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Relative Performance of Metallic Fuel and Oxide Fuel during Unprotected Accident Conditions in a 3500 MWth Pool-Type LMR

by R. A. Wigeland,
R. B. Turski, and R. K. Lo

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September 1990

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RELATIVE PERFORMANCE OF METALLIC FUEL AND OXIDE FUEL DURING
UNPROTECTED ACCIDENT CONDITIONS IN A 3500 MWth POOL-TYPE LMR

by

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TABLE OF CONTENTS

	<u>Page</u>
ABSTRACT.....	xi
I. INTRODUCTION.....	1
II. SAS4A MODELLING.....	4
A. Core Models.....	4
1. Oxide Fuel Core Modelling.....	4
2. Metallic Fuel Core Modelling.....	9
3. Reactivity Feedback Modelling.....	10
a. Radial Core Expansion.....	10
b. Control Rod Driveline Expansion.....	13
c. Other Reactivity Feedback Models.....	16
B. Balance of Plant Modelling.....	17
1. Primary Circuit.....	22
2. Intermediate Loop.....	23
3. DRACS Loops.....	23
III. RESULTS OF UNPROTECTED TRANSIENTS.....	25
A. Unprotected LOF.....	25
1. Oxide Fuel Core.....	25
2. Metallic Fuel Core.....	30
3. Relative Performance.....	34
B. Unprotected TOP.....	36
1. Oxide Fuel Core.....	36
2. Metallic Fuel Core.....	40
3. Relative Performance.....	44
C. Unprotected LOHS.....	46
1. Oxide Fuel Core.....	46
2. Metallic Fuel Core.....	51
3. Relative Performance.....	55
IV. SUMMARY.....	57
APPENDIX A. SAS4A MODELLING OF THE OXIDE FUEL CORE.....	A-1
I. Channel Modelling of the Core.....	A-1
II. General Integer Input and Options.....	A-5

TABLE OF CONTENTS (Contd.)

	<u>Page</u>
A. Block 1, INPCOM.....	A-5
B. Block 11, OPCIN.....	A-8
C. Block 51, INPCHN.....	A-9
III. Core Geometry.....	A-13
A. Block 61, GEOMIN.....	A-13
IV. SAS4A Reactivity Worth Tables.....	A-19
A. Block 12, POWINA.....	A-20
B. Block 62, POWINC.....	A-24
V. Thermophysical Properties.....	A-26
A. Block 13, PMATCH.....	A-26
B. Block 63, PMATCH.....	A-30
C. Block 65, FUELIN.....	A-32
VI. Coolant Flow.....	A-32
A. Block 14, PRIMIN.....	A-33
B. Block 64, COOLIN.....	A-34
APPENDIX B. SAS4A MODELLING OF THE METALLIC FUEL CORE.....	B-1
I. Channel Modelling of the Core.....	B-1
II. General Integer Input and Options.....	B-1
A. Block 1, INPCOM.....	B-5
B. Block 11, OPCIN.....	B-7
C. Block 51, INPCHN.....	B-8
III. Core Geometry.....	B-11
A. Block 61, GEOMIN.....	B-12
IV. SAS4A Reactivity Worth Tables.....	B-14
A. Block 12, POWINA.....	B-15
B. Block 62, POWINC.....	B-19
V. Thermophysical Properties.....	B-20
A. Block 13, PMATCH.....	B-21
B. Block 63, PMATCH.....	B-26
C. Block 65, FUELIN.....	B-28

TABLE OF CONTENTS (Contd.)

	<u>Page</u>
VI. Coolant Flow.....	B-28
A. Block 14, PRIMIN.....	B-29
B. Block 64, COOLIN.....	B-29
APPENDIX C. RADIAL CORE EXPANSION REACTIVITY FEEDBACK MODEL.....	C-1
I. Metallic Fuel Design.....	C-1
A. Block 1, INPCOM.....	C-1
B. Block 12, POWINA.....	C-3
II. Oxide Fuel Design.....	C-5
A. Block 1, INPCOM.....	C-5
B. Block 12, POWINA.....	C-6
APPENDIX D. PRIMARY AND INTERMEDIATE SYSTEMS MODELLING.....	D-1
I. Primary Circuit.....	D-1
II. Intermediate Loop.....	D-8
III. DRACS Loops.....	D-8
IV. Block 3, INPMR4.....	D-9
V. Block 18, PMR4IN.....	D-22
APPENDIX E. DEFORM-4 INPUT DATA FOR THE OXIDE FUEL CORE.....	E-1
I. Block 1, INPCOM.....	E-1
II. Block 13, PMATCM.....	E-1
III. Block 51, INPCHN.....	E-5
IV. Block 61, GEOMIN.....	E-6
V. Block 62, POWINC.....	E-7
VI. Block 63, PMATCH.....	E-7
APPENDIX F. SAS4A INPUT DATA LISTING FOR THE OXIDE FUEL CORE.....	F-1
I. Steady-State Input Data.....	F-1
II. DEFORM-4 Input Data.....	F-14
III. Input Data for the Transients.....	F-17
A. Unprotected Loss-of-Flow Accident.....	F-17
B. Unprotected Transient Overpower Accident.....	F-17
C. Unprotected Loss-of-Heat-Sink Accident.....	F-18
APPENDIX G. SAS4A INPUT DATA LISTING FOR THE METALLIC FUEL CORE....	G-1
I. Steady-State Input Data.....	G-1
II. Input Data for the Transients.....	G-14

TABLE OF CONTENTS (Contd.)

	<u>Page</u>
A. Unprotected Loss-of-Flow Accident.....	G-14
B. Unprotected Transient Overpower Accident.....	G-15
C. Unprotected Loss-of-Heat-Sink Accident.....	G-16
APPENDIX H. SAS4A CODE MODIFICATIONS.....	H-1

LIST OF FIGURES

	<u>Page</u>
1. LSPB Pool Reactor Plant.....	2
2. Subassembly Designs for the Oxide Fuel and Metallic Fuel Cores.....	8
3. Reactivity Feedback from Radial Core Expansion as a Function of Normalized Power-to-Flow Ratio for the Oxide Fuel Core Design.....	14
4. Reactivity Feedback from Radial Core Expansion as a Function of Normalized Power-to-Flow Ratio for the Metallic Fuel Core Design.....	15
5. Schematic of the PRIMAR-4 Model of the LSPB Reactor Plant....	18
6. Normalized Power and Flow as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Flow Accident with an Oxide Fuel Core.....	27
7. Reactivity Feedback as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Flow Accident with an Oxide Fuel Core.....	28
8. Peak Fuel, Cladding, and Coolant Temperatures, Core Inlet Temperature and Coolant Saturation Temperature in Channel 1 as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Flow Accident with an Oxide Fuel Core.....	29
9. Normalized Power and Flow as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Flow Accident with a Metallic Fuel Core.....	31
10. Reactivity Feedback as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Flow Accident with a Metallic Fuel Core.....	32
11. Peak Fuel, Cladding, and Coolant Temperatures, Core Inlet Temperature and Coolant Saturation Temperature in Channel 1 as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Flow Accident with a Metallic Fuel Core.....	33
12. Normalized Power and Flow as a Function of Time from the Start of the Transient for an Unprotected Transient Overpower Accident with an Oxide Fuel Core and a Control Rod Initiator of 0.10 \$......	37

LIST OF FIGURES (Contd.)

		<u>Page</u>
13.	Reactivity Feedback as a Function of Time from the Start of the Transient for an Unprotected Transient Overpower Accident with an Oxide Fuel Core and a Control Rod Initiator of 0.10 \$.....	38
14.	Peak Fuel, Cladding, and Coolant Temperatures, Core Inlet Temperature and Coolant Saturation Temperature in Channel 1 as a Function of Time from the Start of the Transient for an Unprotected Transient Overpower Accident with an Oxide Fuel Core and a Control Rod Initiator of 0.10 \$.....	39
15.	Normalized Power and Flow as a Function of Time from the Start of the Transient for an Unprotected Transient Overpower Accident with a Metallic Fuel Core and a Control Rod Initiator of 0.11 \$.....	41
16.	Reactivity Feedback as a Function of Time from the Start of the Transient for an Unprotected Transient Overpower Accident with a Metallic Fuel Core and a Control Rod Initiator of 0.11 \$.....	42
17.	Peak Fuel, Cladding, and Coolant Temperatures, Core Inlet Temperature and Coolant Saturation Temperature in Channel 1 as a Function of Time from the Start of the Transient for an Unprotected Transient Overpower Accident with a Metallic Fuel Core and a Control Rod Initiator of 0.11 \$.....	43
18.	Normalized Power and Flow as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Heat-Sink Accident with an Oxide Fuel Core.....	47
19.	Reactivity Feedback as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Heat-Sink Accident with an Oxide Fuel Core.....	48
20.	Peak Fuel, Cladding, and Coolant Temperatures, Core Inlet Temperature and Coolant Saturation Temperature in Channel 1 as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Heat-Sink Accident with an Oxide Fuel Core.....	49
21.	Normalized Power and Flow as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Heat-Sink Accident with a Metallic Fuel Core.....	52
22.	Reactivity Feedback as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Heat-Sink Accident with a Metallic Fuel Core.....	53

LIST OF FIGURES (Contd.)

	<u>Page</u>
23. Peak Fuel, Cladding, and Coolant Temperatures, Core Inlet Temperature and Coolant Saturation Temperature in Channel 1 as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Heat-Sink Accident with a Metallic Fuel Core.....	54
A-1. Assembly Assignment for the 3500 MWth Oxide Fuel Heterogeneous Core.....	A-2
A-2. SAS4A Axial Zones and Fuel Nodes for the Oxide Fuel Core.....	A-10
B-1. Assembly Assignment for the 3500 MWth Metallic Fuel Heterogeneous Core.....	B-2
B-2. SAS4A Axial Zones and Fuel Nodes for the Metallic Fuel Core..	B-9
C-1. Core Restraint System Model for the 3500 MWth Metallic and Oxide Fuel Core Designs.....	C-2
D-1. Schematic of the PRIMAR-4 Model of the LSPB Reactor Plant....	D-2
D-2. Elevations of the Reactor Internals in LSPB.....	D-25
D-3. Cross-Section Elevations of the Primary Heat Transfer System in LSPB.....	D-26
D-4. Intermediate Loop Heat Transfer System Profile in LSPB.....	D-27
D-5. Schematic of the Decay Heat Removal Systems in LSPB.....	D-30
D-6. Schematic of the Intermediate Heat Exchanger in LSPB.....	D-35
D-7. Schematic of the Two-Stage Primary Pump in LSPB.....	D-36
D-8. Schematic of the Decay Heat Removal Heat Exchanger in LSPB...	D-39
D-9. Elevations for the Air Dump Heat Exchanger in LSPB.....	D-42

LIST OF TABLES

	<u>Page</u>
I. Subassembly Design for Metallic and Oxide Fuel Cores.....	6
II. Subassembly Axial Geometry for Metallic Fuel and Oxide Fuel Designs.....	7
III. Reactivity Feedback Coefficients at EOEC.....	11
IV. PRIMAR-4 Model of the LSPB Reactor System: Volumes.....	19
V. PRIMAR-4 Model of the LSPB Reactor System: Elements.....	20
A-1. SAS4A Channel/Assembly Parameters at EOEC for Oxide Core.....	A-3
B-1. SAS4A Channel/Assembly Parameters at EOEC for Metal Core.....	B-3
D-1. PRIMAR-4 Model of the LSPB Reactor System: Volumes.....	D-3
D-2. PRIMAR-4 Model of the LSPB Reactor System: Elements.....	D-4

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ABSTRACT

Interchangeable oxide fuel and metallic fuel core designs for a 3500 MWth pool-type, liquid-metal-cooled reactor are analyzed to estimate the response of each core design to hypothetical unprotected loss-of-flow, transient overpower, and loss-of-heat-sink accidents. The SAS4A accident analysis computer code is used to provide best-estimate calculations of the consequences of each accident for each fuel type. A complete description of the generation of the SAS4A modelling is included. The relative safety advantages of oxide and metallic fuel are established by analysis and comparison of the calculated results.

