TON #2 COND NOT AVAILABLE P8-C-334	
	+ 0 2 W0001 WGV343PQ	
	RX BLD FREON ISOLATION DOES NOT REMAIN OPEN P17-VC-346B	
	+ 1 1 H3461 HAV346PQ	
	RX BLD FREON ISOLATION DOES NOT REMAIN OPEN P17-VC-348B	
	+ 1 1 H3481 HAV348PQ	

H2221	NO FREON	FLOWPATH TO	COOLER COILS	R8-HE-718
H2225	+ 2 2 H2222	H2223 HSV359SP	H2223 HSV359SP	R8-HE-718
H2231	SPACE COOLING	UNIT LEFT	ON LINE	R17-AU-308
H2235	* 1 1 H2226	HSV358LN		
H0093	LOSS OF FREON	FLOWPATH TO	COOLER COILS	R8-HE-718
H2235	+ 2 2 H2232	H2233 HSV359SQ	H2233 HSV359SQ	R8-HE-718
H0093	SPACE COOLING	UNIT LEFT	ON LINE	R17-AU-308
H2235	* 1 1 H2236	HSV358LO		
H0093	NO ELECTRICAL POWER & OUTPUT OF BREAKER	MCC R1B-9C		
H2254	+ 1 1 H0993	E0150		
H2254	AUTOMATIC DAMPER	725 NOT OPEN	R8-DM-725	
H5112	+ 2 0 H7251	H7252		
H0941	NO OUTPUT FROM	480/120 V RM-51	STARTER TRANSFORMER	
H0941	+ 1 1 H0041	HXT051SW		
H7059	BREAKER NOT RESET AFTER SPURIOUS TRIP	MCC R2-4A		
H7459	VORTEX DAMPER	ON FAN 705	DOES NOT	REMAIN OPEN
H0921	+ 0 2 HPR641SW	HLC641SW		
H7067	DAMPER 745	INDICATES	NOT FULLY OPEN	DM-745/MD-51-1
H7274	+ 0 2 HT2745SW	HDM745SQ		
H7269	BREAKER NOT RESET AFTER SPURIOUS TRIP	MCC R2-2A		
H7221	* 0 2 ECB02AYB	HHU002YL		
H7222	VORTEX DAMPER	ON FAN 706	CLOSED	FN-706/RM-97
H7274	+ 2 0 H7068	H7069		
H7269	AUTOMATIC DAMPER	727 NOT OPEN	R8-DM-727	
H7221	+ 2 0 H7271	H7272		
H7222	AUTOMATIC DAMPER	726 NOT SHUT	R8-DM-726	
H7236	+ 2 0 H7266	H7267		
H7021	AUTOMATIC DAMPER	722 DOES NOT OPEN	R8-DM-722	
H7031	+ 2 2 H0200	H0043 HRE00MSW	HDM722SP	
H0994	AUTO DAMPER	722 DOES NOT	REMAIN OPEN	R8-DM-722
H3471	+ 1 1 H0100	HDM722SQ		
H3491	AUTOMATIC DAMPER	723 DOES NOT CLOSE	R8-DM-723	
H2122	+ 1 3 H0023	HDM723SN HRE99MSW	HRE99CSN	
H2123	RX BLD FREON SOLENOID	DOES NOT REMAIN OPEN	P17-VS-702A	
H2126	+ 1 3 K0900	KIN003UW HSV702PQ	HGV702PQ	
H2132	RX BLD FREON SOLENOID	DOES NOT REMAIN OPEN	P17-VS-703A	
H2133	+ 1 3 K0900	KIN003UW HSV703PQ	HGV703PQ	
H2136	BREAKER NOT RESET AFTER SPURIOUS TRIP	MCC R1B-9D		
H0992	* 0 2 ECBB9DYB	HHU002YL		
H3461	RX BLD FREON SOLENOID	DOES NOT REMAIN OPEN	P17-VS-347A	
	+ 1 3 K0900	KIN003UW HSV347PQ	HGV347PQ	
	RX BLD FREON SOLENOID	DOES NOT REMAIN OPEN	P17-VS-349A	
	+ 1 3 K0900	KIN003UW HSV349PQ	HGV349PQ	
	NO FREON	FLOWPATH TO	COOLER COILS	R8-HE-715 #1
	+ 0 3 HXV960SL	HSV767SP	HGV736SL	
	NO FREON	FLOWPATH TO	COOLER COILS	R8-HE-715 #2
	+ 0 3 HXV958SL	HSV758SP	HGV735SL	
	SPACE COOLING	COILS NOT	ISOLATED	R17-AU-309
	+ 0 2 HSV766SN	HSV759SN		
	LOSS OF FREON	FLOWPATH TO	COOLER COILS	R8-HE-715 #1
	+ 0 3 HXV960SQ	HSV767SQ	HGV736SQ	
	LOSS OF FREON	FLOWPATH TO	COOLER COILS	R8-HE-715 #2
	+ 0 3 HXV958SQ	HSV758SQ	HGV735SQ	
	SPACE COOLING	COILS DO NOT	REMAIN ISOLATED	R17-AU-309
	+ 0 2 HSV766SO	HSV759SO		
	BREAKER NOT RESET AFTER SPURIOUS TRIP	MCC R1B-9B		
	* 0 2 ECBB9BYB	HHU002YL		
	RX BLD FREON SOLENOID	DOES NOT REMAIN OPEN	P17-VS-346A	
	+ 1 3 K0900	KIN003UW HSV346PQ	HGV346PQ	

H3481	RX BLD FREON SOLENOID DOES NOT REMAIN OPEN P	-348A
H2222	+ 1 3 K0900 KIN003UW HSV348PQ HGV348PQ	
H2223	NO FREON FLOWPATH TO COOLER COILS R8-HE-718 #1	
H2226	+ 0 3 HKV961SL HSV757SP HGV737SL	
H2232	NO FREON FLOWPATH TO COOLER COILS R8-HE-718 #2	
H2233	+ 0 3 HKV962SL HSV770SP HGV738SL	
H2236	SPACE COOLING COILS NOT ISOLATED R17-AU-308	
H0993	+ 0 2 HSV400LN HSV771LN	
H7251	LOSS OF FREON FLOWPATH TO COOLER COILS R8-HE-718 #1	
H7252	+ 0 3 HKV961SQ HSV757SQ HGV737SQ	
H7068	LOSS OF FREON FLOWPATH TO COOLER COILS R8-HE-718 #2	
H7069	+ 0 3 HKV962SQ HSV770SQ HGV738SQ	
H7271	SPACE COOLING COILS NOT ISOLATED R17-AU-308	
H7272	+ 0 2 HSV400LO HSV771LO	
H7266	BREAKER NOT RESET AFTER SPURIOUS TRIP MCC R1B-9C	
H7267	* 0 2 ECB9CYB HHU002YL	
H0200	AUTOMATIC DAMPER 725 FAILS TO OPEN R8-DM-725	
H0043	+ 1 2 H0102 HRE02MSW HDM725SP	
H0100	AUTO DAMPER 725 FAILS TO REMAIN OPEN R8-DM-725	
H0023	+ 1 1 H0202 HDM725SQ	
K0900	VORTEX DAMPER ON FAN 706 FAILS TO OPEN FN-706/RM-97	
H0102	+ 1 2 H0069 HRE097SW HDM706SP	
H0202	VORTEX DAMPER FAN 706 FAILS TO REMAIN OPEN FN-706/RM-97	
H0069	+ 0 3 HDM706SQ HPR642SW HLC642SW	
H9500	AUTOMATIC DAMPER 727 DOES NOT OPEN R8-DM-727	
H0095	+ 1 2 H9500 I0900 HDM727SP	
H0943	AUTO DAMPER 727 DOES NOT REMAIN OPEN R8-DM-727	
H0923	+ 1 2 H0095 I0900 HDM727SQ	
H5024	AUTOMATIC DAMPER 726 FAILS TO CLOSE R8-DM-726	
H7468	+ 0 2 HSV096SN HDM726SN	
	AUTO DAMPER 726 DOES NOT REMAIN CLOSED R8-DM-726	
	+ 0 2 HSV096SO HDM726SO	
	NO OPENING SIGNAL TO DAMPER 722 DM-722/CR-100	
	+ 1 2 H0043 HXT100SW HRE00CSW	
	NO ELECTRICAL POWER & OUTPUT OF BREAKER MCC R2-4C	
	+ 1 1 H0943 E0360	
	CONTROL RELAY CR-100 FALSE CLOSING SIGNAL DM-722/CR-100	
	+ 0 2 HRE00CSO HXT100SW	
	NO ELECTRICAL POWER & OUTPUT OF BREAKER MCC R2-3A	
	+ 1 1 H0923 E0360	
	LOSS OF INST AIR TO REACTOR BLD ISOLATION VALVES	
	* 0 2 I0900 KAU000PH	
	CONTROL RELAY CR-102 NO OPENING SIGNAL DM-725/CR-102	
	+ 0 3 E0150 HTZ742SW HRE02CSW	
	CONTROL RELAY CR-102 FALSE CLOSING SIGNAL DM-725/CR-102	
	+ 0 3 E0150 HTZ742SW HRE02CSO	
	VORTEX DAMPER CONTROLLER ON FAN 706 NO OPEN SIGNAL RM-97	
	+ 1 2 H5024 HPR642SW HLC642SW	
	SOLENOID VALVE RSV 95 DOES NOT OPEN DM-727/RSV-95	
	+ 1 2 H7468 HXT050SW HSV095SP	
	SOLENOID VALVE RSV 95 CLOSED DM-727/RSV-95	
	+ 1 2 H7469 HXT050SW HSV095SQ	
	BREAKER NOT RESET AFTER SPURIOUS TRIP MCC R2-4C	
	* 0 2 ECB04CYB HHU002YL	
	BREAKER NOR RESET AFTER SPURIOUS TRIP MCC R2-3A	
	* 0 2 ECB03AYB HHU002YL	
	NO 24V POWER VORTEX DAMPER CONTROLLER FN-706/RM-97	
	+ 0 2 HXT050SW HXT098SW	
	DAMPER 746 INDICATES NOT FULLY OPEN R8-DM-746	
	+ 0 2 HDM746SL HTZ746SW	

H7469	DAMPER 746	INDICATES	NOT FULLY OPEN DM-746/MD-50-1
Y1000	3 0	1 NO FORCED EXHAUST FLOW (Shield)	(Cooling)
E0150	3 0	1 LOSS OF 480 VAC EMERGENCY BUS R-1B	
I0900	3 0	1 Loss of Instrument Air to Reactor Building	
E0300	3 0	1 LOSS OF 480 VAC EMERGENCY BUS R-2A	
W0001	3 0	1 Loss of Reactor Auxiliaries Cooling Water	
E0360	3 0	1 LOSS OF 480 VAC EMERGENCY BUS R-2B	
H000100	2 0	2 RX BUILDING AMBIENT TEMP TOO HIGH	
HDM722SQ	0 0	3 DAMPER 722 FAILS TO REMAIN CLOSED R8-DM-722	
HDM764SQ	0 0	4 ISOL DAMPER 764 DOES NOT REMAIN OPEN	R8-DM-764
HDM741SQ	0 0	5 ISOL DAMPER 741 DOES NOT REMAIN OPEN	R8-DM-741
HU01BSZ	0 0	6 NO RECOVERY ACTION FOR CLOGGED FILTER R8-FL-717	
HDM745SQ	0 0	7 ISOL DAMPER 745 DOES NOT REMAIN OPEN	R8-DM-745
HFN707SR	0 0	8 RECIRCULATION FAN 707 FAILS TO RUN	R8-F-707
HWR707TB	0 0	9 FAN 707 POWER CABLE FAILS: OPEN CIRCUIT	
HWR707TC	0 0	10 FAN 707 POWER CABLE FAILS: SHORT TO GROUND	
HDM723SQ	0 0	11 DAMPER 723 FAILS TO REMAIN OPEN	R8-DM-723
HFN708SR	0 0	12 RECIRCULATION FAN 708 FAILS TO RUN	R8-F-708
HFN708SS	0 0	13 RECIRCULATION FAN 708 FAILS TO START R8-F-708	
HWR708TB	0 0	14 FAN 708 POWER CABLE FAILS: OPEN CIRCUIT	
HWR708TC	0 0	15 FAN 708 POWER CABLE FAILS: SHORT TO GROUND	
HDM722SW	0 0	16 DAMPER 722 INTERNAL FAILURE R8-DM-722	
HDM723SW	0 0	17 DAMPER 723 INTERNAL FAILURE R8-DM-723	
HDM765SL	0 0	18 ISOLATION DAMPER 765 LEFT CLOSED R8-DM-765	
HDM765SQ	0 0	19 ISOL DAMPER 765 DOES NOT REMAIN OPEN	R8-DM-765
HDM724SQ	0 0	20 DAMPER 724 FAILS TO REMAIN OPEN	R8-DM-724
HDM725SW	0 0	21 DAMPER 725 INTERNAL FAILURE R8-DM-725	
HDM742SL	0 0	22 ISOLATION DAMPER 742 LEFT CLOSED R8-DM-742	
HDM742SQ	0 0	23 ISOL DAMPER 742 DOES NOT REMAIN OPEN	R8-DM-742
HDM747SQ	0 0	24 FLTR 717 INLET DMPR 747 FAILS TO REMAIN OPEN	FL717/R8-DM-747
HFL717SE	0 0	25 ABSOLUTE FILTERS 717 CLOGGED	R8-FL-717
HDM748SP	0 0	26 FLTR 718 INLET DAMPER 748 FAILS TO OPEN	FL718/R8-DM-748
HDM748SQ	0 0	27 FLTR 718 INLET DMPR 748 FAILS TO REMAIN OPEN	FL718/R8-DM-748
HFL718SE	0 0	28 ABSOLUTE FILTERS 718 CLOGGED	R8-FL-718
HFN705SR	0 0	29 EXHAUST FAN 705 FAILS TO RUN	R8-F-705
HRE051SO	0 0	30 MOTOR RELAY RM-51 COIL FAILURE	FN-705/RM-51
HWR705TB	0 0	31 FAN 705 POWER CABLE FAILS: OPEN CIRCUIT	
HWR705TC	0 0	32 FAN 705 POWER CABLE FAILS: SHORT TO GROUND	
HDM705SW	0 0	33 VORTEX DAMPER FAN 705 INTERNAL FAILURE	
HDM726SQ	0 0	34 DAMPER 726 FAILS TO REMAIN OPEN	R8-DM-726
HSV095SO	0 0	35 SOLENOID VALVE RSV-95 FAILS TO REMAIN CLOSED	DM-727/RSV-95
HAV727SO	0 0	36 AUTO DAMPER 727 FAILS TO REMAIN CLOSED	R8-DM-727
HFN706SR	0 0	37 EXHAUST FAN 706 FAILS TO RUN	R8-F-706
HFN706SS	0 0	38 EXHAUST FAN 706 FAILS TO START R8-F-706	
HWR706TB	0 0	39 FAN 706 POWER CABLE FAILS: OPEN CIRCUIT	
HWR706TC	0 0	40 FAN 706 POWER CABLE FAILS: SHORT TO GROUND	
HDM706SW	0 0	41 DAMPER 706 INTERNAL FAILURE R8-DM-706	
HDM727SW	0 0	42 DAMPER 727 INTERNAL FAILURE R8-DM-727	
HDM726SW	0 0	43 DAMPER 726 INTERNAL FAILURE R8-DM-726	
HDM746SL	0 0	44 ISOLATION DAMPER 746 LEFT CLOSED R8-DM-746	
HDM746SQ	0 0	45 ISOL DAMPER 746 DOES NOT REMAIN OPEN	R8-DM-746
HRT099SW	0 0	46 DM723 POSITION INDICATOR ERROR	DM-723/ES-99-1
HRE99CSO	0 0	47 CONTROL RELAY CR-99 COIL FAILURE	DM-723/CR-99
HXT099SW	0 0	48 CR99 120V/24V TRANSFORMER FAILS	
HU0003SL	0 0	49 NO RECOVERY ACTION TO USE DAMPER MANUAL HANDWHEEL	
HKV953SQ	0 0	50 COILS 713/714 ISOLATION FAILS TO REMAIN OPEN	R8-V-953
HCM040SR	0 0	51 40 TON COMPRESSOR FAILURE TO RUN	
HCT741SW	0 0	52 DM741 POSITION INDICATOR ERROR	
HRE01CSW	0 0	53 CONTROL RELAY CR-101 CONTACT FAULT	DM--724/CR-101

HSV347PQ	0	0	114	SOLENOID VALVE VS-347A FAILS	TO REMAIN OPEN	P17-VS-347A
HGV347PQ	0	0	115	REGULATOR FOR VS-347A FAILS	TO REMAIN OPEN	P17-VS-347A
HSV349PQ	0	0	116	SOLENOID VALVE VS-349A FAILS	TO REMAIN OPEN	P17-VS-349A
HGV349PQ	0	0	117	REGULATOR FOR VS-349A FAILS	TO REMAIN OPEN	P17-VS-349A
HXV960SL	0	0	118	COIL 715 #1 ISOLATION LEFT CLOSED	R8-V-960	
HSV767SP	0	0	119	COIL 715 #1 SOLENOID	FAILS TO OPEN	R8-VS-767
HGV736SL	0	0	120	COIL 715 #1 EXPANSION VALVE LEFT CLOSED	R8-VC-736	
HXV958SL	0	0	121	COIL 715 #2 ISOLATION LEFT CLOSED	R8-V-958	
HSV758SP	0	0	122	COIL 715 #2 SOLENOID	FAILS TO OPEN	R8-VS-758
HGV735SL	0	0	123	COIL 715 #2 EXPANSION VALVE LEFT CLOSED	R8-VC-735	
HSV766SN	0	0	124	CLR 309 COIL 2 SOLENOID FAILS TO CLOSE	R8-VS-766	
HSV759SN	0	0	125	CLR 309 COIL 1 SOLENOID FAILS TO CLOSE	R8-VS-759	
HXV960SQ	0	0	126	COIL 715 #1 ISOLATION FAILS TO REMAIN OPEN	R8-V-960	
HSV767SQ	0	0	127	COIL 715 #1 SOLENOID FAILS TO REMAIN OPEN	R8-VS-767	
HGV736SQ	0	0	128	COIL 715 #1 EXP VALVE DOES NOT REMAIN OPEN	R8-VC-736	
HXV958SQ	0	0	129	COIL 715 #2 ISOLATION FAILS TO REMAIN OPEN	R8-V-958	
HSV758SQ	0	0	130	COIL 715 #2 SOLENOID FAILS TO REMAIN OPEN	R8-VS-758	
HGV735SQ	0	0	131	COIL 715 #2 EXP VALVE DOES NOT REMAIN OPEN	R8-VC-735	
HSV766SO	0	0	132	CLR 309 COIL 2 SOLENOID FAILS TO REMAIN CLOSED	R8-VS-766	
HSV759SO	0	0	133	CLR 309 COIL 1 SOLENOID FAILS TO REMAIN CLOSED	R8-VS-759	
ECBB9BYB	0	0	134	CIRCUIT BREAKER SPURIOUS TRIP	MCC R1B-9B	
HSV346PQ	0	0	135	SOLENOID VALVE VS-346A FAILS	TO REMAIN OPEN	P17-VS-346A
HGV346PQ	0	0	136	REGULATOR FOR VS-346A FAILS	TO REMAIN OPEN	P17-VS-346A
HSV348PQ	0	0	137	SOLENOID VALVE VS-348A FAILS	TO REMAIN OPEN	P17-VS-348A
HGV348PQ	0	0	138	REGULATOR FOR VS-348A FAILS	TO REMAIN OPEN	P17-VS-348A
HXV961SL	0	0	139	COIL 718 #1 ISOLATION LEFT CLOSED	R8-V-961	
HSV757SP	0	0	140	COIL 718 #1 SOLENOID	FAILS TO OPEN	R8-VS-757
HGV737SL	0	0	141	COIL 718 #1 EXPANSION VALVE LEFT CLOSED	R8-VC-737	
HXV962SL	0	0	142	COIL 718 #2 ISOLATION LEFT CLOSED	R8-V-962	
HSV770SP	0	0	143	COIL 718 #2 SOLENOID	FAILS TO OPEN	R8-VS-770
HGV738SL	0	0	144	COIL 718 #2 EXPANSION VALVE LEFT CLOSED	R8-VC-738	
HSV400LN	0	0	145	CLR 308 COIL 2 SOLENOID FAILS TO CLOSE	R8-VS-400	
HSV771LN	0	0	146	CLR 308 COIL 1 SOLENOID FAILS TO CLOSE	R8-VS-771	
HXV961SQ	0	0	147	COIL 718 #1 ISOLATION FAILS TO REMAIN OPEN	R8-V-961	
HSV757SQ	0	0	148	COIL 718 #1 SOLENOID FAILS TO REMAIN OPEN	R8-VS-757	
HGV737SQ	0	0	149	COIL 718 #1 EXP VALVE DOES NOT REMAIN OPEN	R8-VC-737	
HXV962SQ	0	0	150	COIL 718 #2 ISOLATION FAILS TO REMAIN OPEN	R8-V-962	
HSV770SQ	0	0	151	COIL 718 #2 SOLENOID FAILS TO REMAIN OPEN	R8-VS-770	
HGV738SQ	0	0	152	COIL 718 #2 EXP VALVE DOES NOT REMAIN OPEN	R8-VC-738	
HSV400LO	0	0	153	CLR 308 COIL 2 SOLENOID FAILS TO REMAIN CLOSED	R8-VS-400	
HSV771LO	0	0	154	CLR 308 COIL 1 SOLENOID FAILS TO REMAIN CLOSED	R8-VS-771	
ECBB9CYB	0	0	155	CIRCUIT BREAKER SPURIOUS TRIP	R1B-9C	
HRE02MSW	0	0	156	MOTOR OPERATOR RM-102 FAILS	TO OPERATE	DM-725/RM-102
HDM725SP	0	0	157	DAMPER 725 MECHANICAL	FAILS TO OPEN	R8-DM-725
HDM725SQ	0	0	158	DAMPER 725 FAILS TO	REMAIN OPEN	R8-DM-725
HRE097SW	0	0	159	MOTOR OPERATOR VORTEX DAMPER	FAILS	FN-706/RM-97
HDM706SP	0	0	160	VORTEX DAMPER ON FAN 706	DOES NOT OPEN	FN-706
HDM706SQ	0	0	161	VORTEX DAMPER FAN 706 DOES	NOT REMAIN OPEN	FN-706
HPR642SW	0	0	162	PRESSURE TRANSDUCER	FAILS	R13-1564-2
HLC642SW	0	0	163	PRESSURE CONTROLLER	FAILS	R13-1564-2
HDM727SP	0	0	164	DAMPER 727 MECHANICAL	FAILS TO OPEN	R8-DM-727
HDM727SQ	0	0	165	DAMPER 727 FAILS TO	REMAIN OPEN	R8-DM-727
HSV096SN	0	0	166	SOLENOID VALVE RSV-96 FAILS TO CLOSE	DM-726/RSV-96	
HDM726SN	0	0	167	DAMPER 726 MECHANICAL FAILS TO CLOSE	R8-DM-726	
HSV096SO	0	0	168	SOLENOID VALVE RSV-96 FAILS TO REMAIN CLOSED	DM-726/RSV-96	
HDM726SQ	0	0	169	DAMPER 726 FAILS TO REMAIN CLOSED	R8-DM-726	
HXT100SW	0	0	170	CR100 120V/24V TRANSFORMER	FAILS	
HRE00CSW	0	0	171	CONTROL RELAY CR-100 CONTACT FAILURE		DM-722/CR-100
HRE00CSO	0	0	172	CONTROL RELAY CR-100 COIL FAILURE		DM-722/CR-100
KAU000PH	0	0	173	REACTOR BLD ISOLATION VALVE ACCUMULATES	FAIL	