I. INTRODUCTION

The U. S. Department of Energy and the Steering Committee for European Fast Breeder Research and Development Cooperation have agreed to carry out an informal comparison of the safety issues between oxide fuel and metallic fuel reactor designs. The basis for the comparison is focused on the following five unprotected (i.e., without scram) accident initiators:

1. Loss-of-flow
2. Loss-of-heat-sink
3. Slow transient overpower (e.g. control rod withdrawal)
4. Loss of inlet piping integrity
5. Local faults

This report summarizes initial reference calculations for the first three transients listed above.

For this study, 3500 MWth oxide and metallic fuel core designs were developed [1,2] which use the same geometry, control rod arrangement, assembly length, and essentially the same core layout. Within these constraints, the oxide core design was optimized first to provide the best core layout which provides a combination of a low peaking factor (i.e., low peak/average subassembly power) which changes as little as possible with burnup, and a low burnup control swing. For the metallic fuel core design, the oxide core layout and geometry were changed slightly by replacing six drivers with internal blankets, reducing the core height by 4 in., and removing the axial blankets to provide a better balance between the peaking factor and burnup control swing. Both cores have the same overall dimensions. The remainder of the reactor plant was based on the 1986 version of the Large Scale Prototype Breeder (LSPB) plant [3]; see Fig. 1. This plant design was selected mainly because it was of an appropriate size and a significant amount of design data was readily available at Argonne National Laboratory (ANL).

The agreement also proposed that Argonne National Laboratory perform the initial safety analyses using SAS4A [4], and that the SAS4A input data would be transmitted to Kernforschungszentrum Karlsruhe (KfK), where the SAS4A code

Fig. 1. LSPB Pool Reactor Plant.

is available. This would allow calculations to be performed at both installations, verifying that both groups were starting from the same point. The design data for the new core designs and for LSPB were reviewed, and information on all of the major systems was assembled into a basic data package, which included core and balance of plant information. This data package was sent to the Steering Committee for European Fast Breeder Research and Development Cooperation [5], and was used at ANL as the basis for generating all of the input data for SAS4A. The initial SAS4A input data decks were also sent to KfK, along with some minor code modifications [6]. The intent is that both groups start with the same basic data during the analysis, so that any differences will be on other issues, and the safety assessments will at least have a common foundation.

The first section of this report is devoted to a detailed description of the assembling of the SAS4A input data, including all of the assumptions and any other information which is required but not in the original data package. The remainder of the report covers the results obtained at ANL using SAS4A for analyzing the response of both reactor cores to unprotected loss-of-flow, transient overpower, and loss-of-heat-sink accidents, along with comparisons and the relative advantages of one design over the other, where appropriate. These calculations are intended to provide the best estimate currently available, and use all of the features of the SAS4A code required for this initial study. For this first effort, calculations are continued only as long as coolant boiling in the core does not occur. This proves to be sufficient for the purposes of the study as far as these accidents are concerned.

II. SAS4A MODELLING

The modelling of the reactor plant with SAS4A can be separated into two areas; the core and the remainder of the reactor system. The section covering the core modelling is first, including the SAS4A channel groupings, neutronics, and geometry. The remainder of the system is then discussed, including the intermediate heat transport system and both decay heat removal systems.

A. Core Models

The modelling of the core with SAS4A is based on the grouping of similar subassemblies into "channels", where each channel represents the average of the group. For each channel, the appropriate data for the geometry, neutronics, thermophysical properties, and coolant flow parameters are developed which also reflect the average of the group. This process is reviewed in detail in this section and Appendices A and B, for the oxide and metallic fuel cores, respectively. Many of the calculations are similar, if not identical, for the metallic fuel and oxide fuel core designs.

1. Oxide Fuel Core Modelling

The fundamental step in forming a core model using SAS4A is to group the core subassemblies into channels, where each channel represents one or more subassemblies. The grouping of assemblies into SAS4A channels follows the following guidelines. For safety analyses, one channel is devoted to the "lead" assembly, that is, the assembly which will exhibit the highest temperatures during any particular transient and which will most likely be the assembly to reach any given failure threshold. The criteria for selecting the lead driver assembly includes isolating the assembly with the combination of highest power-to-flow ratio and the highest power. For the present oxide core design the lead assembly has the highest power-to-flow ratio coupled with a high power. This assembly is in the second row of driver assemblies. The details of the selection process, and the other channels groupings are described in Appendix A.

For the purposes of analyzing the unprotected loss-of-flow, transient overpower, and loss-of-heat-sink accidents in heterogeneous core designs, it has proven to be sufficiently accurate to include all of the remaining drivers in the core in a single channel, and all of the internal blankets in another channel. The three channels model all of the core within the boundary of the first row of radial blanket assemblies except for control assemblies. For the subassemblies exterior to this region, another channel could be used for the radial blankets, and so on, but these assemblies can also be included as part of the PRIMAR-4 model of the primary system. This turns out to be computationally more efficient, without any real loss in accuracy. As a result, a three channel model is selected for the core model. The treatment of the three channels is described in this section, while the modelling of the radial blankets, shields, control rods, and core bypass flow is discussed in Section II.B.1 with the remainder of the primary system.

The majority of the numerical details for generating the input data are included in Appendix A, while the general design data is given in this section. The basic geometry of the core assemblies is given in Table I, for both oxide and metallic fuel designs [1]. It should be noted that many of the features are identical between the two designs, such as duct pitch, hexcan dimensions, pin number and dimensions, and so on. These data are converted into the appropriate SAS4A input as described in Appendix A. The design of the subassembly in the axial direction is given in Table II [1] for both the oxide and metallic fuel designs and illustrated in Fig. 2. There are many differences between the two designs, especially the core location and fission gas plenum treatment. The oxide core design has a divided fission gas plenum, with the larger section below the core. This locates the active core region fairly high in the assembly. The modelling of this axial geometry is given in Appendix A, covering the creation of several axial zones for the SAS4A channel models, and further nodalization in each zone, including the core. The axial node structure in the core region is the same as that used for generating the neutronics information [1], and is also discussed in Appendix A.

The neutronics information, including the worth curves for fuel, cladding, coolant, and Doppler, is generated on an axial node structure which

TABLE I. Subassembly Design for Metallic and Oxide Fuel Cores

<u>Subassembly Design</u>	<u>Metallic Fuel</u>	<u>Oxide Fuel</u>
Duct Design	Hexagonal	
Duct Pitch	0.155727 m (6.131 in.)	
Duct Wall Thickness	0.003810 m (0.150 in.)	
Duct Outer Flat-to-Flat	0.150647 m (5.931 in.)	
Interassembly Gap	0.005080 m (0.200 in.)	
Spacer Type	straight-start wire wrap, 12 in. pitch	
Number of Pins/Assembly (driver)	271	
Number of Pins/Assembly (blanket)	169	
P/D Ratio (driver)	1.180	
P/D Ratio (blanket)	1.087	
<u>Driver Fuel Pin Design</u>		
Pin Outer Diameter	0.007239 m (0.285 in.)	
Cladding Thickness	0.000559 m (0.022 in.)	
Wire Spacer Diameter	0.001245 m (0.049 in.)	
Bond Type	Sodium	Helium
Pellet Density, % T.D.	15.7 gm/cc	86.8
Smear Density, % T.D.	75	82.5
Fuel Length	0.9144 m (36 in.)	1.0160 m (40 in.)
Upper/Lower Blanket Length		0.3556 m (14 in.)
Top Plenum	1.3208 m (52 in.)	0.4064 m (16 in.)
Bottom Plenum		1.0160 m (40 in.)
<u>Internal/Radial Blanket Pin Design</u>		
Pin Outer Diameter	0.009957 m (0.392 in.)	
Cladding Thickness	0.000559 m (0.022 in.)	
Wire Spacer Diameter	0.000864 m (0.034 in.)	
Bond Type	Sodium	Helium
Pellet Density, % T.D.	15.7 gm/cc	95.7
Smear Density, % T.D.	85.0	93.3
Blanket Length	0.9144 m (36 in.)	1.7272 m (68 in.)
Top Plenum	1.3208 m (52 in.)	0.4064 m (16 in.)
Bottom Plenum		1.0160 m (40 in.)

TABLE. II. Subassembly Axial Geometry for Metallic Fuel and Oxide Fuel Designs

Subassembly Component	Component Length	
	Metallic Fuel	Oxide Fuel
Top End Fitting	0.2286 m (9.0 in.)	0.2286 m (9.0 in.)
Top Shield (including plenum and support)	0.5588 m (22.0 in.)	0.5588 m (22.0 in.)
Expansion Space	0.0508 m (2.0 in.)	0.0508 m (2.0 in.)
Fuel Pin		
Top End Cap	0.0254 m (1.0 in.)	0.0254 m (1.0 in.)
Top Plenum	1.3208 m (52.0 in.)	0.4064 m (16.0 in.)
Top Axial Blanket		0.3556 m (14.0 in.)
Fuel Region	0.9144 m (36.0 in.)	1.0160 m (40.0 in.)
Bottom Axial Blanket		0.3556 m (14.0 in.)
Bottom Plenum		1.0160 m (40.0 in.)
Bottom End Cap	0.0254 m (1.0 in.)	0.0254 m (1.0 in.)
Pin Support	0.0635 m (2.5 in.)	0.0635 m (2.5 in.)
Bottom Shield (including plenum and support)	0.6096 m (24.0 in.)	0.1524 m (6.0 in.)
Space	0.4572 m (18.0 in.)	
Orificing	0.1016 m (4.0 in.)	0.1016 m (4.0 in.)
Inlet Nozzle	0.3429 m (13.5 in.)	0.3429 m (13.5 in.)
<hr/>		
Overall Length	4.6990 m (185.0 in.)	4.6990 m (185.0 in.)
Elevation of Core Axial Midplane Relative to the Bottom of the Assembly	2.0574 m (81.0 in.)	2.5654 m (101.0 in.)
Assumed Elevation of the Above-Core Load Pad*	2.7178 m (107.0 in.)	3.5306 m (139.0 in.)

*Load pad elevation assumed to be 0.1016 m (4 in.) above the top of the SAS4A neutronics axial mesh.

Fig. 2. Subassembly Designs for the Oxide Fuel and Metallic Fuel Cores.

covers the region from the bottom of the lower axial blanket to the top of the upper axial blanket. Given the axial extent of the blankets, the worths are quite small at the boundaries, and little error is introduced by ignoring the reactivity implications of motion outside of these boundaries. The neutronics information is initially generated on a subassembly-by-subassembly basis. For use in SAS4A, the information is grouped as described in Appendix A.

The thermophysical properties of the oxide fuel are listed in Appendix A for all of the fuel types used in simulating the fuel pins. This includes the driver fuel, driver axial blanket, and internal blanket material. The properties are typical values for mixed-oxide fuel. The properties of the cladding and structure are those for HT-9. The details on the core orificing [2] are also included in Appendix A, as are calculations for the remaining channel input variables.