HTZ742SW	0	0	174	DM742 POSITION INDICATOR	ERROR	
HRE02CSW	0	0	175	CONTROL RELAY CR-102 CONTACT FAILURE		DM-725/CR-102
HRE02CSO	0	0	176	CONTROL RELAY CR-102 COIL FAILURE		DM-725/CR-102
HXT050SW	0	0	177	RM50 480V/120V 250VA STARTER TRANSFORMER		FAILS
HSV095SP	0	0	178	SOLENOID VALVE RSV-95	FAILS TO OPEN	DM-727/RSV-95
HSV095SQ	0	0	179	SOLENOID VALVE RSV-95	FAILS TO REMAIN OPEN	DM-727/RSV-95
ECB04CYB	0	0	180	CIRCUIT BREAKER SPURIOUS TRIP MCC R2B-4C		
ECB03AYB	0	0	181	CIRCUIT BREAKER SPURIOUS TRIP		MCC R2-3A
HXT098SW	0	0	182	RM98 120V/24V TRANSFORMER	FAILS	
HTZ746SW	0	0	183	DM746 POSITION INDICATOR	ERROR	DM-746/MD-50-1
0.744E+03						
0.744E+03						
1	0	0.100E-09	0.100E+02	0.000E+00		
2	F	0.000E+00	0.100E+01			
3	2	0.100E-06	0.100E+02			
4	2	0.360E-07	0.830E+01			
5	2	0.360E-07	0.830E+01			
6	0	0.200E-01	0.100E+02	0.000E+00		
7	1	0.360E-07	0.830E+01			
8	2	0.300E-04	0.100E+02			
9	2	0.300E-05	0.100E+02			
10	2	0.300E-06	0.100E+02			
11	2	0.100E-06	0.100E+02			
12	1	0.300E-04	0.100E+02			
13	0	0.100E-02	0.100E+02	0.000E+00		
14	1	0.300E-05	0.100E+02			
15	1	0.300E-06	0.100E+02			
16	1	0.360E-07	0.830E+01			
17	1	0.360E-07	0.830E+01			
18	0	0.100E-03	0.100E+02	0.000E+00		
19	1	0.360E-07	0.830E+01			
20	2	0.100E-06	0.100E+02			
21	1	0.360E-07	0.830E+01			
22	0	0.100E-03	0.100E+02	0.000E+00		
23	1	0.360E-07	0.830E+01			
24	2	0.360E-07	0.830E+01			
25	2	0.300E-04	0.100E+02			
26	0	0.200E-03	0.100E+02	0.000E+00		
27	1	0.360E-07	0.830E+01			
28	2	0.300E-04	0.100E+02			
29	2	0.300E-04	0.100E+02			
30	2	0.300E-05	0.100E+02			
31	2	0.300E-05	0.100E+02			
32	2	0.300E-06	0.100E+02			
33	2	0.360E-07	0.830E+01			
34	2	0.100E-06	0.100E+02			
35	2	0.100E-06	0.300E+01			
36	2	0.100E-06	0.100E+02			
37	1	0.300E-04	0.100E+02			
38	0	0.100E-02	0.100E+02	0.000E+00		
39	1	0.300E-05	0.100E+02			
40	1	0.300E-06	0.100E+02			
41	1	0.360E-07	0.830E+01			
42	1	0.360E-07	0.830E+01			
43	1	0.360E-07	0.830E+01			
44	0	0.100E-03	0.100E+02	0.000E+00		
45	1	0.360E-07	0.830E+01			
46	2	0.300E-05	0.100E+02			
47	2	0.300E-05	0.100E+02			
48	2	0.100E-05	0.300E+01	0.100E+01		

49 0	0.200E-01	0.100E+02	0.000E+00
50 2	0.100E-06	0.100E+02	
51 2	0.600E-05	0.100E+02	
52 2	0.300E-05	0.100E+02	
53 0	0.300E-03	0.100E+02	0.000E+00
54 2	0.600E-05	0.100E+02	
55 0	0.100E-03	0.100E+02	0.000E+00
56 2	0.600E-05	0.100E+02	
57 0	0.100E-03	0.100E+02	0.000E+00
58 2	0.300E-05	0.100E+02	
59 0	0.200E-01	0.100E+02	0.000E+00
60 2	0.100E-06	0.300E+01	
61 2	0.100E-05	0.300E+01	0.100E+01
62 2	0.100E-04	0.100E+02	
63 0	0.200E-01	0.100E+02	0.000E+00
64 1	0.100E-06	0.100E+02	
65 2	0.300E-05	0.100E+02	
66 2	0.100E-06	0.100E+02	
67 2	0.100E-06	0.100E+02	
68 2	0.100E-06	0.100E+02	
69 2	0.100E-06	0.300E+01	
70 2	0.100E-06	0.100E+02	
71 2	0.100E-06	0.100E+02	
72 2	0.100E-06	0.300E+01	
73 2	0.100E-06	0.100E+02	
74 2	0.100E-06	0.300E+01	
75 2	0.100E-06	0.100E+02	
76 2	0.100E-06	0.300E+01	
77 2	0.100E-05	0.300E+01	0.100E+01
78 1	0.300E-05	0.100E+02	
79 2	0.100E-06	0.100E+02	
80 2	0.100E-06	0.100E+02	
81 0	0.100E-02	0.300E+01	0.000E+00
82 0	0.100E-03	0.100E+02	0.000E+00
83 0	0.100E-02	0.300E+01	0.000E+00
84 1	0.100E-06	0.300E+01	
85 1	0.100E-06	0.100E+02	
86 1	0.100E-06	0.300E+01	
87 1	0.300E-05	0.100E+02	
88 2	0.100E-06	0.100E+02	
89 2	0.100E-06	0.100E+02	
90 0	0.100E-02	0.300E+01	0.000E+00
91 0	0.100E-03	0.100E+02	0.000E+00
92 0	0.100E-02	0.300E+01	0.000E+00
93 1	0.100E-06	0.300E+01	
94 1	0.100E-06	0.100E+02	
95 1	0.100E-06	0.300E+01	
96 2	0.100E-05	0.300E+01	0.100E+01
97 2	0.100E-04	0.100E+02	
98 2	0.300E-05	0.100E+02	
99 1	0.300E-05	0.100E+02	
100 1	0.300E-05	0.100E+02	
101 2	0.100E-04	0.100E+02	
102 0	0.300E-02	0.100E+02	0.000E+00
103 0	0.300E-02	0.100E+02	0.000E+00
104 1	0.100E-06	0.100E+02	
105 0	0.300E-02	0.100E+02	0.000E+00
106 0	0.300E-02	0.100E+02	0.000E+00
107 0	0.300E-03	0.100E+02	0.000E+00
108 2	0.230E-05	0.100E+02	

109	2	0.100E-06	0.300E+01	0.000E+00
110	2	0.100E-06	0.100E+02	0.000E+00
111	2	0.100E-06	0.300E+01	0.000E+00
112	2	0.100E-06	0.100E+02	0.000E+00
113	2	0.100E-04	0.100E+02	0.000E+00
114	2	0.100E-06	0.300E+01	0.000E+00
115	2	0.100E-06	0.100E+02	0.000E+00
116	2	0.100E-06	0.300E+01	0.000E+00
117	2	0.100E-06	0.100E+02	0.000E+00
118	0	0.100E-03	0.100E+02	0.000E+00
119	0	0.100E-02	0.300E+01	0.000E+00
120	0	0.100E-03	0.100E+02	0.000E+00
121	0	0.100E-03	0.100E+02	0.000E+00
122	0	0.100E-02	0.300E+01	0.000E+00
123	0	0.100E-03	0.100E+02	0.000E+00
124	0	0.100E-02	0.300E+01	0.000E+00
125	0	0.100E-03	0.100E+02	0.000E+00
126	1	0.100E-06	0.100E+02	0.000E+00
127	1	0.100E-06	0.300E+01	0.000E+00
128	1	0.100E-06	0.100E+02	0.000E+00
129	1	0.100E-06	0.100E+02	0.000E+00
130	1	0.100E-06	0.300E+01	0.000E+00
131	1	0.100E-06	0.100E+02	0.000E+00
132	1	0.100E-06	0.300E+01	0.000E+00
133	1	0.100E-06	0.300E+01	0.000E+00
134	2	0.100E-04	0.100E+02	0.000E+00
135	2	0.100E-06	0.300E+01	0.000E+00
136	2	0.100E-06	0.100E+02	0.000E+00
137	2	0.100E-06	0.300E+01	0.000E+00
138	2	0.100E-06	0.100E+02	0.000E+00
139	0	0.100E-03	0.100E+02	0.000E+00
140	0	0.100E-02	0.300E+01	0.000E+00
141	0	0.100E-03	0.100E+02	0.000E+00
142	0	0.100E-03	0.100E+02	0.000E+00
143	0	0.100E-02	0.300E+01	0.000E+00
144	0	0.100E-03	0.100E+02	0.000E+00
145	0	0.100E-02	0.300E+01	0.000E+00
146	0	0.100E-02	0.300E+01	0.000E+00
147	1	0.100E-06	0.100E+02	0.000E+00
148	1	0.100E-06	0.300E+01	0.000E+00
149	1	0.100E-06	0.100E+02	0.000E+00
150	1	0.100E-06	0.100E+02	0.000E+00
151	1	0.100E-06	0.300E+01	0.000E+00
152	1	0.100E-06	0.100E+02	0.000E+00
153	1	0.100E-06	0.300E+01	0.000E+00
154	1	0.100E-06	0.300E+01	0.000E+00
155	2	0.100E-04	0.100E+02	0.000E+00
156	0	0.300E-02	0.100E+02	0.000E+00
157	0	0.300E-02	0.100E+02	0.000E+00
158	1	0.100E-06	0.100E+02	0.000E+00
159	0	0.300E-02	0.100E+02	0.000E+00
160	0	0.300E-02	0.100E+02	0.000E+00
161	1	0.100E-06	0.100E+02	0.000E+00
162	1	0.300E-05	0.100E+02	0.000E+00
163	1	0.100E-06	0.100E+02	0.000E+00
164	0	0.300E-02	0.100E+02	0.000E+00
165	2	0.100E-06	0.100E+02	0.000E+00
166	0	0.100E-02	0.300E+01	0.000E+00
167	0	0.300E-02	0.100E+02	0.000E+00
168	1	0.100E-06	0.300E+01	0.000E+00

169	1	0.100E-06	0.100E+02	
170	2	0.100E-05	0.300E+01	0.100E+01
171	0	0.300E-03	0.100E+02	0.000E+00
172	1	0.300E-05	0.100E+02	
173	1	0.500E-05	0.100E+02	
174	1	0.300E-05	0.100E+02	
175	0	0.300E-03	0.100E+02	0.000E+00
176	1	0.300E-05	0.100E+02	
177	2	0.100E-05	0.300E+01	0.100E+01
178	0	0.100E-02	0.300E+01	0.000E+00
179	1	0.100E-06	0.300E+01	
180	2	0.100E-04	0.100E+02	
181	2	0.100E-04	0.100E+02	
182	2	0.100E-05	0.300E+01	0.100E+01
183	2	0.300E-05	0.100E+02

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NO EXHAUST FLOW

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NO EXHAUST FLOW	
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NO EXHAUST FLOW	
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NO EXHAUST FLOW	
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NO FORCED EXHAUST FLOW (SHIELD) (COOLING)	
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24	25	25			
NO FORCED EXHAUST FLOW (SHIELD) (COOLING)					
Y1000	NO FORCED	EXHAUST FLOW	(Shield)	(Cooling)	
	+ 2 0 Y2000	Y3000			
Y2000	NO EXHAUST	FLOW PATH WITH RX BUILDING	ISOLATION		
	* 1 1 Y2100	KDM000UW			
Y3000	NO EXHAUST	FLOW PATH W/O RX BUILDING	ISOLATION		
	+ 1 2 Y4000	KAV320SQ KAV23ASQ			
Y2100	ISOLATION	VALVE 323B NOT OPEN	R13-VR-323B		
	+ 0 2 KAV23BSP	KAV23BSQ			
Y4000	NO STACK	EXHAUST SYSTEM FLOW			
	* 2 0 Y4100	Y4200			
Y4100	NO EXHAUST	FLOW IN NORMAL (LEG 1)			
	+ 4 0 Y4110	Y4120 Y4130 Y4140			
Y4200	NO EXHAUST	FLOW IN BACKUP (LEG 2)			
	+ 4 0 Y4210	Y4220 Y4230 Y4240			
Y4110	FAN FM-56	NOT RUNNING	FM-56		
	+ 0 2 EBS001UW	YFN056OR			
Y4120	DAMPER DA-1A	NOT OPEN	DA-1A		
	+ 1 1 Y4121	YDMD1AOW			
Y4130	DAMPER D-1B	NOT OPEN	D-1B		
	+ 1 1 Y4131	YDMD1BOW			
Y4140	DAMPER DA-1C	NOT OPEN	DA-1C		
	+ 1 1 Y4141	YDMD1COW			
Y4210	FAN FM-58	NOT RUNNING	FM-58		
	+ 0 3 EBS002UW	YFN058OR YFN058OS			
Y4220	DAMPER DA-2A	NOT OPEN	DA-2A		
	+ 1 1 Y4221	YDMD2AOW			
Y4230	DAMPER D-2B	NOT OPEN	D-2B		
	+ 1 1 Y4231	YDMD2BOW			
Y4240	DAMPER DA-2C	NOT OPEN	DA-2C		
	* 2 0 Y4241	Y4242			
Y4121	DAMPER DA-1A	FAILURE W/O RECOVERY	DA-1A		
	* 0 2 YDMD1AOQ	YHU001OL			
Y4131	DAMPER D-1B	FAILURE W/O RECOVERY	D-1B		
	* 0 2 YDMD1BOQ	YHU001OL			
Y4141	DAMPER DA-1C	FAILURE W/O RECOVERY	DA-1C		
	* 0 2 YDMD1COQ	YHU001OL			
Y4221	DAMPER DA-2A	FAILURE W/O RECOVERY	DA-2A		
	* 1 1 Y4222	YHU001OL			
Y4231	DAMPER D-2B	FAILURE W/O RECOVERY	D-2B		
	* 1 1 Y4232	YHU001OL			
Y4241	DAMPER DA-2C	FAILURE W/O RECOVERY	DA-2C		
	* 1 1 Y4242	YHU001OL			
Y4242	DAMPER DA-2C	EXTERNAL FAILURE	DA-2C		
	+ 0 2 YDMD2COP	YDMD2COQ			
Y4222	DAMPER DA-2A	EXTERNAL FAILURE	DA-2A		
	+ 0 2 YDMD2AOP	YDMD2AOQ			
Y4232	DAMPER D-2B	EXTERNAL FAILURE	D-2B		
	+ 0 2 YDMD2BOP	YDMD2BOQ			
KDM000UW	0 0	1 REACTOR BUILDING ISOLATION SIGNAL			
KAV320SQ	0 0	2 ISOLATION VALVE 320 FAILS TO REMAIN OPEN	R13-VR-320		
KAV23ASQ	0 0	3 ISOLATION VALVE 323A FAILS TO REMAIN OPEN	R13-VR-323A		
KAV23BSP	0 0	4 ISOLATION VALVE 323B FAILS TO OPEN	R13-VR-323B		
KAV23BSQ	0 0	5 ISOLATION VALVE 323B FAILS TO REMAIN OPEN	R13-VR-323B		
EBS001UW	0 0	6 NO POWER AT OUTPUT OF BREAKER MCC E1 1B			
YFN056OR	0 0	7 FAN FM-56 FAILS TO RUN	FM-56		
YDMD1AOW	0 0	8 DAMPER DA-1A FAILS TO REMAIN OPEN	INTERNALLY		
YDMD1BOW	0 0	9 FAN VANES D-1B FAIL TO REMAIN OPEN	INTERNALLY		
YDMD1COW	0 0	10 DAMPER DA-1C FAILS TO REMAIN OPEN	INTERNALLY		

EBS002UW 0 0 11 NO POWER AT OUTPUT OF BREAKER MCC E2 1B
 YFN058OR 0 0 12 FAN FM-58 FAILS TO RUN FM-58
 YFN058OS 0 0 13 FAN FM-58 FAILS TO START FM-58
 YDMD2AOW 0 0 14 DAMPER DA-2A FAILS TO REMAIN OPEN INTERNALLY
 YDMD2BOW 0 0 15 FAN VANES D-2B FAIL TO REMAIN OPEN INTERNALLY
 YDMD1AOQ 0 0 16 DAMPER DA-1A FAILS TO REMAIN OPEN EXTERNALLY
 YHU001OL 0 0 17 NO RECOVERY ACTION TO MANUALLY OPEN DAMPER
 YDMD1BOQ 0 0 18 FAN VANES D-1B FAIL REMAIN OPEN EXTERNALLY
 YDMD1COQ 0 0 19 DAMPER DA-1C FAILS TO REMAIN OPEN EXTERNALLY
 YDMD2COP 0 0 20 DAMPER DA-2C FAILS TO OPEN EXTERNALLY
 YDMD2COQ 0 0 21 DAMPER DA-2C FAILS TO REMAIN OPEN EXTERNALLY
 YDMD2AOP 0 0 22 DAMPER DA-2A FAILS TO OPEN EXTERNALLY
 YDMD2AOQ 0 0 23 DAMPER DA-2A FAILS TO REMAIN OPEN EXTERNALLY
 YDMD2BOP 0 0 24 FAN VANES D-2B FAIL TO OPEN EXTERNALLY
 YDMD2BOQ 0 0 25 FAN VANES D-2B FAIL TO REMAIN OPEN EXTERNALLY