2. Metallic Fuel Core Modelling

The modelling of the metallic fuel core follows the same pattern as the oxide core, and many of the associated calculations are identical. The pin geometry is the same for both fuel types, although there is a difference in the axial location of the core within the subassembly; see Tables I and II [1]. The overall length of the subassembly is the same, as both cores fit within the same design envelope. The subassembly selected for the lead driver has a combination of the highest power and a high power-to-flow ratio. The selection is discussed in detail in Appendix B. The remaining assemblies are grouped in the same manner as for the oxide core.

The numerical and modelling details for generating the SAS4A input data are covered in Appendix B for the metallic core. This is a comparable discussion to the oxide fuel modelling covered in Appendix A. The construction of the channel data, including the neutronics information, is reviewed. The generation of the pertinent thermophysical properties is also included for driver and blanket fuel, along with sodium. The sodium is included for the simulation of fuel pin regions above and below the active fuel. Since there are not any axial blankets in this design, the neutronics calculational mesh had to be extended into the regions above and below the

core in order to provide the fuel worth for axial fuel motion into these regions. This is also the reason that the SAS4A axial node structure for the core is longer than the fuel within the fuel pin. The core orificing [2] is also covered in Appendix B, along with the other required input data.

3. Reactivity Feedback Modelling

An integral part of the core modelling in SAS4A is the accounting for various reactivity feedback mechanisms. The reactivity feedbacks control the course of unprotected transients, and need to be modelled as accurately as possible to ensure the best possible calculation of accident consequences. For most of the reactivity feedbacks, the pertinent data is included as part of the neutronics information generated for the core design. As described in the previous two sections, the values for these feedback components are included in the procedure combining subassembly data into the SAS4A channels. These reactivity feedbacks include fuel Doppler, fuel and cladding axial expansion, and coolant density variation. Two important reactivity feedback mechanisms which require additional input data are control rod driveline expansion and radial core expansion. Since the method for generating the input data is the same for both core designs, the modelling for both core designs is described at the same time.

a. Radial Core Expansion

The reactivity feedback from radial core expansion arises from the change in reactivity as the core changes in size. As such, the reactivity feedback is proportional to the core radius, with the feedback determined by a coefficient with the units of $\$/m$. The current method for determining this coefficient is to evaluate the change in reactivity associated with a uniform expansion of the core [1]. This corresponds to an increase in subassembly pitch throughout the core, with all of the core subassemblies remaining straight and vertical. The value of this coefficient for both the metallic and oxide fuel designs is given in Table III [1]. In the present version of SAS4A, this coefficient is used in connection with calculations which determine the overall size of the core.

TABLE. III. Reactivity Feedback Coefficients at EOE

Reactivity Feedback Component	Metallic Fuel		Oxide Fuel	
Sodium Void Worth, \$				
Drivers	4.98		3.60	
Internal Blankets	2.31		1.68	
Radial Blankets	- 0.03		- 0.08	
Sodium Density Worth, \$				
Drivers	4.51		3.14	
Internal Blankets	1.99		1.42	
Radial Blankets	- 0.06		- 0.11	
Doppler, $10^{-3} T(dk/dT)$				
	Fuel	Structure	Fuel	Structure
Non-voided				
Drivers	1.929	0.508	4.576	1.033
Internal Blankets	1.726	0.174	4.337	0.380
Radial Blankets	0.181	0.015	0.406	0.030
Voided				
Drivers	1.155	0.378	3.439	0.762
Internal Blankets	1.180	0.136	3.518	0.293
Radial Blankets	0.138	0.012	0.340	0.023
Axial Expansion, \$/cm				
Fuel				
Driver	- 0.718		- 0.693	
Internal Blankets	0.081		0.137	
Radial Blankets	- 0.004		- 0.007	
Fuel and Cladding				
Drivers	- 0.641		- 0.617	
Internal Blankets	0.118		0.170	
Radial Blankets	- 0.007		- 0.009	
Fuel and Total Structure				
Drivers	- 0.586		- 0.562	
Internal Blankets	0.145		0.195	
Radial Blankets	- 0.008		- 0.010	
Radial Expansion, \$/m	- 76.5		- 64.0	
Control Rod, \$/m	- 11.6		- 7.12	

The calculation for determining the reactivity change as the core changes size is contained in the detailed radial core expansion reactivity feedback model in SAS4A [7]. This model calculates the axial variation in core radius based on an average subassembly in the outer driver row of the core. The axial profile of core radius is used to define the initial core size at steady-state, and is also used at each step of the transient calculation so that any changes in radius are evaluated for each axial node in the core. The radial core expansion reactivity feedback coefficient is apportioned among the axial nodes in the core according to the power shape. Radial motion near the core midplane is then associated with a larger reactivity change than motions near the top and bottom of the core.

The use of this model requires additional input data concerning the core geometry and temperature profiles. The core restraint system is assumed to be the limited-free-bow type which has a core restraint ring around the load pads at the top of the subassemblies. The subassemblies also have load pads just above the active core region as shown in Fig. 2. The connection at the grid, or core support, plate is assumed to provide a well-defined radial location for each subassembly, but each assembly is free to tilt at the grid plate. The limited free-bow core restraint is typically designed to accommodate a deviation from vertical of a few tenths of a degree, which is large compared to the typical subassembly deflection. The diagram for the core restraint design is shown in Fig. C-1 in Appendix C.

The detailed calculation of the input data is given in Appendix C. Much of the design information is not known at this stage, so assumptions were made. Fortunately the calculation of the reactivity feedback from radial expansion of the core is not sensitive to any one of the variables by themselves, but it is important to have a set of input data which is consistent with typical design goals. In this sense, it is possible to generate any number of sets of input data, all of which would give comparable answers during the transient. However, it is not a simple matter to change an individual value within each set, as they are all interrelated. As stated in Appendix C, the design criteria include: 1) the pitch at the grid plate and at the load pad elevations should be the same at normal operating conditions, and 2) the core should have a well-defined geometry for all normalized power-to-flow ratios above 0.7.

The modelling provides for reactivity feedback from radial core expansion as shown in Figs. 3 and 4 for the oxide and metallic fuel designs, respectively. The metallic fuel design has a reactivity feedback at nominal full power conditions of -0.259 \$ per unit change in power-to-flow ratio, or -0.00162 \$/C change in coolant temperature at the load pads. The oxide fuel design has -0.272 \$ per unit change in normalized power-to-flow ratio at nominal full power conditions, or -0.00171 \$/C for a change in coolant temperature at the load pads. The larger gradient of reactivity for changes in power-to-flow ratio with the oxide core is a result of the high location of the above-core load pad within the assembly, which offsets the effect of a lower radial core expansion reactivity feedback coefficient. These results are for constant inlet temperature at the nominal operating inlet temperature, and the curve represents steady-state conditions at any power-to-flow ratio.

The break in each curve which occurs at a normalized power-to-flow ratio of 0.7 is associated with a change in core loading conditions. Basically, below this point the thermal bending of the subassemblies is not sufficient to cause the top load pads to push out against the core restraint ring and the above-core load pads to be pushed together. This behavior is not normally important in transient calculations unless there is a loss of the major heat sink with core flow being maintained, as shown in detail in Section III. Based on these results, it appears that both core designs have comparable reactivity feedback associated with changes in core size, so that there is not a significant difference in this component with fuel type. Differences in reactivity feedback from radial core expansion during the transients are observed as a result of the different temperature histories that occur for each fuel type.

b. Control Rod Driveline Expansion

The control rod driveline expansion reactivity feedback model is designed to calculate the reactivity effect of relative motion between the core and the control rods. In the LSPB design, the control rods are suspended from the reactor vessel head, Fig. 1, while the core is supported by the lower part of the reactor vessel, which is also suspended from the top. Relative

Fig. 3. Reactivity Feedback from Radial Core Expansion as a Function of Normalized Power-to-Flow Ratio for the Oxide Fuel Core Design.

Fig. 4. Reactivity Feedback from Radial Core Expansion as a Function of Normalized Power-to-Flow Ratio for the Metallic Fuel Core Design.

motion of the core and the control rods can be achieved by either expansion of the control rod drivelines or by changes in length of the reactor vessel wall. For both fuel designs, the top of the subassembly is at the same elevation, which means that the length of the driveline which is in the sodium pool is the same, 6.2992 m. For the purposes of calculating the driveline expansion, the total driveline length subject to thermal expansion is the sum of this value plus the length from the top of the subassembly to the top of the fuel. This is 2.1844 m in the metallic fuel design and 1.6256 m in the oxide fuel design. The other input associated with the driveline characteristics are listed in Appendices A and B, where many of the values are estimates obtained from previous designs and suggested for use in SAS4A when the detailed design is not known. Fortunately, the course of most transients is not very sensitive to these values.

The relative motion of the core and the control rods caused by changes in the length of the reactor vessel is handled in the systems model, PRIMAR-4. PRIMAR-4 requires input on the model components in the primary system which represent the section of the reactor vessel wall from the top of the reactor vessel down to the top of the core. The steady-state calculation provides nominal operating temperatures for the reactor vessel wall, and during the transient, the changes in wall temperature are established and converted into a change in length. This change in length is included with the change in length of the control rod drivelines to determine the net motion of the control rods with respect to the core, and thus the net reactivity change. The details on generating the appropriate input data for this part of the model data are given in Appendix D.

c. Other Reactivity Feedback Models

There is some additional input required for completing the description of the reactivity feedback models. The reactivity effect from changes in length of the fuel and cladding in the metallic fuel core is calculated using the simple axial fuel expansion reactivity feedback model. This study uses the option where a force balance between the fuel and cladding is calculated to establish the net expansion of both materials. The physical properties on the strength and thermal expansion of the fuel and cladding are

input to the force balance calculation. The model also allows free relative expansion if the fuel and cladding are not in contact.

One option for calculating the axial expansion of the oxide fuel is the model DEFORM-4. The input data for using this model are reviewed in Appendix E. This model is only used for oxide fuel, and establishes both the conditions at the start of the transient based on a user-specified irradiation history, which can be channel dependent, and follows the fuel and cladding behavior in detail throughout the transient. DEFORM-4 is capable of providing more accurate radial and axial temperature profiles in the fuel, which in turn improves the accuracy of the reactivity feedback calculations from Doppler effects. The only word of caution that needs to be applied in this study is that DEFORM-4 cladding is assumed to be 316 stainless-steel, and the high temperature behavior of HT-9 can be substantially different from this, particularly in that HT-9 becomes weaker at lower temperatures and at any high temperature can be very much weaker than 316 stainless-steel. This difference is not important for nominal operating conditions, but can be during some of the transients. The net result would be that DEFORM-4 would not predict failure of 316 stainless-steel cladding, but the HT-9 cladding used in this study would already have failed.

B. Balance of Plant Modelling

The modelling of the balance of plant requires that an overall plant schematic be developed where each of the system components can be represented by an appropriate PRIMAR-4 element. Given the overall reactor design, the representation given in Fig. 5 was developed. As shown, the model schematic closely represents the actual reactor design. The reactor components are represented by various volumes and elements in the model, as listed in Tables IV and V. The elements are also grouped into flow segments for the fluid flow calculations and into temperature groups for the heat transfer calculations, as indicated in Table V. All large liquid volumes in the reactor are represented by compressible volumes (CV) in the PRIMAR-4 section of SAS4A. These volumes are connected by the various flow elements, as shown. The input data to support the PRIMAR-4 model are contained in input blocks 3, INPMR4, and 18, PMR4IN. A complete description of each of these blocks is given in

Fig. 5. Schematic of the PRIMAR-4 Model of the LSPB Reactor Plant.