0.744E+03

0.744E+03

1	2	0.100E-03	0.300E+02	
2	1	0.640E-07	0.100E+02	
3	2	0.640E-07	0.100E+02	
4	0	0.200E-02	0.100E+02	0.000E+00
5	1	0.100E-06	0.100E+02	
6	2	0.160E-05	0.370E+01	
7	2	0.200E-04	0.300E+01	
8	1	0.360E-07	0.830E+01	
9	1	0.360E-07	0.830E+01	
10	1	0.360E-07	0.830E+01	
11	1	0.160E-05	0.370E+01	
12	1	0.200E-04	0.300E+01	
13	0	0.300E-03	0.300E+01	0.000E+00
14	1	0.360E-07	0.100E+02	
15	1	0.360E-07	0.100E+02	
16	2	0.640E-07	0.100E+02	
17	0	0.100E+00	0.100E+02	0.000E+00
18	2	0.640E-07	0.100E+02	
19	2	0.640E-07	0.100E+02	
20	0	0.300E-02	0.100E+02	0.000E+00
21	1	0.640E-07	0.100E+02	
22	0	0.300E-02	0.100E+02	0.000E+00
23	1	0.640E-07	0.100E+02	
24	0	0.300E-02	0.100E+02	0.000E+00
25	1	0.640E-07	0.100E+02

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NO EXHAUST FLOW PATH (THIMBLE) (COOLING)

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NO EXHAUST FLOW PATH (THIMBLE) (COOLING)	
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NO EXHAUST FLOW PATH (THIMBLE) (COOLING)	
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NO EXHAUST FLOW PATH (THIMBLE) (COOLING)				
Y1001	NO EXHAUST FLOW PATH (Thimble)	(Cooling)		
+ 2 0 Y2000	Y3001	ISOLATION		
Y2000	NO EXHAUST FLOW PATH WITH RX BUILDING	ISOLATION		
Y3001	* 1 1 Y2100 KDM0000W	ISOLATION		
Y2100	NO EXHAUST FLOW PATH W/O RX BUILDING	ISOLATION		
+ 1 2 Y4001	KAV320SQ KAV23ASQ	ISOLATION		
Y2100	ISOLATION VALVE 323B NOT OPEN	R13-VR-323B		
Y4001	+ 0 2 KAV23BSP KAV23BSQ	ISOLATION		
Y4101	NO STACK EXHAUST SYSTEM FLOW	ISOLATION		
* 2 0 Y4101	Y4201	ISOLATION		
Y4101	NO EXHAUST FLOW IN NORMAL (LEG 1)	ISOLATION		
+ 3 0 Y4120	Y4130 Y4140	ISOLATION		
Y4201	NO EXHAUST FLOW IN BACKUP (LEG 2)	ISOLATION		
+ 3 0 Y4220	Y4230 Y4240	ISOLATION		
Y4120	DAMPER DA-1A NOT OPEN	ISOLATION		
+ 1 1 Y4121	YDMD1AOW	ISOLATION		
Y4130	DAMPER D-1B NOT OPEN	ISOLATION		
+ 1 1 Y4131	YDMD1BOW	ISOLATION		
Y4140	DAMPER DA-1C NOT OPEN	ISOLATION		
+ 1 1 Y4141	YDMD1COW	ISOLATION		
Y4220	DAMPER DA-2A NOT OPEN	ISOLATION		
+ 1 1 Y4221	YDMD2AOW	ISOLATION		
Y4230	DAMPER D-2B NOT OPEN	ISOLATION		
+ 1 1 Y4231	YDMD2BOW	ISOLATION		
Y4240	DAMPER DA-2C NOT OPEN	ISOLATION		
* 2 0 Y4241	Y4242	ISOLATION		
Y4121	DAMPER DA-1A FAILURE W/O RECOVERY	ISOLATION		
* 0 2 YDMD1AOQ	YHU001OL	ISOLATION		
Y4131	DAMPER D-1B FAILURE W/O RECOVERY	ISOLATION		
* 0 2 YDMD1BOQ	YHU001OL	ISOLATION		
Y4141	DAMPER DA-1C FAILURE W/O RECOVERY	ISOLATION		
* 0 2 YDMD1COQ	YHU001OL	ISOLATION		
Y4221	DAMPER DA-2A FAILURE W/O RECOVERY	ISOLATION		
* 1 1 Y4222	YHU001OL	ISOLATION		
Y4231	DAMPER D-2B FAILURE W/O RECOVERY	ISOLATION		
* 1 1 Y4232	YHU001OL	ISOLATION		
Y4241	DAMPER DA-2C FAILURE W/O RECOVERY	ISOLATION		
* 1 1 Y4242	YHU001OL	ISOLATION		
Y4242	DAMPER DA-2C EXTERNAL FAILURE	ISOLATION		
+ 0 2 YDMD2COP	YDMD2COQ	ISOLATION		
Y4222	DAMPER DA-2A EXTERNAL FAILURE	ISOLATION		
+ 0 2 YDMD2AOP	YDMD2AOQ	ISOLATION		
Y4232	DAMPER D-2B EXTERNAL FAILURE	ISOLATION		
+ 0 2 YDMD2BOP	YDMD2BOQ	ISOLATION		
KDM0000W	0 0 1 REACTOR BUILDING ISOLATION SIGNAL	ISOLATION		
KAV320SQ	0 0 2 ISOLATION VALVE 320 FAILS TO REMAIN OPEN	ISOLATION		
KAV23ASQ	0 0 3 ISOLATION VALVE 323A FAILS TO REMAIN OPEN	ISOLATION		
KAV23BSP	0 0 4 ISOLATION VALVE 323B FAILS TO REMAIN OPEN	ISOLATION		
KAV23BSQ	0 0 5 ISOLATION VALVE 323B FAILS TO REMAIN OPEN	ISOLATION		
YDMD1ACOW	0 0 6 DAMPER DA-1A FAILS TO REMAIN OPEN	ISOLATION		
YDMD1BOW	0 0 7 FAN VANES D-1B FAIL TO REMAIN OPEN	ISOLATION		
YDMD1COW	0 0 8 DAMPER DA-1C FAILS TO REMAIN OPEN	ISOLATION		
YDMD2BOW	0 0 9 DAMPER DA-2A FAILS TO REMAIN OPEN	ISOLATION		
YDMD2COW	0 0 10 FAN VANES D-2B FAIL TO REMAIN OPEN	ISOLATION		
YDMD1AOQ	0 0 11 DAMPER DA-1A FAILS TO REMAIN OPEN EXTERNALLY	ISOLATION		
YHU001OL	0 0 12 NO RECOVERY ACTION TO MANUALLY OPEN DAMPER	ISOLATION		
YDMD1BOQ	0 0 13 FAN VANES D-1B FAIL REMAIN OPEN EXTERNALLY	ISOLATION		
YDMD1COQ	0 0 14 DAMPER DA-1C FAILS TO REMAIN OPEN EXTERNALLY	ISOLATION		

YDM2COP	0	0	15	DAMPER DA-2C FAILS TO OPEN EXTERNALLY	
YDM2COQ	0	0	16	DAMPER DA-2C FAILS TO REMAIN OPEN EXTERNALLY	
YDM2AOP	0	0	17	DAMPER DA-2A FAILS TO OPEN EXTERNALLY	
YDM2AOQ	0	0	18	DAMPER DA-2A FAILS TO REMAIN OPEN EXTERNALLY	
YDM2BOP	0	0	19	FAN VANES D-2B FAIL TO OPEN EXTERNALLY	
YDM2BOQ	0	0	20	FAN VANES D-2B FAIL TO REMAIN OPEN EXTERNALLY	
* 0.744E+03					
0.744E+03					
1	2	0.100E-03	0.300E+02		
2	1	0.640E-07	0.100E+02		
3	2	0.640E-07	0.100E+02		
4	0	0.200E-02	0.100E+02	0.000E+00	
5	1	0.100E-06	0.100E+02		
6	1	0.360E-07	0.830E+01		
7	1	0.360E-07	0.830E+01		
8	1	0.360E-07	0.830E+01		
9	1	0.360E-07	0.100E+02		
10	1	0.360E-07	0.100E+02		
11	2	0.640E-07	0.100E+02		
12	0	0.100E+00	0.100E+02	0.000E+00	
13	2	0.640E-07	0.100E+02		
14	2	0.640E-07	0.100E+02		
15	0	0.300E-02	0.100E+02	0.000E+00	
16	1	0.640E-07	0.100E+02		
17	0	0.300E-02	0.100E+02	0.000E+00	
18	1	0.640E-07	0.100E+02		
19	0	0.300E-02	0.100E+02	0.000E+00	
20	1	0.640E-07	0.100E+02

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THIMBLE COOLING DECAY HEAT LESS THAN 31 KW	
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THIMBLE COOLING DECAY HEAT LESS THAN 31 KW	
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THIMBLE COOLING DECAY HEAT LESS THAN 31 KW	
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THIMBLE COOLING DECAY HEAT LESS THAN 31 KW	
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THIMBLE COOLING DECAY HEAT LESS THAN 31 KW
 T1000 THIMBLE COOLING DECAY HEAT REMOVAL LESS THAN 31 KW
 + 2 3 T2000 T3000 TFLJINFU TXVISOFU TXVBYDFU
 T2000 NO FLOW IN THIMBLE COOLING LEG A
 + 2 0 T2100 T0770
 T3000 NO FLOW IN THIMBLE COOLING LEG B
 + 2 0 T3100 T3300
 T2100 NO FLOW IN LEG A AND NO CROSS-CONNECT RECOVERY ACTION
 * 1 1 T2110 THU01BSZ
 T0770 SHIELD COOLING CROSS-CONNECT ACTION NOT EFFECTIVE
 + 0 3 H8100 TXV770SP TXV770SQ
 T3100 NO FLOW IN LEG B AND NO CROSS-CONNECT RECOVERY ACTION
 * 1 1 T3110 THU01BSZ
 T3300 LEG B SPECIFIC CROSS-CONNECT ACTION NOT EFFECTIVE
 + 1 2 T0770 TXV752SQ TXV753SQ
 T2110 NO NORMAL FLOW IN LEG A
 + 1 1 T4000 TXV753SQ
 T3110 NO NORMAL FLOW IN LEG B
 + 1 1 T4000 TXV752SQ
 T4000 NO THIMBLE COOLING EXHAUST FLOW
 + 2 1 T5000 T6000 Y1001
 T5000 NO FLOW IN FILTER SECTION
 * 3 0 T5100 T5200 T5300
 T6000 NO FLOW IN FAN SECTION
 * 2 0 T6100 T6200
 T5100 NO FLOW THROUGH HEPA FILTER #1
 + 0 4 TXV01ISQ TXV01OSQ TFL001SE TFLHEPSU
 T5200 NO FLOW THROUGH HEPA FILTER #2
 + 0 4 TXV02ISQ TXV02OSQ TFL002SE TFLHEPSU
 T5300 NO FLOW THROUGH HEPA FILTER #3
 + 0 4 TXV03ISQ TXV03OSQ TFL003SE TFLHEPSU
 T6100 NO FLOW IN RUNNING FAN SECTION R8-F-709
 + 2 3 T7280 T7090 TDM762SQ TDM749SQ TDM729SQ
 T6200 NO FLOW IN STANDBY FAN SECTION R8-F-710
 + 5 0 T7290 T7100 T7610 T0510 T7285
 T7280 DISC DAMPER 728 DOES NOT REMAIN OPEN R8-DM-728
 + 1 2 T0091 I0900 TDM728SQ
 T7090 FAN 709 IN EXHAUST LEG 1 NOT RUNNING
 + 2 3 T0204 T7091 TEN709SR TWR709TB TWR709TC
 T7290 DISCHARGE DAMPER 729 CLOSED AND REMAINS CLOSED
 + 1 1 T7293 TDM729SQ
 T7100 FAN 710 IN EXHAUST LEG 2 NOT RUNNING
 + 2 4 T0203 T7101 TEN710SR TEN710SS TWR710TB TWR710TC
 T7610 THROTTLING DAMPER 761 IS NOT OPEN R8-DM-761
 + 0 2 TDM761SQ TSW761ST
 T0510 ISOL DAMPER 510 IS NOT OPEN R8-DM-751
 + 0 2 TDM751SL TDM751SQ
 T7285 DISCHARGE DAMPER 728 OPEN AND REMAINS OPEN
 + 1 1 T7286 TDM728SQ
 T0091 SOLENOID VALVE RSV 91 DOES NOT REMAIN OPEN RSV-91
 + 1 1 T7499 TSV091SQ
 T0204 NO ELECTRICAL POWER @ OUTPUT OF BREAKER MCC R2A-2C
 + 1 1 T0206 E0300
 T7091 FAN 709 RUN CIRCUIT NOT COMPLETE R8-FN-709
 + 1 2 T7092 TRE52XSO TXT052SW
 T7293 DISCHARGE DAMPER 729 CLOSED AND NOT RECOVERED
 * 1 1 T7294 HHU003SL
 T0203 NO ELECTRICAL POWER @ OUTPUT OF BREAKER MCC R2B-4C
 + 1 1 T0205 E0360

T7101	FAN 710	STANDBY CIRCUIT NOT COMPLETE	R8-FN-710
	+ 2 2 T7102	T7103 TRE53XSO TXT053SW	
T7286	DISCHARGE DAMPER 728	OPEN AND NOT RECOVERED	
	* 1 1 T7287	HU003SL	
T7499	MAN DAMPER 749	INDICATES NOT FULLY OPEN DM-749	
	+ 0 2 TTZ749SW	TDM749SQ	
T0206	BREAKER	MCC R2A-3A OPEN AND	NO RECOVERY
	* 0 2 ECB02CYB	THU002YL	
T7092	CONTROL CONTACT CR-91-2	OPEN	FN-709/CR-91-2
	+ 1 1 T7289	TRE091SO	
T7294	DISCHARGE DAMPER 729	IS NOT OPEN R8-DM-729	
	+ 2 0 T7291	T7292	
T0205	BREAKER	MCC R2B-3A OPEN AND	NO RECOVERY
	* 0 2 ECB03AYB	THU002YL	
T7102	CONTROL CONTACT CR-92-2	OPEN	FN-710/CR-92-2
	+ 1 1 T7299	TRE092SO	
T7103	FAN 710 STBY	PRESSURE OR TIME DELAY	FAILS
	+ 0 2 TPR01ASW	TRE53TSN	
T7287	DISCHARGE DAMPER 728	NOT SHUT R8-DM-728	
	+ 0 2 TDM728SN	TDM728SO	
T7289	DISC DAMPER 728	DOES NOT	INDICATE OPEN R8-DM-728
	+ 1 1 T7280	TTZ728SW	
T7291	DISCHARGE	DAMPER 729 DOES NOT OPEN	R8-DM-729
	+ 1 2 T0920	IO900 TDM729SP	
T7292	DISC DAMPER 729	DOES NOT	REMAIN OPEN R8-DM-729
	+ 1 2 T0092	IO900 TDM729SQ	
T7299	DISC DAMPER 729	DOES NOT	INDICATE OPEN R8-DM-729
	+ 1 1 T7290	TTZ729SW	
T0920	SOLENOID VALVE RSV 92	DOES NOT OPEN	DM-729/RSV-92
	+ 1 1 T7518	TSV092SP	
T0092	SOLENOID VALVE RSV 92	DOES NOT REMAIN OPEN	RSV-92
	+ 1 1 T7519	TSV092SQ	
T7518	MAN DAMPER 751	NOT INDICATING NOT FULLY OPEN DM-751	
	+ 0 2 TTZ751SW	TDM751SL	
T7519	MAN DAMPER 751	INDICATES NOT FULLY OPEN DM-751	
	+ 0 2 TTZ751SW	TDM751SQ	
H8100	3 0	1 NO EXHAUST	FLOW
Y1001	3 0	1 NO EXHAUST	FLOW PATH (Thimble)
IO900	3 0	1 Loss of Instrument Air to Reactor Building	(Cooling)
E0300	3 0	1 LOSS OF 480 VAC EMERGENCY BUS R-2A	
E0360	3 0	1 LOSS OF 480 VAC EMERGENCY BUS R-2B	
TFLJINFU	0 0	2 THIMBLE INLET FILTERS	PLUGGED DUE TO COMMON CAUSE
TXVISOFU	0 0	3 INLET ISOLATION VALVES FAIL TO REMAIN OPEN	- COMMON CAUSE
TXVBYPFU	0 0	4 J AND O BYPASS VALVES FAIL TO REMAIN CLOSED	- COMMON CAUSE
THU01BS2	0 0	5 NO RECOVERY ACTION TO CROSS-CONNECT SHIELD COOLING	
TXV770SP	0 0	6 SHIELD COOLING X-CONN FAILS TO OPEN	R8-DM-770
TXV770SQ	0 0	7 SHIELD COOLING X-CONN FAILS TO REMAIN OPEN	R8-DM-770
TXV752SQ	0 0	8 BALANCE DAMPER LEG B FAILS TO REMAIN OPEN	R8-DM-752
TXV753SQ	0 0	9 BALANCE DAMPER LEG A FAILS TO REMAIN OPEN	R8-DM-753
TXV011SQ	0 0	10 HEPA FILTER #1 INLET FAILS TO REMAIN OPEN	
TXV010SQ	0 0	11 HEPA FILTER #1 OUTLET FAILS TO REMAIN OPEN	
TFL001SE	0 0	12 HEPA FILTER #1	PLUGGED
TFLHEPSU	0 0	13 HEPA FILTERS	PLUGGED DUE TO COMMON CAUSE
TXV021SQ	0 0	14 HEPA FILTER #2 INLET FAILS TO REMAIN OPEN	
TXV020SQ	0 0	15 HEPA FILTER #2 OUTLET FAILS TO REMAIN OPEN	
TFL002SE	0 0	16 HEPA FILTER #2	PLUGGED
TXV031SQ	0 0	17 HEPA FILTER #3 INLET FAILS TO REMAIN OPEN	
TXV030SQ	0 0	18 HEPA FILTER #3 OUTLET FAILS TO REMAIN OPEN	
TFL003SE	0 0	19 HEPA FILTER #3	PLUGGED
TDM762SQ	0 0	20 THROTTLE DMP 762 FAILS TO REMA	EN R8-DM-762