TABLE IV. PRIMAR-4 Model of the LSPB Reactor System: Volumes

Volume	Compressible Volume No.
Core inlet plenum	1
Core outlet plenum/hot pool	2
Cold pool	3
Pump outlet plenum for pump group 1	4
Pump outlet plenum for pump group 2	5
Sodium volume between baffle and redan, hot side	6
Sodium volume between baffle and redan, cold side	7
Intermediate loop pump bowl	8
Pipe tee section	9
Intermediate loop sodium expansion tank	10
DRACS-I loop sodium tank	11
DRACS-II loop sodium tank	12

TABLE V. PRIMAR-4 Model of the LSPB Reactor System: Elements

Component	Liquid Flow Element	Liquid Flow Segment	Temperature Group
PRIMARY CIRCUIT			
Core	1	1	-
Radial Blankets:			
Lower section	2	2	1
Blanket section	3	2	2
Upper section	4	2	3
Radial Shields and Control Assemblies:			
Lower section	5	3	4
Blanket section	6	3	5
Upper section	7	3	6
Core Bypass Flow:			
Lower section	8	4	7
Blanket section	9	4	8
Upper section	10	4	9
IHX:			
Inlet section	11	5	10
Tube section	12	5	11
Outlet section	13	5	12
Pump Group 1:			
Pump inlet pipe	14	6	13
Pump	15	6	13
Pump outlet pipe	16	7	14
Pump outlet pipe	17	7	14
Pump outlet pipe	18	8	15
Pump outlet pipe	19	8	15
Pump Group 2:			
Pump inlet pipe	20	9	16
Pump	21	9	16
Pump outlet pipe	22	10	17
Pump outlet pipe	23	10	17
Pump outlet pipe	24	11	18
Pump outlet pipe	25	11	18
Outlet plenum to redan/ baffle volume flow path	26	12	19
Cold pool to redan/baffle volume flow path	27	13	20

TABLE V. PRIMAR-4 Model of the LSPB Reactor System: Elements (Cont'd)

Component	Liquid Flow Element	Liquid Flow Segment	Temperature Group
DRACS-I IHX:			
Inlet section	28	14	21
Shell section	29	14	22
Outlet section	30	14	23
DRACS-II IHX:			
Inlet section	31	15	24
Tube section	32	15	25
Outlet section	33	15	26
INTERMEDIATE HEAT TRANSPORT SYSTEM			
Pump	34	16	27
Horizontal pipe	35	16	27
Vertical downcomer	36	16	27
IHX shell side	37	16	11
Vertical outlet pipe	38	16	28
Horizontal pipe	39	16	28
Steam generator	40	16	29
Vertical pipe	41	16	30
Horizontal pipe	42	17	31
Sodium expansion line	43	18	32
DRACS-I			
Pipe	44	19	33
Air heat exchanger	45	19	34
Pipe	46	19	35
Vertical pipe	47	19	35
IHX tube side	48	19	22
Vertical pipe	49	19	36
Pipe	50	19	36
DRACS-II			
Pipe	51	20	37
Air heat exchanger	52	20	38
Pipe	53	20	39
Vertical pipe	54	20	39
IHX shell side	55	20	25
Vertical pipe	56	20	40
Pipe	57	20	40

Appendix D. Basically, INPMR4 covers the data governing the overall arrangement of all of the components in the model, while PMR4IN contains the associated geometrical, flow, and heat transfer data.

The development of the PRIMAR-4 model can be treated in several sections, one for the primary circuit, one for the main intermediate heat transport system, and one for the DRACS loops. In each section, the general outline of the model is given. The generation of the input data, along with any assumptions that may have been required is discussed in detail in Appendix D. In some cases, information is used which was not included in Ref. 11, and this is also included.

1. Primary Circuit

The primary circuit contains all of the sodium which flows through the core in the reactor vessel. The driver and internal blanket subassemblies in the core are modelled with the detailed channel models, as described in Section II.A. The radial blankets are modelled by a segment and using the PRIMAR-4 "bypass channel" for the heat generation region. All radial blankets are represented by an average radial blanket. The flow through the radial shields and control rod assemblies is modelled with another segment in PRIMAR-4, and another segment accounts for the leakage flow from the core inlet plenum to the outlet plenum. The flow from all of these segments exits into the core outlet plenum.

The sodium in the core outlet plenum flows into the four intermediate heat exchangers (IHXs) and two heat exchangers associated with the decay heat removal systems. Almost all of the flow is through the intermediate heat exchangers. The model assumes that all four are identical. The sodium flows out of the intermediate heat exchangers into the "cold" pool. The reactor has four primary pumps, all of them nominally identical. These are modelled as two pump groups in the present calculation. This configuration was selected to enable one pump to be isolated, in order to investigate the effect of a core inlet pipe break, with the other group representing the remaining three pumps. Each pump group includes the inlet pipe, the pump, the pump outlet plenum, and the pipes from the pump outlet plenum to the core inlet plenum.

2. Intermediate Loop

The reactor plant has four nominally identical intermediate loops which take heat from the primary system out to the steam generators. These are modelled as identical loops, represented by the one loop shown in Fig. 5. This loop contains all of the intermediate system piping, the shell side of the intermediate heat exchanger, and the sodium side of the steam generator. The intermediate loop pump and the sodium overflow tank are also included. The water side of the steam generator is not modelled, but a simplified treatment is used instead. For this study, it is assumed that the steam generators reject heat at their nominal full-power capacity during the transient overpower accident, and that there is no heat rejection at the steam generators for the unprotected loss-of-flow and loss-of-heat-sink accidents. The calculation of the associated input variables for the intermediate loop is discussed in Appendix D.

3. DRACS Loops

The LSPB plant design incorporates two independent and diverse decay heat removal systems. For the purposes of this discussion, they are identified as DRACS-I and DRACS-II. DRACS-I uses a heat exchanger within the outlet plenum, or "hot" pool, to take heat from the primary circuit sodium and remove it from the system through an air/sodium heat exchanger. The flow in the DRACS-I loop is assumed to be sodium, and is driven by natural circulation. The air flow through the heat exchanger is also natural circulation, with the flow rate controlled by adjustable dampers. For this study, it is assumed that this decay heat removal system is operating at its nominal capacity at the beginning of each transient. This was done for computational reasons, and does not have any effect on the transients until reactor power has been reduced to near decay heat levels.

The second decay heat removal system, DRACS-II, uses a similar sodium loop with an air/sodium heat exchanger. However, the heat exchanger in the reactor vessel is connected between the hot pool and the cold pool, as shown in Fig. 5. The DRACS-II loop is intended to operate only when the sodium in the reactor vessel expands and overflows the cylinder containing the

heat exchanger. As with DRACS-I, the DRACS-II loop operates on natural circulation. The DRACS-II loop is also assumed to be operating at capacity at the start of each transient. The details on generating the input data for these models is included in Appendix D.

III. RESULTS OF UNPROTECTED TRANSIENTS

The main purpose of this study is to determine the differences, if any, between the response of a large reactor with metallic fuel and a large reactor with oxide fuel to various unprotected transients. The SAS4A modelling described in Section II and Appendices A-G is designed to provide models of each reactor which are as identical as possible, so that any differences would be attributable only to the fuel type. For the first phase of the study, the effort is concentrated on analyzing the unprotected loss-of-flow (LOF), the unprotected transient overpower (TOP), and the unprotected loss-of-heat-sink (LOHS) accidents. For each accident scenario, the basic scenario for both fuel types is described, the response of each core design is discussed, and the results compared. All calculations were performed using the SAS4A computer code, modified as indicated in Appendix H.

A. Unprotected LOF

The unprotected loss-of-flow (LOF) accident is assumed to be initiated by a total loss of offsite power. This causes the electrical power to be lost to all primary pumps, intermediate loop pumps, and feedwater pumps. There is also a total failure to scram the reactor, so that the reactor power changes only in response to the reactivity feedbacks introduced as a result of the electrical power loss. The primary and intermediate loop pumps coast down according to their inertial characteristics only, which are assumed to provide an initial flow-halving time in both cores of 8 sec in the primary loop and 6 sec in the intermediate loop, as described in Appendix D. For this initial stage of the investigation, no attempt has been made to consider methods to lengthen the coastdown or otherwise improve the core response. Both cores are subjected to identical accident conditions. The loss of the feedwater pumps causes the loss of heat rejection capability at the steam generators. The DRACS loops are operating at their nominal capacity of 14 MW each.

1. Oxide Fuel Core

The calculated response of the oxide fuel core to the unprotected loss-of-flow accident is shown in Figs. 6 through 8, for the power and flow,

reactivities, and peak core temperatures, respectively. The calculations are done using DEFORM-4 for steady-state fuel characterization and transient fuel and cladding conditions. As Fig. 6 indicates, the change in core flow is accompanied by a change in power, with the flow dropping faster than the power. The power-to-flow ratio exceeds 2.0 at 15 sec after the start of the transient, and 3.0 at about 40 sec. The calculation is terminated shortly after 55 sec as the lead driver assemblies begin boiling in the core region.

The behavior of the reactor power is determined by the net reactivity, as shown by the plot of the net reactivity as a function of time during the transient, Fig. 7. At the time of boiling initiation, the net reactivity is approximately -0.25β . There is substantial negative reactivity feedback from radial expansion of the reactor core, but this is partially offset by the positive reactivity feedback from the increasing coolant temperature and the fuel Doppler.

During the first 10-15 sec, the average fuel temperature in each channel is increasing in response to the decrease in coolant flow rate. This increase in fuel temperature causes the Doppler feedback to be initially negative. However, the negative net reactivity rapidly decreases the power, which subsequently lowers the average fuel temperature resulting in positive Doppler feedback. After the first 10 sec of the transient, the average fuel temperature is generally increasing in the upper part of the core due to the reduced coolant flow, while it is decreasing in the lower part due to the lower power generation. The calculations indicate that the decrease in power is lowering the fuel temperature in the lower half of the core much faster than the lower flow rate is raising the fuel temperatures in the upper half, and significant positive Doppler feedback is generated, as shown in Fig. 7.

The axial coolant temperature profile at the initiation of boiling indicates that boiling would start near the core/upper axial blanket interface. At this stage of the study, the further progression of the accident has not been investigated to determine the spread of boiling in the assembly and likely fuel failure sites. It would be necessary to use a more

Fig. 6. Normalized Power and Flow as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Flow Accident with an Oxide Fuel Core.

Fig. 7. Reactivity Feedback as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Flow Accident with an Oxide Fuel Core.

CHANNEL 1 TEMPERATURE, DEG. C

Fig. 8. Peak Fuel, Cladding, and Coolant Temperatures, Core Inlet Temperature and Coolant Saturation Temperature in Channel 1 as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Flow Accident with an Oxide Fuel Core.

detailed channel arrangement in the core since the average driver channel, channel 2, includes many driver assemblies with high powers and power-to-flow ratios, as discussed in Section II and Appendix A. While the three-channel representation of the core is accurate for the conditions encountered in this calculation, a three-channel representation would not be for a further investigation into the gradual initiation of boiling in the remainder of the core.

2. Metallic Fuel Core

The same accident initiator is used for the metallic fuel core, with the results plotted in Figs. 9 through 11. The flow coastdown is essentially identical to that for the oxide core, as shown in Fig. 9. The response of the core power to the flow coastdown is also shown on Fig. 9. The power-to-flow ratio peaks at about 35 sec after the start of the transient with a value of 2.22. The peak coolant temperature in the channel reaches a maximum at about 42 seconds. The coolant temperature rises to 806°C at this stage of the accident, where the coolant saturation temperature is 952°C, for a margin to boiling of about 146°C. The plot of peak coolant temperature in Fig. 11 shows that the peak coolant temperature in the channel remains at about 806°C for the next 50 sec or so, as the high temperature coolant is swept out of the assembly. While the peak temperature is relatively constant, the axial node in which the peak temperature occurs is changing. At 60 sec after the start of the transient, the peak coolant temperature in the channel occurs above the top of the fission gas plenum, with a margin to boiling of 133°C, due to the lower saturation temperature in the upper part of the assembly.

The main reactivity feedback responsible for the behavior of the reactor power is from the radial expansion of the core, for at least the first 50 sec. During this time, the highest fuel and coolant temperatures are attained. After 50 sec, the feedback from control rod expansion becomes significant as well. The only substantial positive reactivity feedback is from the decrease in the coolant density as the coolant temperature rises. The other feedbacks are near zero for this stage of the transient.

Fig. 9. Normalized Power and Flow as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Flow Accident with a Metallic Fuel Core.

Fig. 10. Reactivity Feedback as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Flow Accident with a Metallic Fuel Core.

Fig. 11. Peak Fuel, Cladding and Coolant Temperatures, Core Inlet Temperature and Coolant Saturation Temperature in Channel 1 as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Flow Accident with a Metallic Fuel Core.

The fuel pin failure mechanism of cladding thinning and failure by eutectic material penetration is also of concern with metallic fuel. The threshold for the formation of the fuel/cladding alloy with the eutectic composition is around 725°C to 750°C. During the first 100 sec of this transient, this threshold temperature is exceeded in the lead driver assemblies, and in approximately half of the total driver assemblies. However, at the threshold temperature, penetration of the cladding is extremely slow. Rapid thinning of the cladding occurs around 200°C higher, which is approximately the same as the coolant saturation temperature. The penetration rate at 800°C is about 0.36 mm/hr, which means that the cladding would be breached in approximately 5600 sec. The peak fuel/cladding interface temperature of 792°C is reached by 38 sec, by 65 sec is already reduced to 773°C at the top of the core, and by 110 sec it is 725°C. The cladding damage would be minimal in this case, and should not impact the structural integrity of the fuel pins. The margin to rapid penetration is comparable to the margin to boiling.

3. Relative Performance

Using the same loss-of-flow accident initiator, different outcomes are obtained for the core with metallic fuel and the core with oxide fuel. The oxide fuel core is calculated to reach boiling at 55 sec into the transient, while the metallic fuel core calculations indicated a margin to boiling of around 146°C throughout the first 100 sec of the transient. The margin is rapidly increasing at 100 sec, indicating that the most serious near-term consequences have already been observed. Since the flow coastdown is identical for both cases, the difference in response can only be attributable to the power history during the transient, which implies that the source of the difference is in the reactivity feedbacks.

Comparing the results on Figs. 7 and 10, the main difference in the behavior of the reactivity is that for the metallic fuel, the net reactivity closely follows the radial core expansion reactivity feedback, while with oxide fuel, the net reactivity is only about half of the value from radial core expansion. While there is some difference in the feedback from expansion of the control rod driveline, the main cause appears to be the Doppler

component. In the metallic fuel, the Doppler remains negative for the first 100 sec, while with oxide fuel, the Doppler is initially negative but becomes rapidly positive after 15 to 20 sec. The difference in Doppler feedback is due to the relative temperature changes that the fuel is experiencing in each case, as a result of the different thermophysical properties for the two fuel types, and the smaller Doppler coefficient with metallic fuel.

The main differences in the thermophysical properties between metallic and oxide fuel are in the thermal conductivity of the fuel and in the thermal resistance across the boundary between the fuel and the cladding. The much higher thermal conductivity of metallic fuel, combined with the low thermal resistance across the fuel to cladding gap, leads to very much lower peak and average fuel temperatures. As an example, the peak fuel temperature with oxide fuel at steady-state is 1800°C, while with metallic fuel it is 722°C. This difference in the fuel temperatures and the smaller Doppler coefficient make Doppler feedback much less important in a reactor with a metallic fuel core. The relatively large difference in the Doppler coefficient is also inherent in the fuel type, since it is due to the presence of oxygen in the oxide fuel, which acts as a moderator lowering the energy of more neutrons into the resonance region. The hard spectrum for metallic fuel has fewer neutrons at this energy level.

The result of these differences in the Doppler reactivity feedback is that the smaller negative net reactivity in the oxide core causes the power to drop more slowly than in the metallic fuel core. For example, at 50 sec into the transient, the oxide core power is 36.8% of nominal, while the power in the metallic fuel core is 27.5% of nominal. Given that the flow has decreased to 12.6%, the power-to-flow ratio in the metallic core is approximately 2.18 and in the oxide core it is 3.00. At such a high, and still increasing, power-to-flow ratio in the oxide core, boiling occurs within a short time. Since the reason for this difference in the response of the core power is due to the thermophysical properties, such as the thermal conductivity, it would appear to be a fundamental difference between the two fuel types, not a design difference.

Overall, it appears that the differences in behavior of oxide fuel relative to metallic fuel is one of fundamental properties, given the current design, and that the behavior observed in these calculations would be typical of other large reactor designs where all other parameters are as identical as possible. However, many design changes which may improve the response for oxide fuel, such as lengthening the flow coastdown and reducing the pin size, have not been considered, although it is expected that such design changes would also have a beneficial effect for a reactor core using metallic fuel. The oxide fuel response could also be improved by reducing its specific power, at the cost of a higher fuel inventory.

B. Unprotected TOP

The accident initiator for the unprotected transient overpower (TOP) accident is assumed to be the uncontrolled withdrawal of one control rod, with failure to scram the reactor. The reactivity increase in the core causes the power to increase, raising fuel and coolant temperatures. The course of the accident is determined by the amount of reactivity added to the core, the reactivity feedbacks generated by the higher power and temperatures, and the capability of the remainder of the reactor system to handle the larger power generation.

1. Oxide Fuel Core

The response of the oxide fuel core to an unprotected withdrawal of one control rod is shown in Figs. 12 to 14. The worth of a single control rod withdrawal from nominal full power conditions is estimated at 0.10 \$, without accounting for any uncertainties [1]. The control rod insertion corresponding to this initiator is 10.2 cm at the beginning of an equilibrium cycle (BOEC). The control rods are nominally at the top of the core at the end of the equilibrium cycle (EOEC). A control rod speed of 3.81 mm/sec (9.0 in./min) is assumed, so that the control rod withdrawal from nominal full power conditions requires almost 27 sec to complete. The average reactivity insertion rate is then 0.0037 \$/sec, which was increased to 0.005 \$/sec as an upper bound for the calculations. The fuel and cladding behavior are calculated with DEFORM-4 for this accident.

Fig. 12. Normalized Power and Flow as a Function of Time from the Start of the Transient for an Unprotected Transient Overpower Accident with an Oxide Fuel Core and a Control Rod Initiator of 0.10 \$.

Fig. 13. Reactivity Feedback as a Function of Time from the Start of the Transient for an Unprotected Transient Overpower Accident with an Oxide Fuel Core and a Control Rod Initiator of 0.10 \$.

CHANNEL 1 TEMPERATURE, DEG. C

Fig. 14. Peak Fuel, Cladding, and Coolant Temperatures, Core Inlet Temperature and Coolant Saturation Temperature in Channel 1 as a Function of Time from the Start of the Transient for an Unprotected Transient Overpower Accident with an Oxide Fuel Core and a Control Rod Initiator of 0.10 \$.

Using a reactivity insertion rate of 0.005 $\$/\text{sec}$, the reactivity ramp terminated at 20 sec for the 0.10 $\$$ initiator, as shown in Fig. 13 where the reactivity added by the withdrawal of the control rod is identified as 'programmed' reactivity. The result of the reactivity insertion is an increase in the core power to 107.2% at 20 sec. The power slowly drops after the termination of the reactivity insertion in response to the steam generator modelling. Since the steam generators are assumed to have a constant heat rejection rate of nominal power, an increase in the core power will cause a heating of the primary and intermediate loops until the reactivity feedbacks in the core are sufficient to bring the power generation back to nominal full power, balancing the heat rejection rate. Once more is known about the steam generator design, it is possible that the steam generator behavior would be different. It is quite likely that the steam generators could handle 107.2% of nominal power without any change in the core inlet temperature, since this amount of performance margin would normally be present in the design.

The main reactivity feedback counteracting the reactivity insertion from the control rod withdrawal is the negative Doppler due to the increased power and fuel temperatures. As would be expected from such a small reactivity insertion, the consequences are not severe, but are rather more like those for an operational transient. The peak coolant temperature in the lead driver channel is 569°C, an increase of 20°C from the steady-state conditions. The margins to all failure criteria for the oxide core are large throughout the transient.

2. Metallic Fuel Core

The response of the metallic fuel core design to the unprotected withdrawal of one control rod with a total reactivity addition of 0.11 $\$$ [1] is shown in Figs. 15 through 17. The rod insertion corresponding to this amount of reactivity is 9.14 cm at the end of an equilibrium cycle (EOEC). In contrast to the oxide fuel core, the control rods are nominally at the top of the core at BOEC, and the reactor gains reactivity during the cycle. This accounts for the larger initiator in the metallic fuel core. The nominal rate of 0.005 $\$/\text{sec}$ is used in this case as well, where the reactivity ramp terminates at 22 sec into the transient.

Fig. 15. Normalized Power and Flow as a Function of Time from the Start of the Transient for an Unprotected Transient Overpower Accident with a Metallic Fuel Core and a Control Rod Initiator of 0.11 \$.

Fig. 16. Reactivity Feedback as a Function of Time from the Start of the Transient for an Unprotected Transient Overpower Accident with a Metallic Fuel Core and a Control Rod Initiator of 0.11 \$.

CHANNEL 1 TEMPERATURE, DEG. C

Fig. 17. Peak Fuel, Cladding, and Coolant Temperatures, Core Inlet Temperature and Coolant Saturation Temperature in Channel 1 as a Function of Time from the Start of the Transient for an Unprotected Transient Overpower Accident with a Metallic Fuel Core and a Control Rod Initiator of 0.11 \$.

The result of the reactivity insertion is that the power is calculated to reach a peak of 117.0% of nominal power at 30 sec. As in the case with oxide fuel, the steam generators continue to reject heat at nominal full power conditions, with the result that the primary and intermediate system temperatures increase until the reactivity feedbacks in the core are sufficient to bring the core back to nominal power. There is not a dominant negative reactivity feedback with metallic fuel to counteract the positive reactivity from the control rod withdrawal as Fig. 16 demonstrates. The two most important feedbacks appear to be control rod driveline expansion and core radial expansion. The reactivity feedback from Doppler is only a small contributor.

The change in peak core temperatures is shown in Fig. 17. The initial increase in all temperatures is followed by a period of almost constant value. As for the loss-of-flow accident, the failure criterion of most importance with metallic fuel is probably the fuel/clad interface temperature. At the peak in the core power, the maximum fuel/clad interface temperature is 600°C, which is an increase of almost 36°C from steady-state. This temperature is about 125°C below the threshold for formation of liquid fuel/clad alloy with a eutectic composition and at least 315°C below the threshold for rapid penetration. As a result the overall consequences of the accident are minimal, with only a slight heating of the primary and intermediate systems.

There is a possibility that the steam generator may have difficulty with continued operation at this power level. The peak of 117.0% of nominal capacity is close to the nominal 20% excess capacity that has been suggested for such steam generators. The details of the design could have a large impact on such an accident, for if the reactivity insertion were larger, or the excess steam generator capacity were less than 20%, it is possible that the unprotected rod withdrawal would lead directly to a loss-of-heat-sink at the steam generators.