TDM749SQ	0	0	21	ISOL DAMPER 749 FAILS TO REMAIN OPEN	R8-DM-749
TDM729SQ	0	0	22	DISC DAMPER 729 FAILS TO REMAIN CLOSED	R8-DM-729
TDM728SQ	0	0	23	DISC DAMPER 728 FAILS TO REMAIN OPEN	R8-DM-728
TFR709SR	0	0	24	TURBO FAN 709 FAILS TO RUN	R8-FN-709
TWR709TB	0	0	25	FAN 709 POWER CABLE FAILS: OPEN CIRCUIT	
TWR709TC	0	0	26	FAN 709 POWER CABLE FAILS: SHORT TO GROUND	
TDM729SW	0	0	27	DAMPER 729 INTERNAL FAILURE	R8-DM-729
TFR710SR	0	0	28	TURBO FAN 710 FAILS TO RUN	R8-FN-710
TFR710SS	0	0	29	TURBO FAN 710 FAILS TO START	R8-FN-710
TWR710TB	0	0	30	FAN 710 POWER CABLE FAILS: OPEN CIRCUIT	
TWR710TC	0	0	31	FAN 710 POWER CABLE FAILS: SHORT TO GROUND	
TDM761SQ	0	0	32	THROTTLE DMP 761 FAILS TO REMAIN OPEN	R8-DM-761
TSW761ST	0	0	33	THROTTLE DMR 761 SWITCH IN WRONG POSITION	DM-761/RM93
TDM751SL	0	0	34	ISOLATION DAMPER 751 LEFT CLOSED	R8-DM-751
TDM751SQ	0	0	35	ISOL DAMPER 751 FAILS TO REMAIN OPEN	
TDM728SW	0	0	36	DAMPER 728 INTERNAL FAILURE	R8-DM-728
TSV091SQ	0	0	37	SOLENOID VALVE RSV-91 FAILS TO REMAIN OPEN	RSV-91
TRE52XSO	0	0	38	CONTROL RELAY RM-52X COIL FAILURE	FN-709/CR-52X
TXT052SW	0	0	39	RM52 480V/120V 250VA STARTER TRANSFORMER FAILS	
HHU003SL	0	0	40	NO RECOVERY ACTION TO USE DAMPER MANUAL HANDWHEEL	
TRE53XSO	0	0	41	CONTROL RELAY RM-53X COIL FAILURE	FN-710/CR-53X
TXT053SW	0	0	42	RM53 480V/120V 250VA STARTER TRANSFORMER FAILS	
ECB02CYB	0	0	43	DM749 POSITION INDICATOR ERROR	DM-749/RSV-91
THU002YL	0	0	44	CIRCUIT BREAKER SPURIOUS TRIP	MCC R2-4C
TRE091SO	0	0	45	NO RECOVERY ACTION TO RESET BREAKER	
ECB03AYB	0	0	46	CONTROL RELAY RM-91 COIL FAILURE	FN-709/CR-91
TRE092SO	0	0	47	CIRCUIT BREAKER SPURIOUS TRIP	MCC R2-3A
TPR01ASW	0	0	48	CONTROL RELAY RM-92 COIL FAILURE	FN-710/CR-92
TRE53TSN	0	0	49	PRESSURE TRANSDUCER FAILS TO CLOSE	DP-PS-1A
TDM728SN	0	0	50	TIME DELAY RELAY TD-53A CONTACT FAILS	
TDM728SO	0	0	51	DISCHARGE DAMPER 728 FAILS TO REMAIN CLOSED	R8-DM-728
TT2728SW	0	0	52	DISC DAMPER 728 FAILS TO REMAIN CLOSED	R8-DM-728
TDM729SF	0	0	53	DM728 POSITION INDICATOR ERROR	DM-728
TDM729SQ	0	0	54	DISCHARGE DAMPER 729 FAILS TO OPEN	R8-DM-729
TT2729SW	0	0	55	DISC DAMPER 729 FAILS TO REMAIN OPEN	R8-DM-729
TSV092SP	0	0	56	DM729 POSITION INDICATOR ERROR	DM-729
TSV092SQ	0	0	57	SOLENOID VALVE RSV-92 FAILS TO OPEN	RSV-92
TT2751SW	0	0	58	SOLENOID VALVE RSV-92 FAILS TO REMAIN OPEN	RSV-92
	0	0	59	DM751 POSITION INDICATOR ERROR	DM-751/RSV-92
	0.744E+03				
	0.744E+03				
	1 0	0.100E-09	0.100E+02	0.000E+00	
	2 2	0.300E-05	0.100E+02		
	3 1	0.100E-07	0.100E+02		
	4 1	0.100E-07	0.100E+02		
	5 0	0.200E-01	0.100E+02	0.000E+00	
	6 0	0.200E-03	0.100E+02	0.000E+00	
	7 1	0.100E-06	0.100E+02		
	8 1	0.100E-06	0.100E+02		
	9 1	0.100E-06	0.100E+02		
	10 2	0.100E-06	0.100E+02		
	11 2	0.100E-06	0.100E+02		
	12 2	0.300E-04	0.100E+02		
	13 2	0.300E-05	0.100E+02		
	14 2	0.100E-06	0.100E+02		
	15 2	0.100E-06	0.100E+02		
	16 2	0.300E-04	0.100E+02		
	17 2	0.100E-06	0.100E+02		
	18 2	0.100E-06	0.100E+02		
	19 2	0.300E-04	0.100E+02		

20	2	0.100E-06	0.100E+02	
21	2	0.100E-06	0.100E+02	
22	2	0.100E-06	0.100E+02	
23	2	0.100E-06	0.100E+02	
24	2	0.300E-04	0.100E+02	
25	2	0.300E-05	0.100E+02	
26	2	0.300E-06	0.100E+02	
27	1	0.360E-07	0.830E+01	
28	1	0.300E-04	0.100E+02	0.000E+00
29	0	0.100E-02	0.100E+02	
30	1	0.300E-05	0.100E+02	
31	1	0.300E-06	0.100E+02	
32	1	0.100E-06	0.100E+02	
33	0	0.100E-03	0.100E+02	0.000E+00
34	0	0.100E-03	0.100E+02	0.000E+00
35	1	0.100E-06	0.100E+02	
36	1	0.360E-07	0.830E+01	
37	2	0.100E-06	0.300E+01	
38	2	0.300E-05	0.100E+02	
39	2	0.100E-05	0.300E+01	
40	0	0.200E-01	0.100E+02	0.000E+00
41	1	0.300E-05	0.100E+02	
42	2	0.100E-05	0.300E+01	
43	2	0.300E-05	0.100E+02	
44	2	0.100E-04	0.100E+02	
45	0	0.200E-01	0.100E+02	0.000E+00
46	2	0.300E-05	0.100E+02	
47	2	0.100E-04	0.100E+02	
48	1	0.300E-05	0.100E+02	
49	2	0.300E-05	0.100E+02	
50	0	0.300E-03	0.100E+02	0.000E+00
51	0	0.300E-02	0.100E+02	0.000E+00
52	1	0.100E-06	0.100E+02	
53	2	0.300E-05	0.100E+02	
54	0	0.300E-02	0.100E+02	0.000E+00
55	1	0.100E-06	0.100E+02	
56	1	0.300E-05	0.100E+02	
57	1	0.100E-06	0.300E+01	
58	0	0.100E-02	0.300E+01	0.000E+00
59	1	0.300E-05	0.100E+02	

MINIMAL SETS INFORMATION

NSETS	MEAN	IMPORTANCE	MINIMAL CUT SETS	61	2.884E-09	0.0018	TSDC	DCR0010F	BOO02200	Tf	U
1	2.400E-07	0.1527	LONG	62	2.600E-09	0.0017	LF2D				
2	1.920E-07	0.1222	LONG	63	2.550E-09	0.0016	TAIR	9CR0010F	9CR0020F		
3	9.000E-08	0.0573	OCSL	64	2.545E-09	0.0016	TSEC	9HU0000L	9AM4ALAW		
4	9.000E-08	0.0573	OCSL	65	2.545E-09	0.0016	TSEC	9HU0000L	9ARX82QW		
5	8.400E-08	0.0535	SFTR	66	2.545E-09	0.0016	TSEC	9HU0000L	9AM4BLQW		
6	8.000E-08	0.0509	SFTR	67	2.520E-09	0.0016	LF2A	RHU001CL	RCZ120QU		
7	5.040E-08	0.0321	LF2A	68	2.400E-09	0.0015	RIFS	RCZC00QU	BOO992UW		
8	4.032E-08	0.0257	LF2A	69	2.400E-09	0.0015	LF1A	RCZC00QU	RHU002CL	RAM221QU	
9	3.200E-08	0.0204	SFTR	70	2.310E-09	0.0015	RISB	RHU302QL	BOO02200	KTN003UW	
10	3.000E-08	0.0191	OCSL	71	2.212E-09	0.0014	TSDC	DCR0010F	9CR0020E		
11	2.910E-08	0.0185	RIFL	72	2.138E-09	0.0014	TSEC	9CR0010E	9CR0020F		
12	2.858E-08	0.0182	TSDC	73	2.138E-09	0.0014	TSEC	CSR301RU			
13	2.858E-08	0.0182	TSDC	74	2.100E-09	0.0013	LF2A	CSR10GRU			
14	2.587E-08	0.0165	TSDC	75	2.100E-09	0.0013	LF2A	RREC00QU			
15	2.400E-08	0.0153	OCSL	76	1.920E-09	0.0012	RIFS	RREC00QU	BOO992UW		
16	1.920E-08	0.0122	OCSL	77	1.920E-09	0.0012	LF1A	RHU002CL	RCZ220QU		
17	1.848E-08	0.0118	RISB	78	1.848E-09	0.0012	RISB	RHU001CL	ROO1AB00		
18	1.478E-08	0.0094	RISB	79	1.680E-09	0.0011	LONG	RHU001CL	ROO1AB00		
19	1.260E-08	0.0080	SFPB	80	1.641E-09	0.0010	TSHC	DCR0010F	9CR0020F		
20	1.260E-08	0.0080	SFPB	81	1.495E-09	0.0010	TBOP	9HU0000L	9AM4BLQW		
21	1.200E-08	0.0076	LONG	82	1.495E-09	0.0010	TBOP	9HU0000L	9ARX82QW		
22	1.200E-08	0.0076	OCSL	83	1.495E-09	0.0010	TBOP	RHU002CL	RRE220QU		
23	1.128E-08	0.0072	LF2B	84	1.478E-09	0.0009	RISB	RHU002CL	RRE220QU		
24	1.100E-08	0.0070	LCOL	85	1.260E-09	0.0008	SFPB	FPP991RF	FPP003RF		
25	1.020E-08	0.0065	SHDL	86	1.256E-09	0.0008	TBOP	9CR0010E	9CR0020F	FPP112RF	CND003RW
26	1.002E-08	0.0064	TSDC	87	1.200E-09	0.0008	LF1C	9CR0010F	9CR0020E		
27	1.000E-08	0.0064	LOCP	88	1.200E-09	0.0008	SHDL	9HU0000L	9RE53ACP		
28	1.000E-08	0.0064	LONG	89	1.200E-09	0.0008	SHDL	9HU0000L	9ARX82QW		
29	1.000E-08	0.0064	LONG	90	1.200E-09	0.0008	SHDL	9HU0000L	9ARX82QW		
30	1.000E-08	0.0064	LOCP	91	1.200E-09	0.0008	OCSL	CCL00PRU	RHU002CL		
31	9.537E-09	0.0061	TSDC	92	1.200E-09	0.0008	LONG	9CR0010E	9CR0020E		
32	9.537E-09	0.0061	TSDC	93	1.129E-09	0.0007	SHDL	DCR0010F	DCR0020F		
33	9.024E-09	0.0057	LF2B	94	1.099E-09	0.0007	SHDL	DCR0010F	DCR0020F		
34	8.630E-09	0.0055	TSDC	95	1.099E-09	0.0007	SHDL	RHU302QL	RHU002CL		
35	6.930E-09	0.0044	RISB	96	1.080E-09	0.0007	OCSL	RHU302QL	RHU002CL		
36	6.930E-09	0.0044	RISB	97	1.080E-09	0.0007	OCSL	RHU302QL	RHU002CL		
37	6.427E-09	0.0041	TSEC	98	1.080E-09	0.0007	OCSL	RHU302QL	RHU002CL		
38	4.500E-09	0.0029	HELIF	99	1.080E-09	0.0007	OCSL	RHU302QL	RHU002CL		
39	4.500E-09	0.0029	OELF	100	1.080E-09	0.0007	OCSL	RHU302QL	RHU002CL		
40	4.500E-09	0.0029	OELF	101	1.080E-09	0.0007	OCSL	RHU302QL	RHU002CL		
41	4.040E-09	0.0026	SHDL	102	1.080E-09	0.0007	OCSL	RHU302QL	RHU002CL		
42	4.040E-09	0.0026	SHDL	103	1.080E-09	0.0007	OCSL	RHU302QL	RHU002CL		
43	4.040E-09	0.0026	SHDL	104	1.080E-09	0.0007	OCSL	RHU302QL	RHU002CL		
44	3.774E-09	0.0024	TBOP	105	1.080E-09	0.0007	OCSL	RHU302QL	RHU002CL		
45	3.394E-09	0.0022	SHDL	106	1.080E-09	0.0007	OCSL	RHU302QL	RHU002CL		
46	3.394E-09	0.0022	SHDL	107	1.080E-09	0.0007	SHDS	RHU302QL	RHU002CL		
47	3.360E-09	0.0021	LONG	108	1.047E-09	0.0007	SHDS	VOO000UW	9CR0010F		
48	3.332E-09	0.0021	TSDC	109	1.020E-09	0.0006	LONG	9CR0010F	9CR0020F		
49	3.240E-09	0.0021	OCSL	110	1.010E-09	0.0006	TAIR	9HU0000L	9ARX82QW		
50	3.240E-09	0.0021	OCSL	111	1.010E-09	0.0006	TAIR	9HU0000L	9AM4BLQW		
51	3.240E-09	0.0021	OCSL	112	1.010E-09	0.0006	TAIR	9HU0000L	9AM4ALAW		
52	3.240E-09	0.0021	OCSL	113	9.946E-10	0.0006	SHDL	DCR0010F	DCR0020F		
53	3.240E-09	0.0021	OCSL	114	9.750E-10	0.0006	LONG	RHU002CL	CCD10GRU		
54	3.240E-09	0.0021	OCSL	115	9.623E-10	0.0006	TSDC	DCR0010E	BOO02200		
55	3.240E-09	0.0021	OCSL	116	9.600E-10	0.0006	LF1C	RREC00QU	BOO992UW		
56	3.240E-09	0.0021	OCSL	117	9.400E-10	0.0006	RISB	RHU302QL	RHU002CL		
57	3.240E-09	0.0021	OCSL	118	9.000E-10	0.0006	RIFS	RHU302QL	RHU002CL		
58	3.240E-09	0.0021	OCSL	119	9.000E-10	0.0006	RIFS	RHU302QL	RHU002CL		
59	3.240E-09	0.0021	OCSL	120	8.640E-10	0.0005	LF2F	ACZC00QU	9CR0020F		
60	3.240E-09	0.0021	OCSL	121	8.484E-10	0.0005	TAIR	9CR0010E	9CR0020E		
				122	8.484E-10	0.0005	TAIR	9CR0010F	9CR0020E		
				123	7.700E-10	0.0005	RISB	CSR301RU			
				124	7.700E-10	0.0005	RISB	CSR10GRU			

125	7.693E-10	0.0005	SCRM	DCR0010F	DCR0020F	BOO02200	WFL095YE	188	3.667E-10	0.0002	SHDL	DCR0010E	DCR0020F	BOO02200	HFL717SE
126	7.693E-10	0.0005	SCRM	DCR0010F	DCR0020F	BOO02200	HFL717SE	189	3.667E-10	0.0002	SHDL	DCR0010E	DCR0020F	BOO02200	WFL095YE
127	7.560E-10	0.0005	TSEC	9HU0000L	9ARX82QP			190	3.667E-10	0.0002	SHDL	DCR0010F	DCR0020E	BOO02200	WFL095YE
128	7.560E-10	0.0005	TSEC	9HU0000L	9ARX82QP			191	3.667E-10	0.0002	SHDL	DCR0010F	DCR0020E	BOO02200	WFL095YE
129	7.379E-10	0.0005	TSDC	DCR0010E	BOO02200	KIN0030W		192	3.600E-10	0.0002	OCSL	RHU3020L	RHU002CL	RAM2B1QW	RAM2B1QW
130	7.200E-10	0.0005	LF1B	DCR0010E	BOO0920W			193	3.600E-10	0.0002	OCSL	RHU3020L	RHU002CL	RAM2A1QW	RAM2A1QW
131	7.112E-10	0.0005	TSEC	9CR0001OE	9CR0002OE			194	3.600E-10	0.0002	OCSL	RHU3020L	RHU002CL	RAM2A1QW	RAM2A1QW
132	7.056E-10	0.0004	LF2A	RHU001CL	ROO1A800	RV1120QU		195	3.528E-10	0.0002	LF2A	ROO1A800	RBI120QU		
133	6.962E-10	0.0004	SCRM	DCR0010F	DCR0020F	BOO02200	HU001BSZ	196	3.482E-10	0.0002	SHDS	VOO000UW	9CR0010F	9CR0020F	
134	6.924E-10	0.0004	TSEC	DCR0010F	DCR0020F	BOO02200	WFL095YE	197	3.482E-10	0.0002	SHDS	VOO000UW	9CR0010E	9CR0020F	
135	6.924E-10	0.0004	TSEC	DCR0010F	DCR0020F	BOO02200	HFL717SE	198	3.394E-10	0.0002	LONP	9CR0010E	9CR0020F		
136	6.912E-10	0.0004	LF2F	RREC00QU				199	3.394E-10	0.0002	LONP	9CR0010F	9CR0020E		
137	6.700E-10	0.0004	LONP	RHU001CL	RRE120QU			200	3.326E-10	0.0002	SPTF	VHEMO2WF	VHESPRMW	VHEDM2WF	
138	6.317E-10	0.0004	TSDC	DCR0010F	BOO02200	HFN705SR	HFN706SR	201	3.318E-10	0.0002	SHDL	DCR0010F	DCR0020E	BOO02200	HU001BSZ
139	6.317E-10	0.0004	TSDC	DCR0010F	BOO02200	HFN707SR	HFN708SR	202	3.318E-10	0.0002	SHDL	DCR0010E	DCR0020F	BOO02200	HU001BSZ
140	6.268E-10	0.0004	TSEC	DCR0010F	DCR0020F	BOO02200	HU001BSZ	203	3.297E-10	0.0002	OCPF	DCR0010F	DCR0020F	BOO02200	HFL717SE
141	6.000E-10	0.0004	OCSL	RHU3020L	RHU002CL	RCZ229QU		204	3.297E-10	0.0002	OCPF	DCR0010F	DCR0020F	BOO02200	WFL095YE
142	5.760E-10	0.0004	LF1B	RREC00QU	BOO9920W			205	3.000E-10	0.0002	TAIR	9HU0000L	9ARX82QP		
143	5.640E-10	0.0004	LF2B	RHU001CL	RCZ120QU			206	3.000E-10	0.0002	TAIR	9HU0000L	9ARX82QP		
144	5.495E-10	0.0003	LF1A	DCR0010F	DCR0020F	BOO02200	HFL717SE	207	3.000E-10	0.0002	RIFS	RHU3020L	RHU002CL	RAM221QU	HU001BSZ
145	5.495E-10	0.0003	LF1A	DCR0010F	DCR0020F	BOO02200	WFL095YE	208	2.984E-10	0.0002	OCPF	DCR0010F	DCR0020F	BOO02200	
146	5.475E-10	0.0003	TSHC	DCR0010E	DCR0020F			209	2.944E-10	0.0002	TSHC	DHU0000L	DARX82QW		
147	5.475E-10	0.0003	TSHC	DCR0010F	DCR0020E			210	2.944E-10	0.0002	TSHC	DIU0000L	DM4B1QW		
148	5.466E-10	0.0003	TSDC	ROO99A00	DCR0010F	BOO02200	EGE1EFGW	211	2.944E-10	0.0002	TSHC	DHU0000L	DM4A1QW		
+ ESHR1BYO								212	2.832E-10	0.0002	TSDC	DCR0010F	BOO02200	YFN0560R	YFN0580R
149	5.400E-10	0.0003	SFLE	FPP991RF	FPP112RF	RBS004QP	FPP002RF	213	2.822E-10	0.0002	TAIR	9CR0010E	9CR0020E		
150	5.400E-10	0.0003	SFLE	FPP991RF	FPP112RF	FPP902RF	FPP002RF	214	2.748E-10	0.0002	LF1C	DCR0010F	DCR0020F	BOO02200	HFL717SE
151	5.280E-10	0.0003	LF2C	DCR000QU				215	2.748E-10	0.0002	LF1C	DCR0010F	DCR0020F	BOO02200	HFL717SE
152	4.973E-10	0.0003	LF1A	DCR0010F		BOO02200	HU001BSZ	216	2.748E-10	0.0002	TAIR	DCR0010F	DCR0020F	BOO02200	WFL095YE
153	4.888E-10	0.0003	TSDC	VOO000UW		BOO02200	WFL095YE	217	2.748E-10	0.0002	TAIR	DCR0010F	DCR0020F	BOO02200	WFL095YE
154	4.888E-10	0.0003	TSDC	VOO000UW		BOO02200	WFL095YE	218	2.587E-10	0.0002	TSDC	DCR0010F	BOO02200	TXV770SP	
155	4.800E-10	0.0003	OCSL	RHU3020L	RHU002CL			219	2.567E-10	0.0002	SCRM	DCR0010F	DCR0020E	BOO02200	HFL717SE
156	4.700E-10	0.0003	LF2B	CSR301RU				220	2.567E-10	0.0002	SCRM	DCR0010F	DCR0020E	BOO02200	WFL095YE
157	4.700E-10	0.0003	LF2B	CSR10GRU				221	2.567E-10	0.0002	SCRM	DCR0010E	DCR0020E	BOO02200	WFL095YE
158	4.500E-10	0.0003	MF1F	FPP112RF		FPP004RF		222	2.567E-10	0.0002	SCRM	DCR0010F	DCR0020E	BOO02200	WFL095YE
159	4.500E-10	0.0003	OELF	FPP112RF		GND002RW		223	2.520E-10	0.0002	LF2A	CCL00PRU	RHU002CL		
160	4.440E-10	0.0003	TBOP	9HU0000L	9ARX82QP			224	2.495E-10	0.0002	RISB	RHU3020L	RHU002CL	RDC2B1QW	RND2C0RW
161	4.440E-10	0.0003	TBOP	9HU0000L	9ARX82QP			225	2.495E-10	0.0002	RISB	RHU3020L	RHU002CL	RDC2B1QW	RND2C0RW
162	4.423E-10	0.0003	TSDC	VOO000UW		BOO02200	HU001BSZ	226	2.495E-10	0.0002	RISB	RHU3020L	RHU002CL	RDC2B1QW	RND2C0RW
163	4.320E-10	0.0003	OCSL	RHU3020L	RHU002CL	RDC2B1QW	RCA2C0RW	227	2.495E-10	0.0002	RISB	RHU3020L	RHU002CL	RDC2B1QW	RND2C0RW
164	4.320E-10	0.0003	OCSL	RHU3020L	RHU002CL	RND2B0RW	RCA2C0RW	228	2.495E-10	0.0002	RISB	RHU3020L	RHU002CL	RDC2B1QW	RND2C0RW
165	4.320E-10	0.0003	OCSL	RHU3020L	RHU002CL	RDC2A1QW	RCA2C0RW	229	2.495E-10	0.0002	RISB	RHU3020L	RHU002CL	RDC2A1QW	RND2C0RW
166	4.320E-10	0.0003	OCSL	RHU3020L	RHU002CL	RND2A0RW	RCA2B0RW	230	2.495E-10	0.0002	RISB	RHU3020L	RHU002CL	RDC2A1QW	RND2C0RW
167	4.320E-10	0.0003	OCSL	RHU3020L	RHU002CL	RDC2A1QW	RCA2B0RW	231	2.495E-10	0.0002	RISB	RHU3020L	RHU002CL	RDC2A1QW	RND2C0RW
168	4.320E-10	0.0003	OCSL	RHU3020L	RHU002CL	RND2A0RW	RCA2B0RW	232	2.495E-10	0.0002	RISB	RHU3020L	RHU002CL	RDC2A1QW	RND2C0RW
169	4.320E-10	0.0003	OCSL	RHU3020L	RHU002CL	RCA2A0RW	RCA2C0RW	233	2.495E-10	0.0002	RISB	RHU3020L	RHU002CL	RDC2A1QW	RND2C0RW
170	4.320E-10	0.0003	OCSL	RHU3020L	RHU002CL	RCA2A0RW	RCA2C0RW	234	2.495E-10	0.0002	RISB	RHU3020L	RHU002CL	RDC2A1QW	RND2C0RW
171	4.320E-10	0.0003	OCSL	RHU3020L	RHU002CL	RCA2A0RW	RCA2C0RW	235	2.495E-10	0.0002	RISB	RHU3020L	RHU002CL	RDC2A1QW	RND2C0RW
172	4.320E-10	0.0003	OCSL	RHU3020L	RHU002CL	RCA2A0RW	RCA2C0RW	236	2.486E-10	0.0002	TAIR	DCR0010F	DCR0020F	BOO02200	HU001BSZ
173	4.320E-10	0.0003	OCSL	RHU3020L	RHU002CL	RCA2A0RW	RCA2C0RW	237	2.486E-10	0.0002	TAIR	DCR0010F	DCR0020F	BOO02200	HU001BSZ
174	4.320E-10	0.0003	OCSL	RHU3020L	RHU002CL	RCA2A0RW	RCA2C0RW	238	2.400E-10	0.0002	RIFS	RHU002CL	RC2220QU		
175	4.224E-10	0.0003	LF2C	RREC00QU				239	2.400E-10	0.0002	OCSL	RC2C00QU	NPMOVCHN		
176	4.177E-10	0.0003	TBOP	9CR0010E		9AM4B1QW		240	2.323E-10	0.0001	SCRM	DCR0010F	DCR0020E	BOO02200	HU001BSZ
177	4.145E-10	0.0003	SHDS	VOO000UW		9AM4A1QW		241	2.323E-10	0.0001	SCRM	DCR0010E	DCR0020F	BOO02200	HFL717SE
178	4.145E-10	0.0003	SHDS	VOO000UW		9ARX82QW		242	2.310E-10	0.0001	TSEC	DCR0010E	DCR0020F	BOO02200	WFL095YE
179	4.145E-10	0.0003	SHDS	VOO000UW		BOO02200	WFL095YE	243	2.310E-10	0.0001	TSEC	DCR0010E	DCR0020F	BOO02200	HFL717SE
180	4.066E-10	0.0003	TBOP	DCR0010F	DCR0020F			244	2.310E-10	0.0001	TSEC	DCR0010F	DCR0020E	BOO02200	WFL095YE
181	4.066E-10	0.0003	TBOP	DCR0010F	DCR0020F			245	2.310E-10	0.0001	TSEC	DCR0010F	DCR0020E	BOO02200	WFL095YE
182	4.040E-10	0.0003	LONP	9HU0000L	9AM4A1QW			246	2.250E-10	0.0001	LONP	RHU002CL	CCD301RU		
183	4.040E-10	0.0003	LONP	9HU0000L	9AM4B1QW			247	2.108E-10	0.0001	TSDC	DCR0010E	BOO02200	HFN705SR	HFN706SR
184	4.040E-10	0.0003	LONP	9HU0000L	9AM4B1QW			248	2.108E-10	0.0001	TSDC	DCR0010E	BOO02200	HFN707SR	HFN708SR
185	3.960E-10	0.0003	EVTR	VHERIXWF	VHESPRMW	VHEDM2WF		249	2.090E-10	0.0001	TSEC	DCR0010E	BOO02200	HU001BSZ	
186	3.750E-10	0.0002	OCSL	RHU3020L	RHU002CL	RHU201QL		250	2.090E-10	0.0001	TSEC	DCR0010F	DCR0020E	BOO02200	HU001BSZ
187	3.680E-10	0.0002	TBOP	DCR0010F	DCR0020F	BOO02200	HU001BSZ	251	2.047E-10	0.0001	LF2A	RHU002CL	CCD10GRU		

252	1.971E-10	0.0001	SHDL	DHU0000L	DAM4ALAW	BOO02200	WFL095YE	1.242E-10	0.0001	TSEC	DHU0000L	DAM4BLQW	BOO02200	HFL717SE
253	1.971E-10	0.0001	SHDL	DHU0000L	DAM4BLQW	BOO02200	HFL717SE	1.231E-10	0.0001	SHDS	VOOOOUW	9HU0000L	9RE53ACP	BOO02200
254	1.971E-10	0.0001	SHDL	DHU0000L	DAM4BLQW	BOO02200	WFL095YE	1.231E-10	0.0001	SHDS	VOOOOUW	9HU0000L	9ARX82QP	HUI01BS2
255	1.971E-10	0.0001	SHDL	DHU0000L	DAM4BLQW	BOO02200	HFL717SE	1.228E-10	0.0001	TBOP	DCR0010F	DCR0020F	BOO02200	HUI01BS2
256	1.971E-10	0.0001	SHDL	DHU0000L	DAM4ALAW	BOO02200	WFL717SE	1.228E-10	0.0001	TBOP	DCR0010E	DCR0020F	BOO02200	WFL095YE
257	1.971E-10	0.0001	SHDL	DHU0000L	DAM4BLQW	BOO02200	WFL095YE	1.223E-10	0.0001	SHDL	DCR0010E	DCR0020E	BOO02200	HFL717SE
258	1.920E-10	0.0001	OCSL	RRECO00Q	NPHOV0CW	BOO02200		1.223E-10	0.0001	SHDL	DCR0010E	DCR0020E	BOO02200	
259	1.920E-10	0.0001	RIFS	RRECO00Q	RRE2200Q	BOO02200		1.221E-10	0.0001	SCRM	VOOOOUW	9CR0010F	9CR0020F	
260	1.855E-10	0.0001	TSDC	DCR0010F	BOO02200	BOO02200	KAV238SP	1.200E-10	0.0001	LONP	9HU0000L	9RE53ACP		
261	1.833E-10	0.0001	LF1A	DCR0010E	DCR0020F	BOO02200	WFL095YE	1.200E-10	0.0001	LONP	9HU0000L	9ARX82QP	RCA220RU	
262	1.833E-10	0.0001	LF1A	DCR0010F	DCR0020E	BOO02200	HFL717SE	1.200E-10	0.0001	RIFS	RHU302QL	RHU002CL		
263	1.833E-10	0.0001	LF1A	DCR0010E	DCR0020F	BOO02200	HFL717SE	1.183E-10	0.0001	TSDC	VOOOOUW	9CR0010F		
264	1.833E-10	0.0001	LF1A	DCR0010F	DCR0020E	BOO02200	WFL095YE	1.158E-10	0.0001	SHDS	VOOOOUW	9CR0010F	9CR0020E	
265	1.827E-10	0.0001	TSHC	DCR0010E	DCR0020E	BOO02200	WFL095YE	1.152E-10	0.0001	RISA	RRECO00Q	BOO992UW		
266	1.824E-10	0.0001	TSDC	ROO99A00	DCR0010E	BOO02200	EGE1EFGW	1.129E-10	0.0001	LONP	9CR0010E	9CR0020E		
267	1.784E-10	0.0001	SHDL	DHU0000L	DAM4ALAW	BOO02200	HUI01BS2	1.124E-10	0.0001	TSEC	DHU0000L	DARX82QW	BOO02200	HUI01BS2
268	1.784E-10	0.0001	SHDL	DHU0000L	DAM4BLQW	BOO02200	HUI01BS2	1.124E-10	0.0001	TSEC	DHU0000L	DAM4BLQW	BOO02200	HUI01BS2
269	1.784E-10	0.0001	SHDL	DHU0000L	DARX82QW	BOO02200	HUI01BS2	1.124E-10	0.0001	TSEC	DHU0000L	DAM4ALAW	BOO02200	TFLJTNFU
270	1.728E-10	0.0001	OCSL	RTT300AU	RHU002CL	RND221QU		1.109E-10	0.0001	SHDL	DCR0010F	DCR0020F	BOO02200	HUI01BS2
271	1.728E-10	0.0001	OCSL	RTT300AU	RHU002CL	RND221QU		1.107E-10	0.0001	SHDL	DCR0010E	DCR0020E	BOO02200	WFL095YE
272	1.692E-10	0.0001	TSDC	DCR0010F	BOO02200	WPH001YW	WPH002YX	1.100E-10	0.0001	OCPP	DCR0010F	DCR0020E	BOO02200	HFL717SE
273	1.659E-10	0.0001	LF1A	DCR0010E	DCR0020F	BOO02200	HUI01BS2	1.100E-10	0.0001	OCPP	DCR0010F	DCR0020E	BOO02200	WFL095YE
274	1.659E-10	0.0001	LF1A	DCR0010F	DCR0020E	BOO02200	HUI01BS2	1.100E-10	0.0001	OCPP	DCR0010F	DCR0020E	BOO02200	HFL717SE
275	1.649E-10	0.0001	LF1B	DCR0010F	DCR0020F	BOO02200	WFL095YE	1.099E-10	0.0001	LONP	DCR0010F	DCR0020F	BOO02200	WFL095YE
276	1.649E-10	0.0001	LF1B	DCR0010F	DCR0020F	BOO02200	HFL717SE	1.099E-10	0.0001	LOCP	DCR0010F	DCR0020F	BOO02200	HFL717SE
277	1.638E-10	0.0001	TSDC	DCR0010F	BOO02200	EGE1EFGW	ECB238YU	1.099E-10	0.0001	OCSL	DCR0010F	DCR0020F	BOO02200	HFL717SE
278	1.638E-10	0.0001	TSDC	ROO99A00	DCR0010F	BOO02200	EGE1EFGW	1.099E-10	0.0001	LOCP	DCR0010F	DCR0020F	BOO02200	WFL095YE
279	1.631E-10	0.0001	TSDC	VOOOOUW	DCR0010E	BOO02200	HFL717SE	1.099E-10	0.0001	OCSL	DCR0010F	DCR0020F	BOO02200	WFL095YE
280	1.631E-10	0.0001	TSDC	VOOOOUW	DCR0010E	BOO02200	WFL095YE	1.099E-10	0.0001	LONP	DCR0010F	DCR0020F	BOO02200	
281	1.579E-10	0.0001	LF2B	RHU001CL	RCO1AB00	RV1120QU		1.037E-10	0.0001	LF1D	RRECO00Q	BOO992UW		
282	1.492E-10	0.0001	LF1B	DCR0010F	DCR0020F	BOO02200	HUI01BS2	1.000E-10	0.0001	LF1A	CSR10GRU	BOO992UW		
283	1.476E-10	0.0001	TSDC	VOOOOUW	DCR0010E	BOO02200	HUI01BS2	1.000E-10	0.0001	LF1A	CSR30IRU	BOO992UW		
284	1.441E-10	0.0001	TSDC	DCR0010F	BOO02200	WPH001YW	WPH002YX	1.000E-10	0.0001	RIFS	CSR30IRU			
285	1.440E-10	0.0001	OCSL	RHU302QL	RHU002CL	RAM2B1QW	RAM2B1QW	1.000E-10	0.0001	RIFS	CSR30IRU			
286	1.440E-10	0.0001	OCSL	RHU302QL	RHU002CL	RAM2B1QW	RAM2B1QW	9.955E-11	0.0001	OCPP	DCR0010E	DCR0020F	BOO02200	HUI01BS2
287	1.440E-10	0.0001	OCSL	RHU302QL	RHU002CL	RAM2A1QW	RCA2C0RW	9.955E-11	0.0001	OCPP	DCR0010F	DCR0020F	BOO02200	HUI01BS2
288	1.440E-10	0.0001	OCSL	RHU302QL	RHU002CL	RAM2A1QW	RCA2B0RW	9.946E-11	0.0001	OCPP	DCR0010F	DCR0020F	BOO02200	HUI01BS2
289	1.440E-10	0.0001	RISA	RC2C00QU	RHU002CL	BOO992UW	RAM2B1QW	9.946E-11	0.0001	LONP	DCR0010F	DCR0020F	BOO02200	HUI01BS2
290	1.440E-10	0.0001	OCSL	RHU302QL	RHU002CL	RAM2A1QW	RAM2B1QW	9.946E-11	0.0001	LOCP	DCR0010F	DCR0020F	BOO02200	WFL095YE
291	1.440E-10	0.0001	OCSL	RHU302QL	RHU002CL	RAM2A1QW	RAM2C1QW	9.857E-11	0.0001	LF1A	DHU0000L	DAM4BLQW	BOO02200	WFL095YE
292	1.407E-10	0.0001	LF2A	RHU001CL	RRE120QU		RAM2C1QW	9.857E-11	0.0001	LF1A	DHU0000L	DAM4ALAW	BOO02200	HFL717SE
293	1.380E-10	0.0001	SCRM	DHU0000L	DAM4BLQW	BOO02200	WFL095YE	9.857E-11	0.0001	LF1A	DHU0000L	DAM4ALAW	BOO02200	WFL095YE
294	1.380E-10	0.0001	SCRM	DHU0000L	DAM4ALAW	BOO02200	WFL095YE	9.857E-11	0.0001	LF1A	DHU0000L	DAM4BLQW	BOO02200	HFL717SE
295	1.380E-10	0.0001	SCRM	DHU0000L	DAM4ALAW	BOO02200	HFL717SE	9.857E-11	0.0001	LF1A	DHU0000L	DAM4BLQW	BOO02200	WFL095YE
296	1.380E-10	0.0001	SCRM	DHU0000L	DARX82QW	BOO02200	HFL717SE	9.857E-11	0.0001	TSDC	DCR0010F	BOO02200	TXV752SQ	
297	1.380E-10	0.0001	SCRM	DHU0000L	DARX82QW	BOO02200	HFL717SE	9.857E-11	0.0001	TSDC	DCR0010F	BOO02200	TXV752SQ	
298	1.380E-10	0.0001	SCRM	DHU0000L	DAM4BLQW	BOO02200	HFL717SE	9.857E-11	0.0001	TSDC	DCR0010F	BOO02200	TXV752SQ	
299	1.357E-10	0.0001	TBOP	DCR0010E	DCR0020F	BOO02200	WFL095YE	9.623E-11	0.0001	TSDC	DCR0010F	BOO02200	TXV752SQ	
300	1.357E-10	0.0001	TBOP	DCR0010F	DCR0020E	BOO02200	WFL095YE	9.623E-11	0.0001	TSDC	DCR0010F	BOO02200	TXV770SQ	
301	1.357E-10	0.0001	TBOP	DCR0010F	DCR0020F	BOO02200	HFL717SE	9.623E-11	0.0001	TSDC	DCR0010F	BOO02200	TXV770SQ	
302	1.357E-10	0.0001	TBOP	DCR0010F	DCR0020E	BOO02200	WFL095YE	9.623E-11	0.0001	TSDC	DCR0010F	BOO02200	TXV753SQ	
303	1.296E-10	0.0001	LF1D	RC2C00QU	BOO992UW	BOO02200	WFL095YE	9.452E-11	0.0001	TSDC	DCR0010E	BOO02200	YFN0560R	
304	1.260E-10	0.0001	SFPB	FPP991RF	FPP003RF	FPP112RF	RBS004QP	9.240E-11	0.0001	RISB	CCL00PRU	RHU002CL		
305	1.249E-10	0.0001	SCRM	DHU0000L	DARX82QW	BOO02200	HUI01BS2	9.167E-11	0.0001	RISB	RHU302QL	RHU002CL	RBI221QU	
306	1.249E-10	0.0001	SCRM	DHU0000L	DAM4BLQW	BOO02200	HUI01BS2	9.167E-11	0.0001	LF1C	DCR0010E	DCR0020F	BOO02200	WFL095YE
307	1.249E-10	0.0001	SCRM	DHU0000L	DAM4ALAW	BOO02200	HUI01BS2	9.167E-11	0.0001	LF1C	DCR0010F	DCR0020E	BOO02200	HFL717SE
308	1.242E-10	0.0001	TSEC	DHU0000L	DAM4ALAW	BOO02200	WFL095YE	9.167E-11	0.0001	LF1C	DCR0010E	DCR0020F	BOO02200	WFL095YE
309	1.242E-10	0.0001	TSEC	DHU0000L	DARX82QW	BOO02200	WFL095YE	9.167E-11	0.0001	TATR	DCR0010F	DCR0020F	BOO02200	WFL095YE
310	1.242E-10	0.0001	TSEC	DHU0000L	DARX82QW	BOO02200	HFL717SE	9.167E-11	0.0001	TATR	DCR0010F	DCR0020E	BOO02200	WFL095YE
311	1.242E-10	0.0001	TSEC	DHU0000L	DAM4BLQW	BOO02200	WFL095YE	9.167E-11	0.0001	TATR	DCR0010E	DCR0020F	BOO02200	HFL717SE
312	1.242E-10	0.0001	TSEC	DHU0000L	DAM4ALAW	BOO02200	HFL717SE	8.920E-11	0.0001	LF1A	DHU0000L	DAM4BLQW	BOO02200	WFL095YE