3. Relative Performance

There is some difference in the response to this accident for the oxide and metallic cores, but the difference is not as large as for the

unprotected loss-of-flow accident. The initiator for the metallic fuel is slightly larger than that for the oxide fuel, yet there is a substantial difference in the peak power at the end of the reactivity insertion (107.2% for oxide and 117.0% for metal). The main reason for this appears to be the large difference in the Doppler feedback. The Doppler feedback is a fast-responding term in the overall feedback, and it is much larger in the oxide fuel due to the larger coefficient and the higher temperatures. It is interesting to note that this component of the feedback which appears to be so detrimental to the oxide core response in the unprotected loss-of-flow accident now results in better response for an unprotected control rod withdrawal.

There is a large margin to all of the failure criteria for the oxide core design. Probably the smallest margin exists for the metallic fuel core and formation of low-melting-point fuel/clad alloys at the fuel/clad interface. However, even this margin is 125°C to the formation threshold and 315°C to the rapid penetration threshold for these materials. Both core designs experience minor changes in conditions as a result of the nominal initiators for this accident. The situation may be different if uncertainties were included in the accident initiator.

There are several sources of uncertainty which could lead to a substantial increase in the reactivity insertion resulting from the withdrawal of one control rod. The uncertainty in core reactivity arising from uncertainty in the neutronics calculations and in the manufacture of the core components usually causes the core to be designed with additional reactivity. Such additional reactivity could lead directly to an increase in the size of the initiator, depending on the state of the core when it is loaded. For either core, a substantial increase in the TOP initiator (to 0.25\$ or 0.35\$, for example) greatly increases the likelihood of losing the steam generator heat rejection capability. The change in power in the oxide fuel core in response to an unprotected control rod withdrawal is smaller than the change in power for the metallic fuel core, and the oxide core could probably withstand a somewhat higher initiator without overpowering the steam generator heat rejection capability. The metallic fuel core would not be able to tolerate much of an increase in the initiator without losing the steam generator heat-sink. The conclusion is that the oxide fuel design has a better response to initiators which are of the size of 0.10\$, with larger

margins to the appropriate failure criteria. However, the metallic fuel core still has a margin of at least 125°C to the eutectic formation threshold. For larger initiators, the safety margins will be determined by the response to the loss of the steam generator heat sink.

C. Unprotected LOHS

The unprotected loss-of-heat-sink (LOHS) accident is assumed to start with the loss of heat rejection capability at all of the steam generators, with the primary and intermediate loop pumps continuing to run. There is also a failure to scram the reactor, so that the reactor power changes only in response to the reactivity feedbacks generated in the core. This accident tends to be a longer-term accident because the accident is terminated by raising the system temperatures to a point where only decay heat is being generated in the core and the decay heat generation rate is balanced by the capacity of the decay heat removal systems.

1. Oxide Fuel Core

The response of the oxide fuel core to the loss of heat rejection capability at the steam generators is shown in Figs. 18 through 20 for the power and flows, reactivity, and temperatures, respectively. Simple models are used for fuel and cladding behavior in this case instead of DEFORM-4 due to the length of the transient. The calculations indicate that about 30 sec are required for the loss of the steam generator heat rejection capability to have an effect on the primary system temperatures and the core power. After 30 sec, the change in core power is caused solely by the rise in core inlet temperature which accompanies the loss of the primary heat sink. The power begins to decrease as the net reactivity of the core becomes negative in response to the rising inlet temperature. The main component of the reactivity feedback is the radial expansion of the core, through expansion of the grid plate and the subassembly load pads. The other large term is from the control rod driveline expansion, which is less than half as large as the core radial expansion feedback. The largest positive component is from the decreasing density of the sodium coolant, due to its increasing temperature.

Fig. 18. Normalized Power and Flow as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Heat-Sink Accident with an Oxide Fuel Core.

Fig. 19. Reactivity Feedback as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Heat-Sink Accident with an Oxide Fuel Core.

CHANNEL 1 TEMPERATURE, DEG. C

Fig. 20. Peak Fuel, Cladding, and Coolant Temperatures, Core Inlet Temperature and Coolant Saturation Temperature in Channel 1 as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Heat-Sink Accident with an Oxide Fuel Core.

The power continues to drop for the first 850 sec of the transient as shown in Figs. 18 through 20. At 850 sec, the power is near 20% of nominal, with the peak coolant temperatures approaching 725°C. There is a significant change in the course of the transient beyond about 850 sec, as indicated by the change in slope of the power history. The source of the change is in the reactivity feedback from radial expansion of the reactor core, as seen in Fig. 19. The reason for this change is contained in the details of the subassembly loading, as discussed in Section II.A.3 and Appendix C. Initially, the top load pad of the subassembly is in contact with the top restraint ring through the top load pads of the outer assemblies, and the core is pushed tightly together at the above-core load pads. The assemblies bend outward in the core region in response to this loading. As the power is decreasing during the first 850 sec of the accident, the subassembly straightens, since the thermal bending of the subassembly is proportional to the core power, lessening the forces generated at the load pads and the restraint ring. At a certain combination of power and temperature, the thermal bending of the assembly is just sufficient to fit in the space provided by the restraint ring, outer assemblies, and core assemblies. As the power drops further, the subassembly is not tightly held, but is only loosely pushed together at the above-core load pads.

When the assembly is tightly held, as it is for the first 850 sec, a reduction in power causes the subassembly to straighten, reducing the size of the core and creating a positive reactivity effect. The increasing above-core load pad temperature during the first 850 sec is acting to expand the core, a negative reactivity effect, so that these two components of the reactivity feedback for radial core expansion are of opposite sign. After the change in subassembly loading around 850 sec, the further straightening of the assembly in response to the reduction in power tends to increase the size of the core, so that the two components of the reactivity feedback from radial core expansion both provide a negative reactivity feedback. This is the reason that the reactivity feedback from radial expansion of the core experiences a change in slope around 850 sec, giving a larger negative reactivity effect for a given change in power and temperature.

The net reactivity also reflects this change in the reactivity feedback from radial core expansion, causing the power to drop faster after 850 sec, as observed in Fig. 18. The power continues to drop smoothly until about 1750 sec, when another change occurs in the reactivity feedback from radial core expansion. This change in the reactivity feedback from radial core expansion is caused by the further straightening of the subassembly in response to the drop in core power and the thermal expansion of the grid plate. The subassemblies are pushed together at the top load pad region instead of at the above-core load pad region, introducing gaps between the above-core load pads. This introduces significant further expansion of the core region, but since this is occurring at a very low fission power level, the increase in the negative net reactivity has little effect on the course of the transient.

The overall result is that the reactor power is reduced to near decay heat levels by about 3000 sec after the start of the transient, with fission power being only 0.1% of nominal power. The temperature gradients in the pin region have also collapsed, as shown in Fig. 20, with the coolant and cladding temperatures being about 793°C, while the peak fuel centerline temperatures are 815°C. The corresponding margin to coolant boiling in the fuel region of the subassembly is around 260°C at the top of the core. The total reactor power is 55 MWth at 3000 sec, and the DRACS loops are rejecting heat at 45 MWth, so while there is a mismatch in the power generation and the heat rejection rate, it is small. The system temperatures should not rise much further before a gradual cooling of the entire system occurs.

2. Metallic Fuel Core

The response of the metallic fuel core design to a total loss of the steam generators is shown in Fig. 21 through 23. The effect of the loss of heat rejection capability at the steam generators is calculated to take about 30 sec to be propagated to the core region. The subsequent rise in the core inlet temperature generates negative reactivity feedback, reducing the core power. As Fig. 21 shows, the power is down to decay heat levels at about 1200 sec after the start of the transient.

Fig. 21. Normalized Power and Flow as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Heat-Sink Accident with a Metallic Fuel Core.

Fig. 22. Reactivity Feedback as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Heat-Sink Accident with a Metallic Fuel Core.

CHANNEL 1 TEMPERATURE, DEG. C

Fig. 23. Peak Fuel, Cladding, and Coolant Temperatures, Core Inlet Temperature and Coolant Saturation Temperature in Channel 1 as a Function of Time from the Start of the Transient for an Unprotected Loss-of-Heat-Sink Accident with a Metallic Fuel Core.

The reactivity changes which cause this change in power are shown in Fig. 22 for all of the components of the net reactivity feedback. The only positive reactivity feedback is from the decrease in coolant density associated with the higher core temperatures. This positive reactivity is balanced by negative reactivity feedback from radial core expansion, Doppler, control rod driveline expansion and axial fuel expansion. The negative Doppler feedback and the axial expansion of the fuel arise from the increase in average fuel temperature associated with the rise in coolant temperature, as shown in Fig. 23, even though the core power is dropping. The negative reactivity feedback from radial core expansion is due to the thermal expansion of the core in response to the higher temperatures. At approximately 480 sec after the start of the transient, a change in the subassembly loading occurs as the power drops and the subassemblies straighten. This is the same change in loading conditions that occurs around 850 sec with the oxide fuel core. The loading change results in a larger negative rate of change of the reactivity feedback from radial core expansion with changes in core power and temperatures for the same reasons described for the oxide core. The second loading change occurs at about 1175 sec after the start of the transient. This is the same loading change that occurs at 1750 sec in the oxide core, providing even more negative feedback resulting in a long-term neutronic shutdown. At 1500 sec after the start of the transient, the net reactivity is substantially negative, with reactor fission power essentially gone. The accident then becomes a balance between decay heat generation and DRACS capacity to terminate the temperature rise.

The coolant temperature in the core region reaches 576.3°C by 1000 sec, with essentially no radial or axial temperature gradient in the fuel pin. The peak core temperatures rise very slowly after 1000 sec, reaching 577.6°C by 3000 sec after the start of the transient. The fuel/clad interface temperature is below 580°C, which is at least 150°C below the threshold for formation of fuel/clad alloys with the eutectic composition, and 350°C below that required for any rapid penetration of the cladding.

3. Relative Performance

The results indicate that failure thresholds such as coolant boiling and cladding failure by eutectic penetration are not reached with either core

design. The margin to boiling is at least 260°C in the oxide core and over 500°C with the metallic core. The margin to rapid eutectic penetration with the metallic fuel core is over 350°C. However, the overall higher system temperatures for the oxide fuel design may have some negative consequences with regards to the allowable temperature limits for the reactor structural materials.

For the metallic fuel core, the system temperature at 3000 sec is 577°C, an increase of 23°C from steady-state, which is below the level where the structural materials in the reactor would sustain any damage. The oxide fuel core reaches much higher system temperatures, mostly in response to the positive Doppler feedback and axial contraction of the fuel. The long-term temperatures around 793°C in the oxide fuel design slightly over the limit for service level D, 787°C, which means significant damage would have been done to some components. In addition the long-term behavior of the HT-9 cladding at these temperatures is not well known, with there being a potential for significant creep and possible fuel pin failure.

IV. SUMMARY

Oxide fuel and metallic fuel core designs for a large reactor, 3500 MWth, have been examined for their relative merits in response to unprotected loss-of-flow, transient overpower, and loss-of-heat-sink accidents. The core designs were designed to be interchangeable in the LSPB plant, with the same overall geometry. Both cores were optimized to provide the best balance between a low peaking factor and a low burnup control swing. In support of this investigation, a complete description of the models used with the SAS4A accident analysis computer code has been given. This includes details on the core modelling, with SAS4A channel assignments, geometry, and fuel properties, as well as details on the remainder of the primary and intermediate systems. Both decay heat removal systems are also described and modelled.