377	8.920E-11	0.0001	LF1A	DHU0000L	DARX82QW	BOO02200	HUU01BS2	441	5.914E-11	0.0000	OCPP	DHU0000L	DAM4ALAW	BOO02200	HFL717SE
378	8.920E-11	0.0001	LF1A	DHU0000L	DAM4ALAW	BOO02200	HUU01BS2	442	5.914E-11	0.0000	OCPP	DHU0000L	DAM4BLQW	BOO02200	WFL095YE
379	8.722E-11	0.0001	ANTC	VOOOOOW	9CR001OF	9CR0020F		443	5.914E-11	0.0000	OCPP	DHU0000L	DAM4BLQW	BOO02200	HFL717SE
380	8.722E-11	0.0001	LF1A	VOOOOOW	9CR001OF	9CR0020F		444	5.858E-11	0.0000	TSDC	NSW000C2	9CR001OF	THU01BS2	TFLHEPSU
381	8.640E-11	0.0001	OCSL	RTC300RU	RHU002CL	RND220RU		445	5.769E-11	0.0000	TSDC	DCR001OF	BOO02200	RAM211QU	
382	8.640E-11	0.0001	OCSL	RHU002CL	RVI320QU	RND221QU		446	5.760E-11	0.0000	OCSL	RRT300AU	RHU002CL	RAM211QU	
383	8.640E-11	0.0001	OCSL	RTC300RU	RHU002CL	RDC221QU		447	5.760E-11	0.0000	OCSL	RHU302QL	RHU002CL	RCA2B0RW	RCA2C0RW
384	8.640E-11	0.0001	OCSL	DCR001OE	BOO02200	TXV770SP		448	5.760E-11	0.0000	OCSL	RHU302QL	RHU002CL	RCA2A0RW	RCA2B0RW
385	8.630E-11	0.0001	TSDC	DCR001OE	BOO02200			449	5.760E-11	0.0000	TSDC	DCR001OE	BOO02200	WPM002YX	
386	8.563E-11	0.0001	SCRM	DCR001OE	DCR0020E			450	5.646E-11	0.0000	LF2B	CCL00PRU	RHU002CL	WPM002YX	
387	8.563E-11	0.0001	SHDL	DCR001OF	DCR0020E			451	5.640E-11	0.0000	LF1A	DCR001OF	DCR0020F	BOO02200	TFLJINFU
388	8.504E-11	0.0001	RISB	RHU302QL	RHU002CL	RAM2A1QW		452	5.545E-11	0.0000	LF1A	DCR001OE	DCR0020E	BOO02200	HUU01BS2
389	8.316E-11	0.0001	RISB	RHU302QL	RHU002CL	RAM2A1QW		453	5.535E-11	0.0000	LF1B	DCR001OF	DCR0020E	BOO02200	HFL717SE
390	8.316E-11	0.0001	RISB	RHU302QL	RHU002CL	RAM2A1QW		454	5.500E-11	0.0000	LF1B	DCR001OF	DCR0020E	BOO02200	WFL095YE
391	8.316E-11	0.0001	RISB	RHU302QL	RHU002CL	RAM2A1QW		455	5.500E-11	0.0000	LF1B	DCR001OF	DCR0020F	BOO02200	WFL095YE
392	8.316E-11	0.0001	RISB	RHU302QL	RHU002CL	RAM2A1QW		456	5.500E-11	0.0000	LF1B	DCR001OE	DCR0020F	BOO02200	WFL095YE
393	8.316E-11	0.0001	RISB	RHU302QL	RHU002CL	RAM2A1QW		457	5.500E-11	0.0000	LF1B	DCR001OE	DCR0020F	BOO02200	HFL717SE
394	8.316E-11	0.0001	RISB	RHU302QL	RHU002CL	RAM2A1QW		458	5.466E-11	0.0000	TSDC	DCR001OE	BOO02200	EGE1EFGW	EGE1EFGW
395	8.316E-11	0.0001	RISB	RHU302QL	RHU002CL	RAM2A1QW		459	5.466E-11	0.0000	TSDC	DCR001OE	BOO02200	EGE1EFGW	EGE1EFGW
396	8.316E-11	0.0001	RISB	RHU302QL	RHU002CL	RDC2A1QW		460	5.466E-11	0.0000	TSDC	JTRT560U	EGE4EFHW	DCR001OF	BOO02200
397	8.316E-11	0.0001	RISB	RHU302QL	RHU002CL	RND2A0RW		461	5.466E-11	0.0000	TSDC	ROO99A00	DCR001OF	BOO02200	EGE1EFGW
398	8.316E-11	0.0001	RISB	RHU302QL	RHU002CL	RDC2A1QW		462	5.466E-11	0.0000	TSDC	ROO99A00	DCR001OF	BOO02200	EGE1EFGW
399	8.316E-11	0.0001	RISB	RHU302QL	RHU002CL	RND2B0RW		463	5.466E-11	0.0000	TSDC	ROO99A00	DCR001OF	BOO02200	EGE1EFGW
400	8.316E-11	0.0001	RISB	RHU302QL	RHU002CL	RND2B0RW		464	5.357E-11	0.0000	TSEC	DCR001OF	DCR0020F	BOO02200	KIN0030W
401	8.296E-11	0.0001	TAIR	DCR001OF	DCR0020E	BOO02200	HUU01BS2	465	5.352E-11	0.0000	OCPP	DHU0000L	DAM4ALAW	BOO02200	HUU01BS2
402	8.296E-11	0.0001	LF1C	DCR001OE	DCR0020F	BOO02200	HUU01BS2	466	5.352E-11	0.0000	OCPP	DHU0000L	DAM4BLQW	BOO02200	HUU01BS2
403	8.296E-11	0.0001	TAIR	DCR001OE	DCR0020F	BOO02200	HUU01BS2	467	5.352E-11	0.0000	OCPP	DHU0000L	DARX82QW	BOO02200	HUU01BS2
404	8.296E-11	0.0001	LF1C	DCR001OF	DCR0020E	BOO02200	HUU01BS2	468	5.233E-11	0.0000	OCPP	VOOOOOW	9CR001OF	9CR0020F	
405	7.896E-11	0.0001	LF2B	RHU001CL	ROO1A800	RBI120QU		469	5.000E-11	0.0000	LF1C	CSR301RU	BOO9920W		
406	7.763E-11	0.0000	SCRM	DCR001OF	DCR0020F	BOO02200	TFLJINFU	470	4.977E-11	0.0000	LF1B	CSR10GRU	BOO9920W		
407	7.750E-11	0.0000	SCRM	DCR001OE	DCR0020E	BOO02200	HUU01BS2	471	4.977E-11	0.0000	LF1B	CSR10GRU	BOO9920W		
408	7.707E-11	0.0000	TSEC	DCR001OE	DCR0020E	BOO02200	WFL095YE	472	4.932E-11	0.0000	TSDC	DCR001OE	DCR0020F	BOO02200	TFLJINFU
409	7.707E-11	0.0000	TSEC	DCR001OE	DCR0020E	BOO02200	WFL095YE	473	4.928E-11	0.0000	LF1C	VOOOOOW	DCR001OF	BOO02200	WFL095YE
410	7.508E-11	0.0000	RISB	RHU002CL	CCD10GRU			474	4.928E-11	0.0000	LF1C	DHU0000L	DAM4BLQW	BOO02200	WFL095YE
411	7.500E-11	0.0000	LONP	RHU002CL	CCD20HRU			475	4.928E-11	0.0000	TAIR	DHU0000L	DARX82QW	BOO02200	WFL095YE
412	7.294E-11	0.0000	TBOP	DHU0000L	DAM4BLQW	BOO02200	WFL095YE	476	4.928E-11	0.0000	TAIR	DHU0000L	DARX82QW	BOO02200	WFL095YE
413	7.294E-11	0.0000	TBOP	DHU0000L	DARX82QW	BOO02200	WFL095YE	477	4.928E-11	0.0000	TAIR	DHU0000L	DAM4ALAW	BOO02200	WFL095YE
414	7.294E-11	0.0000	TBOP	DHU0000L	DARX82QW	BOO02200	WFL095YE	478	4.928E-11	0.0000	TAIR	DHU0000L	DAM4ALAW	BOO02200	WFL095YE
415	7.294E-11	0.0000	TBOP	DHU0000L	DAM4ALAW	BOO02200	WFL095YE	479	4.928E-11	0.0000	TAIR	DHU0000L	DAM4ALAW	BOO02200	WFL095YE
416	7.294E-11	0.0000	TBOP	DHU0000L	DAM4ALAW	BOO02200	WFL095YE	480	4.928E-11	0.0000	LF1C	DHU0000L	DAM4ALAW	BOO02200	WFL095YE
417	7.294E-11	0.0000	TBOP	DHU0000L	DAM4BLQW	BOO02200	WFL095YE	481	4.928E-11	0.0000	LF1C	DHU0000L	DAM4BLQW	BOO02200	WFL095YE
418	6.986E-11	0.0000	TSEC	DCR001OF	DCR0020E	BOO02200	HUU01BS2	482	4.928E-11	0.0000	TAIR	DHU0000L	DARX82QW	BOO02200	WFL095YE
419	6.975E-11	0.0000	TSEC	DCR001OE	DCR0020E	BOO02200	HUU01BS2	483	4.928E-11	0.0000	TAIR	DHU0000L	DARX82QW	BOO02200	WFL095YE
420	6.601E-11	0.0000	TBOP	DHU0000L	DARX82QW	BOO02200	HUU01BS2	484	4.928E-11	0.0000	LF1C	DHU0000L	DAM4ALAW	BOO02200	WFL095YE
421	6.601E-11	0.0000	TBOP	DHU0000L	DAM4ALAW	BOO02200	HUU01BS2	485	4.836E-11	0.0000	SCRM	VOOOOOW	9HU0000L	9ARX82QW	
422	6.601E-11	0.0000	TBOP	DHU0000L	DAM4ALAW	BOO02200	HUU01BS2	486	4.836E-11	0.0000	SCRM	VOOOOOW	9HU0000L	9AM4BLQW	
423	6.374E-11	0.0000	TSDC	DCR001OF	BOO02200	HFN706SR	HT2745SW	487	4.836E-11	0.0000	SCRM	VOOOOOW	9HU0000L	9AM4BLQW	
424	6.374E-11	0.0000	TSDC	DCR001OF	BOO02200	HFN706SR	HT2745SW	488	4.836E-11	0.0000	SCRM	VOOOOOW	9HU0000L	9AM4BLQW	
425	6.374E-11	0.0000	TSDC	DCR001OF	BOO02200	HFN706SR	HT2745SW	489	4.741E-11	0.0000	TSDC	VOOOOOW	9HU0000L	9AM4BLQW	
426	6.374E-11	0.0000	TSDC	DCR001OF	BOO02200	HFN706SR	HT2745SW	490	4.741E-11	0.0000	TSDC	VOOOOOW	9HU0000L	9AM4BLQW	
427	6.374E-11	0.0000	TSDC	DCR001OF	BOO02200	HFN706SR	HT2745SW	491	4.741E-11	0.0000	TSDC	VOOOOOW	9HU0000L	9AM4BLQW	
428	6.374E-11	0.0000	TSDC	DCR001OF	BOO02200	HFN706SR	HT2745SW	492	4.741E-11	0.0000	TSDC	VOOOOOW	9HU0000L	9AM4BLQW	
429	6.374E-11	0.0000	TSDC	DCR001OF	BOO02200	HFN706SR	HT2745SW	493	4.741E-11	0.0000	TSDC	VOOOOOW	9HU0000L	9AM4BLQW	
430	6.374E-11	0.0000	TSDC	DCR001OF	BOO02200	HFN706SR	HT2745SW	494	4.741E-11	0.0000	TSDC	VOOOOOW	9HU0000L	9AM4BLQW	
431	6.374E-11	0.0000	TSDC	DCR001OF	BOO02200	HFN706SR	HT2745SW	495	4.741E-11	0.0000	TSDC	VOOOOOW	9HU0000L	9AM4BLQW	
432	6.374E-11	0.0000	TSDC	DCR001OF	BOO02200	HFN706SR	HT2745SW	496	4.741E-11	0.0000	TSDC	VOOOOOW	9HU0000L	9AM4BLQW	
433	6.374E-11	0.0000	TSDC	DCR001OF	BOO02200	HFN706SR	HT2745SW	497	4.741E-11	0.0000	TSDC	VOOOOOW	9HU0000L	9AM4BLQW	
434	6.374E-11	0.0000	TSDC	DCR001OF	BOO02200	HFN706SR	HT2745SW	498	4.741E-11	0.0000	TSDC	VOOOOOW	9HU0000L	9AM4BLQW	
435	6.374E-11	0.0000	TSDC	DCR001OF	BOO02200	HFN706SR	HT2745SW	499	4.741E-11	0.0000	TSDC	VOOOOOW	9HU0000L	9AM4BLQW	
436	6.117E-11	0.0000	LF1A	DCR001OE	DCR0020E	BOO02200	WFL095YE	500	4.741E-11	0.0000	TSDC	VOOOOOW	9HU0000L	9AM4BLQW	
437	5.953E-11	0.0000	LF1A	DCR001OE	DCR0020E	BOO02200	WFL095YE								
438	5.914E-11	0.0000	OCPP	DCR001OF	DCR0020F	BOO02200	HUU01BS2								
439	5.914E-11	0.0000	OCPP	DHU0000L	DARX82QW	BOO02200	WFL095YE								
440	5.914E-11	0.0000	OCPP	DHU0000L	DAM4ALAW	BOO02200	WFL095YE								

501	E-11	0.0000	LF1C	DIHU0000L	DARX82QW	BOO02200	HHU01BS2	564	3.326E-11	0.0000	RISB	RHU302QL	RHU002CL	RCA2A10W	RDC2C10W
502	4.361E-11	0.0000	LF1C	VOOOOOUW	9CR0010F	9CR0020F		565	3.326E-11	0.0000	RISB	RHU302QL	RHU002CL	RCA2A10W	RCA2C0RW
503	4.320E-11	0.0000	OC5L	RHU002CL	RBI320QU	RDC221QU		566	3.326E-11	0.0000	RISB	RHU302QL	RHU002CL	RND2B0RW	RCA2C0RW
504	4.320E-11	0.0000	OC5L	RHU002CL	RBI320QU	RND220RU		567	3.326E-11	0.0000	RISB	RHU302QL	RHU002CL	RDC2B10W	RCA2C0RW
505	4.320E-11	0.0000	LF2F	RHU001CL	RC2120QU			568	3.326E-11	0.0000	RISB	RHU302QL	RHU002CL	RCA2A0RW	RDC2C10W
506	4.252E-11	0.0000	LF1A	DCR0010F	DCR0020F	BOO02200	KIN0030W	569	3.326E-11	0.0000	RISB	RHU302QL	RHU002CL	RCA2A0RW	RND2B0RW
507	4.103E-11	0.0000	TR0P	DCR0010F	DCR0020F	BOO02200	TFLJINFU	570	3.326E-11	0.0000	RISB	RHU302QL	RHU002CL	RCA2A0RW	RND2C0RW
508	4.096E-11	0.0000	TR0P	DCR0010F	DCR0020F	BOO02200	HHU01BS2	571	3.326E-11	0.0000	RISB	RHU302QL	RHU002CL	RCA2A0RW	RND2C0RW
509	4.062E-11	0.0000	SCRM	VOOOOOUW	9CR0010F	9CR0020E		572	3.326E-11	0.0000	RISB	RHU302QL	RHU002CL	RCA2A0RW	RND2C0RW
510	4.062E-11	0.0000	SCRM	VOOOOOUW	9CR0010F	9CR0020E		573	3.326E-11	0.0000	RISB	RHU302QL	RHU002CL	RCA2A0RW	RND2C0RW
511	4.040E-11	0.0000	SHDL	9HU0000L	9HR751AW	9HU500C2		574	3.326E-11	0.0000	OCPP	DCR0010F	DCR0020E	BOO02200	HHU01BS2
512	4.040E-11	0.0000	SHDL	9HU0000L	9HR501AW	9HU500C2		575	3.318E-11	0.0000	LONP	DCR0010F	DCR0020E	BOO02200	HHU01BS2
513	4.040E-11	0.0000	SHDL	9HU0000L	9HR504AW	9HU500C2		576	3.318E-11	0.0000	LOCP	DCR0010F	DCR0020E	BOO02200	HHU01BS2
514	4.040E-11	0.0000	SHDL	9HU0000L	9HR502AW	9HU500C2		577	3.318E-11	0.0000	OC5L	DCR0010F	DCR0020E	BOO02200	HHU01BS2
515	3.960E-11	0.0000	SHDL	9HU0000L	DRE53ACP			578	3.318E-11	0.0000	LONP	DCR0010F	DCR0020F	BOO02200	HHU01BS2
516	3.960E-11	0.0000	TSHC	DIHU0000L	DARX82QP			579	3.318E-11	0.0000	OC5L	DCR0010F	DCR0020F	BOO02200	HHU01BS2
517	3.960E-11	0.0000	TSHC	DIHU0000L	DARX82QP			580	3.318E-11	0.0000	LOCP	DCR0010F	DCR0020F	BOO02200	HHU01BS2
518	3.937E-11	0.0000	TSDC	NOOOOOUW	9CR0010F			581	3.297E-11	0.0000	RISA	DCR0010F	DCR0020F	BOO02200	HHU01BS2
519	3.937E-11	0.0000	TSDC	NOOOOOUW	9CR0010F			582	3.297E-11	0.0000	RISA	DCR0010F	DCR0020F	BOO02200	HHU01BS2
520	3.926E-11	0.0000	TSDC	EDL410HU	DCR0010F	BOO02200		583	3.282E-11	0.0000	OC5P	DCR0010F	DCR0020F	BOO02200	HHU01BS2
521	ESMR1BYO	0.0000	TSDC	VOOOOOUW	ROO99A00			584	3.240E-11	0.0000	RIFS	RHU302QL	RHU002CL	RDC2A10W	RDC2C10W
522	3.782E-11	0.0000	SHDL	DCR0010E	DCR0020F	BOO02200	KIN0030W	585	3.240E-11	0.0000	RIFS	RHU302QL	RHU002CL	RND2B0RW	RDC2C10W
523	3.700E-11	0.0000	SHDL	DCR0010F	DCR0020E	BOO02200	TFLJINFU	586	3.240E-11	0.0000	RIFS	RHU302QL	RHU002CL	RND2B0RW	RND2C0RW
524	3.696E-11	0.0000	RISB	RC2C00QU	BOO9920W	BOO02200	TFLJINFU	587	3.240E-11	0.0000	RIFS	RHU302QL	RHU002CL	RDC2A10W	RND2C0RW
525	3.696E-11	0.0000	RISB	RHU302QL	RHU002CL	RRE2229QU		588	3.240E-11	0.0000	RIFS	RHU302QL	RHU002CL	RND2A0RW	RDC2B10W
526	3.670E-11	0.0000	OCPP	DCR0010E	DCR0020E	BOO02200	WFL095YE	589	3.240E-11	0.0000	RIFS	RHU302QL	RHU002CL	RND2A0RW	RND2C0RW
527	3.670E-11	0.0000	OCPP	DCR0010E	DCR0020E	BOO02200	HFL717SE	590	3.240E-11	0.0000	RIFS	RHU302QL	RHU002CL	RDC2A10W	RDC2C10W
528	3.667E-11	0.0000	OC5L	DCR0010F	DCR0020F	BOO02200	HFL717SE	591	3.240E-11	0.0000	RIFS	RHU302QL	RHU002CL	RDC2B10W	RDC2C10W
529	3.667E-11	0.0000	LOCP	DCR0010F	DCR0020F	BOO02200	WFL095YE	592	3.240E-11	0.0000	RIFS	RHU302QL	RHU002CL	RND2B0RW	RND2C0RW
530	3.667E-11	0.0000	OC5L	DCR0010F	DCR0020F	BOO02200	WFL095YE	593	3.240E-11	0.0000	RIFS	RHU302QL	RHU002CL	RDC2A10W	RDC2B10W
531	3.667E-11	0.0000	OC5L	DCR0010F	DCR0020F	BOO02200	WFL095YE	594	3.240E-11	0.0000	RIFS	RHU302QL	RHU002CL	RDC2A10W	RDC2B10W
532	3.667E-11	0.0000	LOCP	DCR0010F	DCR0020F	BOO02200	WFL095YE	595	3.240E-11	0.0000	RIFS	RHU302QL	RHU002CL	RDC2A10W	RDC2B10W
533	3.667E-11	0.0000	LOCP	DCR0010F	DCR0020E	BOO02200	HFL717SE	596	3.211E-11	0.0000	TSDC	DCR0010E	BOO02200	RDC2A10W	RND2B0RW
534	3.667E-11	0.0000	LONP	DCR0010F	DCR0020E	BOO02200	HFL717SE	597	3.211E-11	0.0000	TSDC	DCR0010E	BOO02200	RDC2A10W	RND2B0RW
535	3.667E-11	0.0000	LONP	DCR0010F	DCR0020E	BOO02200	WFL095YE	598	3.211E-11	0.0000	TSDC	DCR0010E	BOO02200	BOO02200	BOO02200
536	3.667E-11	0.0000	LONP	DCR0010E	DCR0020F	BOO02200	HFL717SE	599	3.211E-11	0.0000	TSDC	DCR0010E	BOO02200	EBSR2AYC	EBSR2BYC
537	3.667E-11	0.0000	LONP	DCR0010E	DCR0020F	BOO02200	WFL095YE	600	3.211E-11	0.0000	TSDC	DCR0010E	BOO02200	EBSR2BYC	EBSR2BYC
538	3.667E-11	0.0000	LOCP	DCR0010F	DCR0020E	BOO02200	HFL717SE	601	3.211E-11	0.0000	TSDC	DCR0010E	BOO02200	TXV7525Q	TXV7535Q
539	3.667E-11	0.0000	OC5L	DCR0010E	DCR0020F	BOO02200	WFL095YE	602	3.200E-11	0.0000	LOCP	DCR0010E	BOO02200	TXV7535Q	TXV7535Q
540	3.627E-11	0.0000	OC5P	DCR0010F	DCR0020F	BOO02200	HFL717SE	603	3.200E-11	0.0000	LONP	DCR0010E	BOO02200	TXV7535Q	TXV7535Q
541	3.627E-11	0.0000	SHDL	DCR0010F	DCR0020F	BOO02200	HFL717SE	604	3.149E-11	0.0000	LF2B	DCR0010F	DCR0020F	BOO02200	BOO02200
542	3.600E-11	0.0000	SHDL	9HU0000L	9AT01UOP	9SV01BON	WFL095YE	605	3.146E-11	0.0000	TR0P	DCR0010F	DCR0020F	BOO02200	BOO02200
543	3.600E-11	0.0000	SHDL	9HU0000L	9AT01UOP	9SV01BON	WFL095YE	606	3.058E-11	0.0000	LF1C	DCR0010E	DCR0020E	BOO02200	BOO02200
544	3.600E-11	0.0000	SHDL	9HU0000L	9AT01UOP	9SV01BON	WFL095YE	607	3.058E-11	0.0000	TAIR	DCR0010E	DCR0020E	BOO02200	BOO02200
545	3.600E-11	0.0000	SHDL	9HU0000L	9AT01UOP	9SV01BON	WFL095YE	608	3.058E-11	0.0000	TAIR	DCR0010E	DCR0020E	BOO02200	BOO02200
546	3.600E-11	0.0000	LF2F	CSR10GRU	9AT01UOP	9AT02UOP	WFL095YE	609	3.058E-11	0.0000	LF1C	DCR0010E	DCR0020E	BOO02200	BOO02200
547	3.600E-11	0.0000	SHDL	9HU0000L	9AT01UOP	9AT02UOP	WFL095YE	610	3.000E-11	0.0000	LF1B	DCR0010E	DCR0020E	BOO02200	BOO02200
548	3.600E-11	0.0000	SHDL	9HU0000L	9AT01UOP	9AT02UOP	WFL095YE	611	3.000E-11	0.0000	LF1B	DCR0010E	DCR0020E	BOO02200	BOO02200
549	3.600E-11	0.0000	SHDL	9HU0000L	9AT01UOP	9AT02UOP	WFL095YE	612	2.984E-11	0.0000	RISA	DCR0010F	DCR0020F	BOO02200	BOO02200
550	3.600E-11	0.0000	LF2F	CSR30IRU	9AT01UOP	9AT02UOP	WFL095YE	613	2.967E-11	0.0000	LF1D	DCR0010F	DCR0020F	BOO02200	BOO02200
551	3.600E-11	0.0000	SHDL	9HU0000L	9AT01UOP	9AT02UOP	WFL095YE	614	2.967E-11	0.0000	LF1D	DCR0010F	DCR0020F	BOO02200	BOO02200
552	3.600E-11	0.0000	SHDL	9HU0000L	9AT01UOP	9AT02UOP	WFL095YE	615	2.957E-11	0.0000	LF1B	DCR0010F	DCR0020F	BOO02200	BOO02200
553	3.520E-11	0.0000	SPTR	VHERIXWF	VHERIXWF	VHERIXWF	VHERIXWF	616	2.957E-11	0.0000	LF1B	DCR0010F	DCR0020F	BOO02200	BOO02200
554	3.466E-11	0.0000	TSDC	DCR0010F	BOO02200	HDM747SQ		617	2.957E-11	0.0000	LF1B	DCR0010F	DCR0020F	BOO02200	BOO02200
555	3.454E-11	0.0000	ANTC	VOOOOOUW	9HU0000L	9AM4ALAW		618	2.957E-11	0.0000	LF1B	DCR0010F	DCR0020F	BOO02200	BOO02200
556	3.454E-11	0.0000	ANTC	VOOOOOUW	9HU0000L	9AM4ALAW		619	2.957E-11	0.0000	LF1B	DCR0010F	DCR0020F	BOO02200	BOO02200
557	3.454E-11	0.0000	LF1A	VOOOOOUW	9HU0000L	9AM4BLQW		620	2.957E-11	0.0000	LF1B	DCR0010F	DCR0020F	BOO02200	BOO02200
558	3.454E-11	0.0000	LF1A	VOOOOOUW	9HU0000L	9AM4BLQW		621	2.957E-11	0.0000	RISB	DCR0010F	DCR0020F	BOO02200	BOO02200
559	3.454E-11	0.0000	LF1A	VOOOOOUW	9HU0000L	9AM4ALAW		622	2.940E-11	0.0000	SPTR	VHEMO2NF	VHEMO2NF	9CR0010F	9CR0010F
560	3.454E-11	0.0000	ANTC	VOOOOOUW	9HU0000L	9AM4BLQW		623	2.902E-11	0.0000	ANTC	VOOOOOUW	VOOOOOUW	9CR0010F	9CR0010F
561	3.327E-11	0.0000	OCPP	DCR0010F	DCR0020F	BOO02200	TFLJINFU	624	2.902E-11	0.0000	LF1A	VOOOOOUW	VOOOOOUW	9CR0010F	9CR0010F
562	3.326E-11	0.0000	RISB	RHU302QL	RHU002CL	RDC2A10W	RCA2B0RW	625	2.902E-11	0.0000	LF1A	VOOOOOUW	VOOOOOUW	9CR0010F	9CR0010F
563	3.326E-11	0.0000	RISB	RHU302QL	RHU002CL	RND2A0RW	RCA2B0RW	626	2.902E-11	0.0000	ANTC	VOOOOOUW	VOOOOOUW	9CR0010F	9CR0010F
563	3.326E-11	0.0000	RISB	RHU302QL	RHU002CL	RND2A0RW	RCA2B0RW	627	2.887E-11	0.0000	RISB	RHU302QL	RHU002CL	RHU2010L	RHU2010L