The SAS4A computer code was used to provide best-estimate calculations of the response of each core design to hypothetical unprotected loss-of-flow, transient overpower, and loss-of-heat-sink accidents. Both reactor cores were subjected to identical accident conditions, and no attempt was made in this stage of the study to investigate methods for improving the response of either core. The results of the calculations were analyzed and compared with the following findings.

Unprotected Loss-of-Flow Accident

The oxide core experienced boiling 55 sec after the start of an inertial coastdown of the primary and intermediate loop pumps without scram in the reactor. The peak coolant temperature in the hottest driver assembly of the metallic fuel core attained a value of 806°C, which indicates a margin to boiling of 146°C. The peak fuel/cladding interface temperature of 792°C in this assembly is about 65°C over the threshold for formation of a liquid fuel/clad alloy with the eutectic composition. The rate of formation is extremely slow at these temperatures, however, and there is a margin of at least 135°C to the threshold for rapid formation of liquid eutectic material at the fuel/cladding interface. The main reason for the difference in core response appears to be the large difference in Doppler reactivity feedback, which is slightly negative in the metallic fuel case and large and positive

with oxide fuel. The large difference in Doppler reactivity feedback is caused by the much higher steady-state fuel temperatures in the oxide fuel core and the larger Doppler coefficient with an oxide fuel.

Unprotected Transient Overpower Accident

The oxide fuel core experienced a rise to 107.2% of nominal power as a consequence of a 0.10 \$ reactivity insertion. The metallic fuel core rose to 117.0% of nominal power as a result of a 0.11 \$ reactivity insertion. Since the reactivity insertions were small, there were large margins to all of the failure thresholds for each core design. The smallest margin of 125°C is for the threshold for formation of a liquid fuel/cladding alloy with the eutectic composition at the fuel/cladding interface for metallic fuel in the hottest driver assembly, although the interface temperature has only increased 36°C from steady-state. A comparison of the peak power achieved in each case shows that the increase in power for the metallic core was more than twice the increase for the oxide fuel design. The reason for the difference in power is that the oxide fuel has a substantial Doppler reactivity feedback, which is a fast-acting and negative reactivity feedback. The much lower Doppler feedback with a metallic fuel core, which was an asset for the consequences of an unprotected loss-of-flow accident, results in a less favorable response for this unprotected transient overpower accident. However, for reactivity insertions of the magnitude of 0.10 \$, the consequences are not serious for either core design.

The results for a larger reactivity insertion could be quite different. For an increase in the reactivity added by the withdrawal of one control rod, to 0.25 \$ or 0.35 \$ as may be required to account for calculational uncertainties in the core design, it is not certain that an unprotected overpower transient would result in an increase in power followed by a slow decrease back to nominal power. For such large accident initiators, the core power could be high enough to cause the steam generators to be isolated from the reactor to protect the remainder of the plant. For this situation the outcome of the accident would depend on the consequences of a loss-of-heat-sink accident as well.

Unprotected Loss-of-Heat-Sink Accident

The oxide fuel core response to a loss of the heat rejection capability at the steam generators is to raise the system temperatures to an almost uniform level of just under 793°C. This is sufficient to reduce the fission power in the reactor to near zero, so that the remainder of the transient would be a balance between decay heat and the capacity of the decay heat removal systems. For the oxide core, a temperature of 793°C appears to be a reasonable long-term limit for this accident, given the results of the calculations after 3000 sec from the start of the transient. There is a margin of almost 260°C to coolant boiling in the core region for this accident. However, a long-term temperature of 793°C is certain to damage some reactor vessel components, as an estimated temperature of 787°C defines the service level temperature above which gross deformations of the reactor vessel and components would occur and serious damage would result. The HT-9 cladding of the fuel pins is also subject to damage and failure at these temperatures.

The metallic fuel core response is to raise the system temperatures to an almost uniform level of just under 580°C which is sufficient to reduce the fission power to zero. A temperature of 580°C provides a margin of at least 500°C to coolant boiling in the core region, a margin of 150°C to eutectic formation at the fuel/cladding interface, and a margin of 350°C to rapid penetration of the cladding. The main reason that the system temperature is much lower with a metallic fuel core appears to be the higher fuel temperatures at steady-state for the oxide core. The reduction of the fission power introduces substantial positive reactivity feedback from Doppler and contraction of the fuel in the oxide core, while the average fuel temperature in the metallic fuel core increases during the unprotected loss-of-heat-sink accident, with the Doppler and axial fuel expansion providing negative reactivity feedback.

The conclusions from this study can be summarized as follows:

1. The metallic fuel core survives the unprotected loss-of-flow accident without coolant boiling, while the oxide core boils.

2. The oxide fuel core experiences a smaller increase in power and lower temperatures than the metallic fuel core for an unprotected transient overpower accident with a reactivity addition on the order of 0.10 \$.
3. The metallic fuel core peak temperatures for an unprotected loss-of-heat-sink accident are much lower than for the oxide core under the same accident conditions, with the system temperatures for an oxide fuel core being high enough to cause component and fuel pin failures.
4. A transient overpower accident with a larger initiator, on the order of 0.25 \$ to 0.35 \$, is expected to cause a subsequent loss-of-heat-sink where the results will be similar to the results for the unprotected loss-of-heat-sink without any reactivity addition.
5. The main reason for the difference in response for the oxide and metallic fuel cores appears to be the higher steady-state fuel temperatures and larger Doppler coefficient in the oxide core. The higher fuel temperatures are a direct result of the much lower thermal conductivity of the oxide fuel and the much lower thermal conductance for oxide fuel at the fuel/cladding boundary.

The overall conclusion is that there is a difference in the response of oxide fuel and metallic fuel cores to various unprotected accidents for a 3500 MWth reactor. The difference in response is basically the result of differences in the properties of the fuel types, given that the two core designs were as comparable as possible. Whether it would be possible to reduce the difference in core response to the unprotected accidents by making design changes to the oxide core is an unanswered question. In the absence of any special design features for either core, it appears that the benefits of metallic fuel in responding to unprotected loss-of-flow, transient overpower, and loss-of-heat-sink accidents which were observed for small and medium size reactor designs are effective for large reactor designs.

V. REFERENCES

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APPENDIX A. SAS4A MODELLING OF THE OXIDE FUEL CORE

The modelling of the oxide fuel core is discussed in detail in this appendix. The first topic covered is the combining of the subassemblies into appropriate SAS4A [A1] channels, including the rationale for selecting the combination used for this study. The generation of the SAS4A input data deck is covered next. This is done block-by-block for every input block. The basis for each value is given, along with any calculations required to arrive at the input value. Background information which was not included in Ref. 5 of the main report is also provided. This was only necessary for the development of the input for PRIMAR-4, which models the reactor balance of plant.

I. Channel Modelling of the Core

The neutronics analysis for the interchangeable metal and oxide core designs specified each assembly uniquely [A2]. The assembly assignment is shown in Fig. A-1 for the oxide fuel core and represents 63 different individual assembly locations. In order to model this core with SAS4A, the assemblies must be grouped into a relatively small number of channels. The grouping of the subassemblies into channels is done according to the following arrangement:

- Channel 1. Assembly with the highest average power-to-flow ratio with a correspondingly high average power.
- Channel 2. All remaining driver assemblies.
- Channel 3. All remaining internal blanket assemblies.

The reasons for this arrangement are briefly discussed in Section II.A.1. The radial blanket assemblies are each treated as a "bypass" channel in PRIMAR-4, as are the control rod assemblies and shield assemblies, and true core bypass flow. This is covered in detail in Appendix D.

A detailed description of the assembly assignment number, orifice zone, relative end of equilibrium cycle (EOEC) average power per assembly, relative flow per assembly, and relative power-to-flow ratio is given in Table A-1 for the oxide core. For the oxide core, two assemblies are good candidates. Assembly 7 has the highest average power per assembly while assembly 4 (which also has a high average power per assembly) has the highest power-to-flow ratio. Assembly 4 is selected as the most appropriate assembly for the hottest channel for the SAS4A analyses. The remaining assemblies are grouped as indicated in Table A-1. The neutronics information [A2], including fuel and cladding worths, coolant worths, and Doppler, is averaged for each channel to form the neutronics characteristics for each channel, where the average is weighted based on the number of each assembly in the core. The results are listed in Appendix F. The average coolant flow rate is calculated in the same manner for each channel.

Fig. A-1. Assembly Assignment for the 3500 MWth Oxide Fuel Heterogeneous Core.

TABLE A-1. SAS4A Channel/Assembly Parameters at EOE for Oxide Core

Asm Num	Asm Type	SAS Channel	Orifice Zone	Total Asm	Pwr/Asm (Rel)	Flow/Asm (Rel)	Pwr/Flow (Rel)
1	Dr	2	1	6	1.53663	1.35266	1.136
3	Dr	2	1	6	1.58921	1.35266	1.175
6	Dr	2	1	6	1.59406	1.35266	1.178
7	Dr	2	1	12	1.60333	1.35266	1.185
8	Dr	2	1	12	1.55478	1.35266	1.149
9	Dr	2	1	12	1.53953	1.35266	1.138
10	Dr	2	1	12	1.56282	1.35266	1.155
2	Dr	2	2	12	1.45672	1.28417	1.134
4	Dr	1	2	12	1.56065	1.28417	1.215
5	Dr	2	2	6	1.53717	1.28417	1.197
11	Dr	2	2	12	1.54668	1.28417	1.204
12	Dr	2	2	6	1.44976	1.28417	1.129
13	Dr	2	2	12	1.43736	1.28417	1.119
16	Dr	2	2	12	1.48436	1.28417	1.156
17	Dr	2	2	6	1.37926	1.28417	1.074
18	Dr	2	2	12	1.39788	1.28417	1.089
19	Dr	2	2	12	1.40707	1.28417	1.096
15	Dr	2	3	12	1.44650	1.24062	1.166
20	Dr	2	3	6	1.25872	1.24062	1.015
21	Dr	2	3	12	1.31941	1.24062	1.064
23	Dr	2	3	12	1.35067	1.24062	1.089
14	Dr	2	4	12	1.41466	1.19707	1.182
22	Dr	2	4	12	1.29917	1.19707	1.085
25	Dr	2	4	12	1.16006	1.19707	0.969
26	Dr	2	4	12	1.22483	1.19707	1.023
28	Dr	2	4	12	1.25007	1.19707	1.044
24	Dr	2	5	6	1.03880	1.14718	0.906
27	Dr	2	5	12	1.23331	1.14718	1.075
30	Dr	2	5	12	1.00762	1.14718	0.878
31	Dr	2	5	12	1.09191	1.14718	0.952
32	Dr	2	5	12	1.10096	1.14718	0.960
33	Dr	2	5	12	1.12113	1.14718	0.977
29	Dr	2	6	12	0.85830	0.93444	0.919
34	Dr	2	6	12	0.77859	0.93444	0.833
35	Dr	2	6	12	0.84731	0.93444	0.907
36	Dr	2	6	12	0.84910	0.93444	0.909
37	Dr	2	6	12	0.85958	0.93444	0.920
38	Dr	2	6	6	0.88143	0.93444	0.943

TABLE A-1. SAS4A Channel/Assembly Parameters at EOE for Oxide Core (Cont'd)