628	2.880E-11	0.0000	OC5L	RHU002CL	RVI320QU	RAM221QU		690	2.127E-11	0.0000	TSDC	DCR001OE	BOO02200	HFN707SR	HWR708TB
629	2.880E-11	0.0000	OC5L	RTC300RU	RHU002CL	RAM221QU		691	2.127E-11	0.0000	TSDC	DCR001OE	BOO02200	HFN707SR	HWR708TB
630	2.858E-11	0.0000	TSDC	DCR001OF	BOO02200	HFN707SR	HFN708SS	692	2.127E-11	0.0000	TSDC	DCR001OE	BOO02200	HFN708SR	HRE99CSO
631	2.858E-11	0.0000	TSDC	DCR001OF	BOO02200	HFN707SR	HFN706SS	693	2.127E-11	0.0000	TSDC	DCR001OE	BOO02200	HRE051SO	HFN706SR
632	2.837E-11	0.0000	SHDL	DCR001OE	DCR0020F	BOO02200	KIN0030W	694	2.127E-11	0.0000	TSDC	DCR001OE	BOO02200	HFN706SR	HTZ745SW
633	2.837E-11	0.0000	SHDL	DCR001OF	DCR0020E	BOO02200	KIN0030W	695	2.127E-11	0.0000	TSDC	DCR001OE	BOO02200	HFN708SR	HTZ099SW
634	2.806E-11	0.0000	TSNC	VOO000UW	DCR001OF	DCR0020F		696	2.126E-11	0.0000	LF1C	DCR001OF	DCR0020F	BOO02200	KIN0030W
635	2.806E-11	0.0000	SFTR	VHERIXWF	VHELKWHF	VHEMO1WF	VHEP22NF	697	2.126E-11	0.0000	TAIR	DCR001OF	BOO02200	BOO02200	KIN0030W
636	2.772E-11	0.0000	LF1C	DCR001OF	DCR0020F	BOO02200	TFLJINFU	698	2.102E-11	0.0000	SHDL	ROO99A00	DCR0010F	DCR0020F	BOO02200
637	2.772E-11	0.0000	TAIR	DCR001OF	DCR0020F	BOO02200	TFLJINFU	+ ESWR18YO							
638	2.772E-11	0.0000	R1SB	RHU030QL	RHU002CL	RAM2A1QW	RAM2B1QW	699	2.089E-11	0.0000	LF2A	DCR0010F	DCR0020F	BOO02200	HU01BS2
639	2.772E-11	0.0000	R1SB	RHU030QL	RHU002CL	RAM2A1QW	RAM2C1QW	700	2.073E-11	0.0000	OCPP	VOO000UW	9IU0000L	9AM4BLQW	
640	2.772E-11	0.0000	R1SB	RHU030QL	RHU002CL	RAM2B1QW	RAM2C1QW	701	2.073E-11	0.0000	OCPP	VOO000UW	9IU0000L	9AM4ALAW	
641	2.768E-11	0.0000	TAIR	DCR001OE	DCR0020E	BOO02200	HU01BS2	702	2.073E-11	0.0000	OCPP	VOO000UW	9IX882QW	9ARX82QW	
642	2.768E-11	0.0000	LF1C	DCR001OF	DCR0020E	BOO02200	HU01BS2	703	2.054E-11	0.0000	TSDC	DCR001OE	BOO02200	KAV23ASQ	
643	2.685E-11	0.0000	LF1D	DCR001OF	DCR0020F	BOO02200	HU01BS2	704	2.054E-11	0.0000	TSDC	DCR001OE	BOO02200	KAV320SQ	
644	2.676E-11	0.0000	LF1B	DCR001OF	DCR0020F	BOO02200	HU01BS2	705	1.989E-11	0.0000	SHDL	DHU0000L	DAM4ALAW	BOO02200	TFLJINFU
645	2.676E-11	0.0000	LF1B	DHU0000L	DAM4BLQW	BOO02200	HU01BS2	706	1.989E-11	0.0000	SHDL	DHU0000L	DAM4BLQW	BOO02200	TFLJINFU
646	2.676E-11	0.0000	LF1B	DHU0000L	DAM4ALAW	BOO02200	HU01BS2	707	1.989E-11	0.0000	SHDL	DHU0000L	DAM4BLQW	BOO02200	TFLJINFU
647	2.652E-11	0.0000	SHDL	DHU0000L	DRE53ACP	BOO02200	HU0175SE	708	1.986E-11	0.0000	SCRM	DCR001OE	DCR0020F	KIN0030W	
648	2.652E-11	0.0000	SHDL	DHU0000L	DARX82QF	BOO02200	HFL717SE	709	1.986E-11	0.0000	SCRM	DCR001OF	DCR0020E	KIN0030W	
649	2.652E-11	0.0000	SHDL	DHU0000L	DRE53ACP	BOO02200	WFL095YE	710	1.971E-11	0.0000	OC5L	DHU0000L	DARX82QW	WFL095YE	
650	2.652E-11	0.0000	SHDL	DHU0000L	DARX82QF	BOO02200	WFL095YE	711	1.971E-11	0.0000	OC5L	DHU0000L	DARX82QW	WFL095YE	
651	2.640E-11	0.0000	LF2C	RHU001CL	RC2120QU			712	1.971E-11	0.0000	OC5L	DHU0000L	DAM4ALAW	WFL095YE	
652	2.617E-11	0.0000	LF1B	VOO000UW	9CR0010F	9CR0020F		713	1.971E-11	0.0000	LONP	DHU0000L	DARX82QW	WFL095YE	
653	2.600E-11	0.0000	LOCP	SCLO00RP	CCD10GRU			714	1.971E-11	0.0000	LONP	DHU0000L	DAM4ALAW	WFL095YE	
654	2.600E-11	0.0000	LONP	SCLO00RP	CCD10GRU			715	1.971E-11	0.0000	LOCP	DHU0000L	DAM4BLQW	WFL095YE	
655	2.590E-11	0.0000	SCRM	DCR001OF	DCR0020E	BOO02200	TFLJINFU	716	1.971E-11	0.0000	OC5L	DHU0000L	DARX82QW	WFL095YE	
656	2.590E-11	0.0000	SCRM	DCR001OE	DCR0020F	BOO02200	TFLJINFU	717	1.971E-11	0.0000	LOCP	DHU0000L	DARX82QW	WFL095YE	
657	2.573E-11	0.0000	TSDC	DCR001OF	BOO02200	HCM040SR	HCM201SR	718	1.971E-11	0.0000	LONP	DHU0000L	DAM4ALAW	WFL095YE	
658	2.573E-11	0.0000	TSDC	DCR001OF	BOO02200	HCM040SR	HCM202SR	719	1.971E-11	0.0000	LONP	DHU0000L	DAM4BLQW	WFL095YE	
659	2.551E-11	0.0000	OCPP	DCR001OF	DCR0020F	BOO02200	KIN0030W	720	1.971E-11	0.0000	OC5L	DHU0000L	DAM4BLQW	WFL095YE	
660	2.545E-11	0.0000	TSEC	9HU0000L	9IR751AW	9IU500CZ		721	1.971E-11	0.0000	LONP	DHU0000L	DAM4BLQW	WFL095YE	
661	2.545E-11	0.0000	TSEC	9HU0000L	9HR504AW	9IU500CZ		722	1.971E-11	0.0000	LONP	DHU0000L	DARX82QW	WFL095YE	
662	2.545E-11	0.0000	TSEC	9HU0000L	9HR502AW	9IU500CZ		723	1.971E-11	0.0000	LOCP	DHU0000L	DARX82QW	WFL095YE	
663	2.545E-11	0.0000	TSEC	9HU0000L	9HR505AW	9IU500CZ		724	1.971E-11	0.0000	LOCP	DHU0000L	DAM4BLQW	WFL095YE	
664	2.545E-11	0.0000	TSEC	9HU0000L	9HR501AW	9IU500CZ		725	1.971E-11	0.0000	OC5L	DHU0000L	DARX82QW	WFL095YE	
665	2.429E-11	0.0000	SHDL	DCR001OF	DCR0020F	BOO02200	HFN707SR	726	1.971E-11	0.0000	LOCP	DHU0000L	DAM4ALAW	WFL095YE	
+ HFN708SR								727	1.971E-11	0.0000	LOCP	DHU0000L	DAM4ALAW	WFL095YE	
666	2.429E-11	0.0000	SHDL	DCR001OF	DCR0020F	BOO02200	HFN705SR	728	1.949E-11	0.0000	TSDC	NSW000CZ	9CR001OE	TTHU01BS2	TFLHEPSU
+ HFN706SR								729	1.925E-11	0.0000	TSDC	DCR001OE	BOO02200	BOO02200	WFL095YE
667	2.400E-11	0.0000	SHDL	DHU0000L	DRE53ACP	BOO02200	HU01BS2	730	1.856E-11	0.0000	SCRM	DHU0000L	DRE53ACP	BOO02200	HFL717SE
668	2.400E-11	0.0000	SHDL	DHU0000L	DARX82QF	BOO02200	HU01BS2	731	1.856E-11	0.0000	SCRM	DHU0000L	DRE53ACP	BOO02200	HFL717SE
669	2.331E-11	0.0000	TSEC	DCR001OE	DCR0020F	BOO02200	TFLJINFU	732	1.856E-11	0.0000	SCRM	DHU0000L	DARX82QF	BOO02200	WFL095YE
670	2.331E-11	0.0000	TSEC	DCR001OF	DCR0020E	BOO02200	TFLJINFU	733	1.856E-11	0.0000	SCRM	DHU0000L	DARX82QF	BOO02200	WFL095YE
671	2.308E-11	0.0000	LF2A	DCR001OF	DCR0020F	BOO02200	HFL717SE	734	1.850E-11	0.0000	LF1A	DCR001OF	DCR0020E	TFLJINFU	
672	2.308E-11	0.0000	LF2A	DCR001OF	DCR0020F	BOO02200	WFL095YE	735	1.850E-11	0.0000	LF1A	DCR001OE	DCR0020E	TFLJINFU	
673	2.304E-11	0.0000	OC5L	RT300AU	RHU002CL	RCA220RU		736	1.835E-11	0.0000	LF1B	DCR001OE	DCR0020E	HFL717SE	
674	2.278E-11	0.0000	TSDC	DCR001OF	BOO02200	EB5001UW	YFN0580R	737	1.835E-11	0.0000	LF1B	DCR001OE	DCR0020E	WFL095YE	
675	2.278E-11	0.0000	TSDC	DCR001OF	BOO02200	YFN056GR	EB5002UW	738	1.824E-11	0.0000	TSDC	JTRT560U	DCR001OE	BOO02200	BOO02200
676	2.268E-11	0.0000	TSEC	9HU0000L	9AT01UOP	9AT02UOP		739	1.824E-11	0.0000	TSDC	ROO99A00	DCR001OE	BOO02200	EGE1EFGW
677	2.268E-11	0.0000	TSEC	9HU0000L	9AT01UOP	9AT01UOP		+ EWR12YC							
678	2.268E-11	0.0000	TSEC	9HU0000L	9AT02UOP	9SVO1BON		740	1.824E-11	0.0000	TSDC	ROO99A00	DCR001OE	BOO02200	EGE1EFGW
679	2.268E-11	0.0000	TSEC	9HU0000L	9AT02UOP	9SVO1AON		+ EWR12YB							
680	2.268E-11	0.0000	TSEC	9HU0000L	9AT02UOP	9SVO1AON		741	1.787E-11	0.0000	TSEC	DCR001OE	DCR0020F	KIN0030W	
681	2.268E-11	0.0000	TSEC	9HU0000L	9AT01UOP	9SVO1BON		742	1.787E-11	0.0000	TSEC	DCR001OF	DCR0020E	KIN0030W	
682	2.268E-11	0.0000	TSEC	9HU0000L	9AT01UOP	9SVO1AON		743	1.784E-11	0.0000	OC5L	DHU0000L	DARX82QW	HU01BS2	
683	2.268E-11	0.0000	TSEC	9HU0000L	9SVO1AON	9SVO1BON		744	1.784E-11	0.0000	OC5L	DHU0000L	DAM4ALAW	HU01BS2	
684	2.268E-11	0.0000	TSEC	9HU0000L	9AT01UOP	9AT02UOP		745	1.784E-11	0.0000	LOCP	DHU0000L	DARX82QW	HU01BS2	
685	2.200E-11	0.0000	LF2C	CSR301RU	9AT01UOP	9AT01UOP		746	1.784E-11	0.0000	LONP	DHU0000L	DAM4BLQW	HU01BS2	
686	2.200E-11	0.0000	LF2C	CSR301RU	9AT01UOP	9AT01UOP		747	1.784E-11	0.0000	LONP	DHU0000L	DAM4BLQW	HU01BS2	
687	2.127E-11	0.0000	TSDC	DCR001OE	BOO02200	HWR705TB	HFN706SR	748	1.784E-11	0.0000	OC5L	DHU0000L	DAM4BLQW	HU01BS2	
688	2.127E-11	0.0000	TSDC	DCR001OE	BOO02200	HFN705SR	HWR706TB	749	1.784E-11	0.0000	LONP	DHU0000L	DARX82QW	HU01BS2	
689	2.127E-11	0.0000	TSDC	DCR001OE	BOO02200	HFN706SR	HTZ096SW	750	1.784E-11	0.0000	LOCP	DHU0000L	DAM4BLQW	HU01BS2	

751	4E-11	0.0000	LOCPL	DHU0000L	DAM4ALAW	BOO02200	HHU01BSZ	809	1.332E-11	0.0000	TBOP	9HU0000L	9SV01AON	9.0N
752	1.44E-11	0.0000	OCSL	VOOOOOW	9CR001OF	BOO02200	9CR002OF	810	1.332E-11	0.0000	TBOP	9HU0000L	9AT01LOP	9AT02LOP
753	1.744E-11	0.0000	LOCPL	VOOOOOW	9CR001OF	9CR002OF	9CR002OF	811	1.332E-11	0.0000	TBOP	9HU0000L	9AT02UOP	9SV01AON
754	1.741E-11	0.0000	OCPP	VOOOOOW	9CR001OF	9CR002OF	9CR002OF	812	1.332E-11	0.0000	TBOP	9HU0000L	9AT01LOP	9SV01BON
755	1.741E-11	0.0000	OCPP	VOOOOOW	9CR001OF	9CR002OF	9CR002OF	813	1.332E-11	0.0000	TBOP	9HU0000L	9AT01LOP	9SV01BON
756	1.733E-11	0.0000	RISB	VOOOOOW	CCD301RU	9ARX82QW	9ARX82QW	814	1.332E-11	0.0000	TBOP	9HU0000L	9AT02LOP	9SV01AON
757	1.727E-11	0.0000	LF1C	VOOOOOW	9HU0000L	9AM4ALAW	9AM4ALAW	815	1.332E-11	0.0000	TBOP	9HU0000L	9AT01LOP	9AT02UOP
758	1.727E-11	0.0000	LF1C	VOOOOOW	9HU0000L	9AM4ALAW	9AM4ALAW	816	1.332E-11	0.0000	TBOP	9HU0000L	9AT01LOP	9AT02UOP
759	1.727E-11	0.0000	LF1C	VOOOOOW	9HU0000L	9AM4ALAW	9AM4ALAW	817	1.332E-11	0.0000	TBOP	9HU0000L	9AT01LOP	9AT02UOP
760	1.700E-11	0.0000	SCRM	DCR001OF	DCR002OF	BOO02200	HFN707SR	818	1.331E-11	0.0000	RISB	9HU0000L	9AT01LOP	9AT02LOP
761	1.700E-11	0.0000	SCRM	DCR001OF	DCR002OF	BOO02200	HFN705SR	819	1.331E-11	0.0000	RISB	9HU0000L	9AT01LOP	9AT02LOP
762	1.680E-11	0.0000	SCRM	DCR001OF	DARX82QP	BOO02200	HHU01BSZ	820	1.326E-11	0.0000	LF1A	9HU0000L	9AT01LOP	9AT02LOP
763	1.680E-11	0.0000	LF2E	DCR0000L	DARX82QP	BOO02200	HHU01BSZ	821	1.326E-11	0.0000	LF1A	9HU0000L	9AT01LOP	9AT02LOP
764	1.680E-11	0.0000	SCRM	DCR0000L	DARX82QP	BOO02200	HHU01BSZ	822	1.326E-11	0.0000	LF1A	9HU0000L	9AT01LOP	9AT02LOP
765	1.671E-11	0.0000	TSEC	DCR0000L	DARX82QP	BOO02200	HHU01BSZ	823	1.326E-11	0.0000	LF1A	9HU0000L	9AT01LOP	9AT02LOP
766	1.671E-11	0.0000	TSEC	DCR0000L	DARX82QP	BOO02200	HHU01BSZ	824	1.324E-11	0.0000	TSEC	9HU0000L	9AT01LOP	9AT02LOP
767	1.671E-11	0.0000	TSEC	DCR0000L	DARX82QP	BOO02200	HHU01BSZ	825	1.316E-11	0.0000	SCRM	9HU0000L	9AT01LOP	9AT02LOP
768	1.671E-11	0.0000	TSEC	DCR0000L	DARX82QP	BOO02200	HHU01BSZ	826	1.316E-11	0.0000	SCRM	9HU0000L	9AT01LOP	9AT02LOP
769	1.663E-11	0.0000	LF1B	DCR001OF	DCR002OF	BOO02200	HHU01BSZ	827	1.310E-11	0.0000	TSDC	9HU0000L	9AT01LOP	9AT02LOP
770	1.661E-11	0.0000	LF1B	DCR001OF	DCR002OF	BOO02200	HHU01BSZ	828	1.310E-11	0.0000	TSDC	9HU0000L	9AT01LOP	9AT02LOP
771	1.646E-11	0.0000	TSDC	VOOOOOW	DCR001OF	DCR002OF	HHU01BSZ	829	1.294E-11	0.0000	SPTA	9HU0000L	9AT01LOP	9AT02LOP
772	1.638E-11	0.0000	TSDC	JSG1002U	EGE4EFHW	DCR001OF	BOO02200	830	1.286E-11	0.0000	TSDC	9HU0000L	9AT01LOP	9AT02LOP
773	1.582E-11	0.0000	TSDC	JO001200	JGE0012W	JBS0010B	EGE4EFHW	831	1.286E-11	0.0000	TSDC	9HU0000L	9AT01LOP	9AT02LOP
774	1.575E-11	0.0000	LF2A	RHU002CL	CCD20WRU	BOO02200	HHU01BSZ	832	1.286E-11	0.0000	TSDC	9HU0000L	9AT01LOP	9AT02LOP
775	1.540E-11	0.0000	EVTR	VHESPRMW	VHESPRMW	BOO02200	HHU01BSZ	833	1.286E-11	0.0000	TSDC	9HU0000L	9AT01LOP	9AT02LOP
776	1.530E-11	0.0000	TSEC	DCR001OF	DCR002OF	BOO02200	HHU01BSZ	834	1.286E-11	0.0000	TSDC	9HU0000L	9AT01LOP	9AT02LOP
777	1.530E-11	0.0000	TSEC	DCR001OF	DCR002OF	BOO02200	HHU01BSZ	835	1.286E-11	0.0000	TSDC	9HU0000L	9AT01LOP	9AT02LOP
778	1.525E-11	0.0000	SHDL	DHU0000L	DAM4BLQW	BOO02200	KIN003UW	836	1.276E-11	0.0000	LF1B	9HU0000L	9AT01LOP	9AT02LOP
779	1.525E-11	0.0000	SHDL	DHU0000L	DAM4BLQW	BOO02200	KIN003UW	837	1.263E-11	0.0000	TSDC	9HU0000L	9AT01LOP	9AT02LOP
780	1.525E-11	0.0000	SHDL	DHU0000L	DAM4BLQW	BOO02200	KIN003UW	838	1.262E-11	0.0000	TSDC	9HU0000L	9AT01LOP	9AT02LOP
781	1.512E-11	0.0000	TSEC	DHU0000L	DARX82QP	BOO02200	HHU01BSZ	839	1.260E-11	0.0000	SFPB	9HU0000L	9AT01LOP	9AT02LOP
782	1.512E-11	0.0000	TSEC	DHU0000L	DARX82QP	BOO02200	HHU01BSZ	840	1.253E-11	0.0000	TSEC	9HU0000L	9AT01LOP	9AT02LOP
783	1.495E-11	0.0000	TBOP	9HU0000L	9HR505AH	9HU500CZ	9HU500CZ	841	1.253E-11	0.0000	TSEC	9HU0000L	9AT01LOP	9AT02LOP
784	1.495E-11	0.0000	TBOP	9HU0000L	9HR751AW	9HU500CZ	9HU500CZ	842	1.253E-11	0.0000	TSEC	9HU0000L	9AT01LOP	9AT02LOP
785	1.495E-11	0.0000	TBOP	9HU0000L	9HR502AW	9HU500CZ	9HU500CZ	843	1.251E-11	0.0000	TSDC	9HU0000L	9AT01LOP	9AT02LOP
786	1.495E-11	0.0000	TBOP	9HU0000L	9HR504AW	9HU500CZ	9HU500CZ	844	1.251E-11	0.0000	TSDC	9HU0000L	9AT01LOP	9AT02LOP
787	1.495E-11	0.0000	TBOP	9HU0000L	9HR501AW	9HU500CZ	9HU500CZ	845	1.251E-11	0.0000	TSDC	9HU0000L	9AT01LOP	9AT02LOP
788	1.471E-11	0.0000	SCRM	ROO99A00	DCR001OF	DCR002OF	BOO02200	846	1.236E-11	0.0000	SHDL	9HU0000L	9AT01LOP	9AT02LOP
789	1.451E-11	0.0000	LF1C	VOOOOOW	9CR001OF	9CR002OF	9CR002OF	847	1.234E-11	0.0000	OCSL	9HU0000L	9AT01LOP	9AT02LOP
790	1.451E-11	0.0000	LF1C	VOOOOOW	9CR001OF	9CR002OF	9CR002OF	848	1.223E-11	0.0000	OCSL	9HU0000L	9AT01LOP	9AT02LOP
791	1.440E-11	0.0000	OCSL	RHU002CL	RB1320QU	9RE53ACP	9RE53ACP	849	1.223E-11	0.0000	OCSL	9HU0000L	9AT01LOP	9AT02LOP
792	1.436E-11	0.0000	SCRM	VOOOOOW	9HU0000L	9ARX82QP	9ARX82QP	850	1.223E-11	0.0000	LOMP	9HU0000L	9AT01LOP	9AT02LOP
793	1.436E-11	0.0000	SCRM	VOOOOOW	9HU0000L	9ARX82QP	9ARX82QP	851	1.223E-11	0.0000	LOMP	9HU0000L	9AT01LOP	9AT02LOP
794	1.419E-11	0.0000	LF1A	DCR001OF	DCR002OF	BOO02200	BOO02200	852	1.223E-11	0.0000	LOCP	9HU0000L	9AT01LOP	9AT02LOP
795	1.419E-11	0.0000	LF1A	DCR001OF	DCR002OF	BOO02200	BOO02200	853	1.223E-11	0.0000	LOCP	9HU0000L	9AT01LOP	9AT02LOP
796	1.408E-11	0.0000	EVTR	VHESPRMW	VHESPRMW	BOO02200	VHESPRMW	854	1.214E-11	0.0000	LF1A	9HU0000L	9AT01LOP	9AT02LOP
797	1.392E-11	0.0000	SCRM	DHU0000L	DAM4ALAW	BOO02200	TFLJINFU	855	1.214E-11	0.0000	LF1A	9HU0000L	9AT01LOP	9AT02LOP
798	1.392E-11	0.0000	SCRM	DHU0000L	DAM4ALAW	BOO02200	TFLJINFU	856	1.210E-11	0.0000	OCSP	9HU0000L	9AT01LOP	9AT02LOP
799	1.392E-11	0.0000	SCRM	DHU0000L	DAM4ALAW	BOO02200	TFLJINFU	857	1.210E-11	0.0000	OCSP	9HU0000L	9AT01LOP	9AT02LOP
800	1.369E-11	0.0000	TBOP	DCR001OF	DCR002OF	BOO02200	TFLJINFU	858	1.210E-11	0.0000	OCSP	9HU0000L	9AT01LOP	9AT02LOP
801	1.369E-11	0.0000	TBOP	DCR001OF	DCR002OF	BOO02200	TFLJINFU	859	1.210E-11	0.0000	OCSP	9HU0000L	9AT01LOP	9AT02LOP
802	1.351E-11	0.0000	SCRM	VOOOOOW	9CR001OF	9CR002OF	9CR002OF	860	1.210E-11	0.0000	LF2F	9HU0000L	9AT01LOP	9AT02LOP
803	1.344E-11	0.0000	SHDL	9HU0000L	9TC40PRD	9HU82LCZ	9HU82LCZ	861	1.200E-11	0.0000	RIFS	9HU0000L	9AT01LOP	9AT02LOP
804	1.344E-11	0.0000	SHDL	9HU0000L	9TC40PRD	9HU82LCZ	9HU82LCZ							
805	1.344E-11	0.0000	SHDL	9HU0000L	9TC40PRD	9HU82LCZ	9HU82LCZ							
806	1.344E-11	0.0000	LF2E	9HU0000L	9TC40PRD	9HU82LCZ	9HU82LCZ							
807	1.344E-11	0.0000	SHDL	9HU0000L	9TC40PRD	9HU82LCZ	9HU82LCZ							
808	1.344E-11	0.0000	SHDL	9HU0000L	9TC40PRD	9HU82LCZ	9HU82LCZ							

862	1.200E-11	0.0000	LF1A	DHU0000L	DRE53ACP	BOO02200	HHU01BSZ	918	1.036E-11	0.0000	LF1B	VOOOOUUW	9HU0000L	9ARX82QW	
863	1.200E-11	0.0000	LF1A	CCLO0PRU	RHU002CL	BOO992UW	HHU01BSZ	919	1.026E-11	0.0000	LF1A	VOOOOUUW	9HU0000L	9RE53ACP	
864	1.200E-11	0.0000	LF1A	DHU0000L	DARX82QP	BOO02200	HHU01BSZ	920	1.026E-11	0.0000	LF1A	VOOOOUUW	9HU0000L	9ARX82QP	
865	1.200E-11	0.0000	RIFS	RHU302QL	RHU002CL	RBI211QU	BOO02200	921	1.026E-11	0.0000	ANTC	VOOOOUUW	9HU0000L	9RE53ACP	
866	1.191E-11	0.0000	SCRM	VOOOOUUW	DCR001OF	DCR002OF	BOO02200	922	1.026E-11	0.0000	ANTC	VOOOOUUW	9HU0000L	9ARX82QP	
	HHU01BSZ							923	1.010E-11	0.0000	TAIR	9HU0000L	9HR505AW	9HU500C2	
867	1.177E-11	0.0000	TSDC	EDL41OHU	DCR001OF	BOO02200	ECB23BYO	924	1.010E-11	0.0000	TAIR	9HU0000L	9HR501AW	9HU500C2	
868	1.177E-11	0.0000	TSDC	EDL41OHU	ROO99A00	DCR001OF	BOO02200	925	1.010E-11	0.0000	TAIR	9HU0000L	9HR502AW	9HU500C2	
	ECB15CYO							926	1.010E-11	0.0000	TAIR	9HU0000L	9HR751AW	9HU500C2	
869	1.175E-11	0.0000	TSDC	JSG10720	JBS001OB	EGE4EFHW	DCR001OF	927	1.010E-11	0.0000	TAIR	9HU0000L	9HR504AW	9HU500C2	
	BOO02200							928	1.000E-11	0.0000	OCSL	CSR10GRU	NPMOVCMW		
870	1.156E-11	0.0000	TSDC	DCR001OE	BOO02200	HDM7475Q		929	1.000E-11	0.0000	OCSL	CSR30IRU	NPMOVCMW		
871	1.152E-11	0.0000	OCSL	RTC300RU	RHU002CL	RCA220RU									
872	1.152E-11	0.0000	OCSL	RHU002CL	RVI320QU	RCA220RU	OGF4EFHW								
873	1.118E-11	0.0000	TSDC	OOO12200	OOO20200	3TRT560U									
	9CR001OF														
874	1.110E-11	0.0000	OCPP	DCR001OF	DCR002OE	BOO02200	TFLJINFU								
875	1.110E-11	0.0000	OCPP	DCR001OE	DCR002OF	BOO02200	TFLJINFU								
876	1.109E-11	0.0000	LONP	DCR001OF	DCR002OF	BOO02200	TFLJINFU								
877	1.109E-11	0.0000	LOCP	DCR001OF	DCR002OF	BOO02200	TFLJINFU								
878	1.109E-11	0.0000	OCSL	DCR001OF	DCR002OF	BOO02200	TFLJINFU								
879	1.109E-11	0.0000	RISB	RHU302QL	RHU002CL	RAM2A1QW	RCA2C0RW								
880	1.109E-11	0.0000	RISB	RHU302QL	RHU002CL	RAM2A1QW	RCA2B0RW								
881	1.109E-11	0.0000	RISB	RHU302QL	RHU002CL	RCA2A0RW	RAM2B1QW								
882	1.109E-11	0.0000	RISB	RHU302QL	RHU002CL	RCA2B0RW	RAM2C1QW								
883	1.109E-11	0.0000	RISB	RHU302QL	RHU002CL	RCA2A0RW	RAM2C1QW								
884	1.109E-11	0.0000	RISB	RHU302QL	RHU002CL	RAM2B1QW	RCA2C0RW								
885	1.107E-11	0.0000	LOCP	DCR001OE	DCR002OE	BOO02200	HHU01BSZ								
886	1.107E-11	0.0000	LONP	DCR001OE	DCR002OE	BOO02200	HHU01BSZ								
887	1.107E-11	0.0000	OCSL	DCR001OE	DCR002OE	BOO02200	HHU01BSZ								
888	1.100E-11	0.0000	RISA	DCR001OF	DCR002OE	BOO02200	HFL717SE								
889	1.100E-11	0.0000	RISA	DCR001OF	DCR002OE	BOO02200	WFL095YE								
890	1.100E-11	0.0000	RISA	DCR001OE	DCR002OF	BOO02200	WFL095YE								
891	1.100E-11	0.0000	RISA	DCR001OE	DCR002OF	BOO02200	HFL717SE								
892	1.095E-11	0.0000	OCSP	DCR001OE	DCR002OF	BOO02200	HHU01BSZ								
893	1.095E-11	0.0000	OCSP	DCR001OF	DCR002OE	BOO02200	HHU01BSZ								
894	1.089E-11	0.0000	SHDL	DCR001OF	DCR002OF	BOO02200	YFN0560R								
	YFN0580R														
895	1.080E-11	0.0000	TSDC	VOOOOUUW	DCR001OF	BOO02200	HFN705SR								
	HFN706SR														
896	1.080E-11	0.0000	TSDC	VOOOOUUW	DCR001OF	BOO02200	HFN707SR								
	HFN708SR														
897	1.080E-11	0.0000	RIFS	RHU302QL	RHU002CL	RDC2B1QW	RAM2C1QW								
898	1.080E-11	0.0000	RIFS	RHU302QL	RHU002CL	RDC2A1QW	RAM2C1QW								
899	1.080E-11	0.0000	RIFS	RHU302QL	RHU002CL	RND2B0RW	RAM2C1QW								
900	1.080E-11	0.0000	RIFS	RHU302QL	RHU002CL	RND2A0RW	RAM2B1QW								
901	1.080E-11	0.0000	RIFS	RHU302QL	RHU002CL	RDC2A1QW	RAM2B1QW								
902	1.080E-11	0.0000	RIFS	RHU302QL	RHU002CL	RND2A0RW	RAM2C1QW								
903	1.080E-11	0.0000	RIFS	RHU302QL	RHU002CL	RAM2B1QW	RDC2C1QW								
904	1.080E-11	0.0000	RIFS	RHU302QL	RHU002CL	RAM2A1QW	RDC2B1QW								
905	1.080E-11	0.0000	RIFS	RHU302QL	RHU002CL	RAM2A1QW	RND2B0RW								
906	1.080E-11	0.0000	RIFS	RHU302QL	RHU002CL	RAM2A1QW	RND2C0RW								
907	1.080E-11	0.0000	RIFS	RHU302QL	RHU002CL	RAM2A1QW	RDC2C1QW								
908	1.080E-11	0.0000	RIFS	RHU302QL	RHU002CL	RAM2B1QW	RND2C0RW								
909	1.068E-11	0.0000	SCRM	DHU0000L	DAM4ALAW	BOO02200	KIN003UW								
910	1.068E-11	0.0000	SCRM	DHU0000L	DAM4BLQW	BOO02200	KIN003UW								
911	1.068E-11	0.0000	SCRM	DHU0000L	DARX82QW	BOO02200	KIN003UW								
912	1.058E-11	0.0000	LF2B	RHU002CL	CCD301RU	BOO02200	KIN003UW								
913	1.051E-11	0.0000	LF1A	ROO99A00	DCR001OF	DCR002OF	BOO02200								
	EGE1EFMW		ESWR1BYO												
914	1.050E-11	0.0000	TBOP	DCR001OF	DCR002OE	BOO02200	KIN003UW								
915	1.050E-11	0.0000	TBOP	DCR001OE	DCR002OF	BOO02200	KIN003UW								
916	1.036E-11	0.0000	LF1B	VOOOOUUW	9HU0000L	9AM4BLQW									
917	1.036E-11	0.0000	LF1B	VOOOOUUW	9HU0000L	9AM4ALAW

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