Asm Num	Asm Type	SAS Channel	Orifice Zone	Total Asm	Pwr/Asm (Rel)	Flow/Asm (Rel)	Pwr/Flow (Rel)
43	IB	3	7	12	0.87386	0.92264	0.947
44	IB	3	7	12	0.83441	0.92264	0.904
45	IB	3	7	6	0.84400	0.92264	0.915
46	IB	3	7	6	0.81218	0.92264	0.880
48	IB	3	7	12	0.80129	0.92264	0.868
49	IB	3	7	12	0.82069	0.92264	0.890
41	IB	3	8	6	0.60102	0.82412	0.729
42	IB	3	8	6	0.55032	0.82412	0.668
47	IB	3	8	12	0.78789	0.82412	0.956
50	IB	3	8	12	0.72386	0.82412	0.878
51	IB	3	8	12	0.71611	0.82412	0.869
52	IB	3	8	12	0.69360	0.82412	0.842
53	IB	3	8	6	0.76233	0.82412	0.925
54	IB	3	8	12	0.68270	0.82412	0.828
55	IB	3	8	12	0.50557	0.82412	0.613
40	IB	3	9	6	0.43347	0.44898	0.965
56	RB	BP	9	6	0.29481	0.44898	0.657
58	RB	BP	9	12	0.27876	0.44898	0.621
39	IB	3	10	1	0.38020	0.37146	1.024
57	RB	BP	10	12	0.20401	0.37146	0.549
59	RB	BP	10	12	0.17522	0.37146	0.472
60	RB	BP	10	12	0.21259	0.37146	0.572
61	RB	BP	10	12	0.22751	0.37146	0.613
62	RB	BP	10	12	0.23160	0.37146	0.623
63	RB	BP	10	12	0.23604	0.37146	0.635

II. General Integer Input and Options

The integer input for the core models is contained in two input blocks. Block 1 contains the integer input which applies on a core-wide basis, while Block 51 integer input data applies to each channel. General input for running the codes is contained in Block 11. The selection of model options is discussed, along with the interactions with other input data blocks. The final result is listed in Appendix F which contains the oxide fuel core input data decks.

A. Block 1, INPCOM

Block 1 contains general integer input data for the SAS4A code and models. It includes time step controls for printing, restart files and problem time. Many selection flags for model options are also set in this input block. Only nonzero values are discussed unless a value of zero has a particular meaning.

Location, Input Variable

#1	NCHAN	The number of channels used to represent the core is 3, as discussed in Section I. NCHAN = 3.
#3	IFUEL1	The number of fuel types used in the channel models. This is used to set the number of fuel property tables in Block 13. For this study, IFUEL1 = 5.
#4	ICLAD1	The number of cladding types used in the channel models, for determining the number of cladding property tables required in Block 13. For this study, ICLAD1 = 1.
#5	IPLUP	This flag selects the location of the fission gas plenum, which for the oxide core is below the core region. The appropriate value is then IPLUP = 1.
#8	IPOWER	This flag selects the option of specifying a table of reactivity as a function of time to be input to the calculation. This table is in Block 12, and is used mainly for the reactivity insertion in a transient overpower accident. IPOWER = 0.
#11	MAXSTP	The maximum number of main time steps for the transient calculation. This is rather arbitrary, selected by the user depending on the length of transient desired. It is also possible to use maximum problem time, Block 11, Loc. 7, for the same purpose, in which case MAXSTP is set to a large value. If only a steady-state calculation is to be done, the value of MAXSTP is 0. For the initial calculations reviewed in this report, several values of MAXSTP were used, depending on the accident:

Accident	MAXSTP
LOF	400
TOP	800
LOHS	3000

This input value must be reset for running continuations of a transient calculation, if it is used to control the length of the calculation.

#12 IPO

This integer selects the frequency of printing the full printout, based on the time step number, and is arbitrary. For this study, the following values were used:

Accident	IPO
LOF	20
TOP	50
LOHS	50

#13 IPOBOI

The same as Loc. 12, but for printing after the initiation of boiling. The values in Loc. 12 are repeated for this study.

#15 NSTEP

This integer selects the frequency of writing a restart file on logical unit 17, and is arbitrary. If a restart file is desired, the file on logical unit 17 must be created by either making a new file or overwriting an existing file during job execution. For the present study, a value of every 100 main time steps is used. NSTEP = 100.

#16 NDELAY

The number of delayed neutron precursor families is 6, as obtained from Ref. 6 and used in Block 12 in several locations. NDELAY = 6.

#17 NDKGRP

The number of decay heat groups is 6, as used in Block 12. NDKGRP = 6.

#18 NPREAT

The number of entries in the table of external reactivity, as a function of time, PREATB Block 12, Loc. 29-48 and Loc. 49-68. This is normally used for transient overpower accidents; otherwise the value is 0.

#22 NTOTAB

The number of entries in the table of core inlet temperature as a function of time. Since PRIMAR-4 is being used to represent the reactor system, only 1 value is needed, usually the nominal steady-state core inlet temperature. The table TOTAB is in Block 14, starting with location 45. NTOTAB = 1.

#23 MULSTR

The value of zero selects the option for saving only 1 restart file, which is the latest one written. The other option can also be used, but it may lead to large total file size. MULSTR = 0.

#24	ICLCMP	A non-zero integer writes plotting information to a plot file for post-processing. The file is identified as logical unit 11, which must be created or overwritten during job execution. ICLCMP = 11.
#25	INEDIT	The detailed description of the input can be written in an edited format, if desired. This is usually not done due to the size of the output. INEDIT = 0.
#26	IYLD	This value is used by DEFORM-4, and is discussed in Appendix E. When DEFORM-4 is not used, the value is ignored.
#27	IPRION	This integer selects the modelling used for the reactor system, which in this case is PRIMAR-4. IPRION = 4.
#28	IMELTV	This value is used by DEFORM-4, and is discussed in Appendix E. When DEFORM-4 is not used, the value is ignored.
#31	ICREXP	This integer activates the model for calculating the reactivity feedback from control rod driveline expansion and controls the printing of the results. For this study, ICREXP = 2, which calculates this feedback and prints the result every transient step.
#36	IRADEX	This integer activates the models for calculating the reactivity feedback from radial core expansion and selects the model to be used. In this study, the "detailed" radial core expansion model is used, with normal printout, and the temperature of the inlet plenum wall is used to determine the temperature of the core support, or grid, plate. IRADEX = -4. Also see Appendix C.
#39	NFUELD	The number of dollars of fuel reactivity worth which has to be reached to terminate the calculation is set to -5, which is the current recommended value. NFUELD = -5.
#45	NPOWDK	The number of decay heat curves used in the modelling of the core. In this study, each channel is assumed to have the same decay heat curve. NPOWDK = 1.
#46	NPKST	The number of time periods used for initializing the decay heat curves is 1, with the time set in Block 12. NPKST = 1.
#51	NSUBTC	NSUBTC = 595; see Appendix C.
#52	MTGRD	MTGRD = 1; see Appendix C.
#53	MTACLP	MTACLP = 2; see Appendix C.
#54	MTTLP	MTTLP = 2; see Appendix C.

#55	MODEEX	This integer selects the model used for calculating the reactivity feedback from axial expansion of the fuel and cladding. This is not used with DEFORM-4; the value of 0 is used for the option of a force balance (or free expansion, if there is a fuel-to-cladding gap) between the fuel and the cladding. The model requires additional input on the relative strength and thermal expansion characteristics, which is in Block 63 for each channel. MODEEX = 0.
#56	JREEXT	This integer includes a correction to the reactivity extrapolation in the code. It should always be used unless comparison with older results is necessary. JREEXT = 1.
#59	NSUBTR	NSUBTR = 991; see Appendix C.
#60	NRRNGS	NRRNGS = 1; see Appendix C.
#61	MTRRAC	MTRRAC = 0; see Appendix C.
#62	MTRRT	MTRRT = 1; see Appendix C.
#63	MTRFAC	MTRFAC = 2; see Appendix C.
#64	MTRFT	MTRFT = 2; see Appendix C.
#65	IROPT	IROPT = 1; see Appendix C.

B. Block 11, OPCIN

This input block contains controls for time step size and for convergence criteria for various models. Many of these have default values, as indicated.

Location, Input Variable

#1	EPSTEM	The steady-state temperature convergence criterion for fuel-pin temperatures. The recommended value is 0.1 K, so EPSTEM = 0.1.
#3	EPSPOW	The neutron flux amplitude convergence criterion is based on typical values for previous studies; EPSPOW = 0.00001.
#5	DT0	The initial and maximum main time step length depends on the transient being analyzed, as follows:
	Accident	DT0
	LOF	0.25
	TOP	0.50
	LOHS	0.50

These values can be increased during stages of the transient when changes are slow.

#6	DTMXB	The maximum heat transfer time step is set to the current recommended value of 0.01: DTMXB = 0.01.
#7	TIMAX	The maximum problem time is arbitrary, depending on the length of calculation desired. This value is used in conjunction with MAXSTP, Block 1, Loc. 11, to control the length of the calculation. For this study,
	Accident	TIMAX
	LOF	60.0
	TOP	200.0
	LOHS	3000.0, with restarts
#9	TCOSTP	The number of CPU seconds at the end of the calculation reserved for writing the restart file is set to 10.0, which is sufficient for operations at ANL; TCOSTP = 10.0.
#10	DTFUEL	The maximum allowable fuel temperature change per heat transfer time step is set to the recommended value of 50 K; DTFUEL = 50.0.
#11	DTCLAD	The maximum allowable clad temperature change per heat transfer time step is set to the recommended value of 30 K; DTCLAD = 30.0.
#13	DTPO	The initial PRIMAR-4 time step size is not input, which forces it to be the same as the initial main time step size, if possible; DTPO = 0.0.
#26	DTSSCP	The heat transfer time step size for the "null transient", which is a calculation to determine the initial temperatures throughout the reactor system modelled by PRIMAR-4 when the component-to-component heat transfer option is used. (See also Block 3, Loc. 1288). The value selected is 20.0 seconds, based on the typical reactor system design; DTSSCP = 20.0.
#27	EPSSCP	The convergence criterion for component temperatures in the "null transient". For this study, 0.001 K is used; EPSSCP = 0.001.

C. Block 51, INPCHN

Block 51 contains the general integer input data for selecting general channel-dependent input options. This includes the number of segments and nodes per axial zone, correspondence of material properties with channel assignment, and selection of appropriate temperature nodes for plotting output. Much of the input is generated in connection with the channel geometry, which is covered in Section III.A, with reference to Fig. A-2.

Not to Scale

Fig. A-2. SAS4A Axial Zones and Fuel Nodes for the Oxide Fuel Core.

Location, Input Variable

#4	NPLN	Number of segments in the fission gas plenum. = 6										
#5	NREFB	Number of reflector zones below the pin section. = 1										
#6	NREFT	Number of reflector zones above the pin section. = 2										
#7-10	NZNODE(KZ)	Number of segments in zone KZ. (KZ=4 for oxide core). This is selected to make the segments in the reflector zone of similar size. <table><tr><td>KZ</td><td>Segments</td></tr><tr><td>1</td><td>3</td></tr><tr><td>2</td><td>30</td></tr><tr><td>3</td><td>2</td></tr><tr><td>4</td><td>4</td></tr></table>	KZ	Segments	1	3	2	30	3	2	4	4
KZ	Segments											
1	3											
2	30											
3	2											
4	4											
#14	NT	Number of radial temperature nodes within the fuel, (4 ≤ NT ≤ 11). Arbitrary within this range, but it is usually set to 11; NT = 11.										
#15	IFUELV	Table number of property value to be used from Block 13 for core fuel. <table><tr><td>Property Table</td><td>Channel Number</td></tr><tr><td>2</td><td>1</td></tr><tr><td>2</td><td>2</td></tr><tr><td>5</td><td>3</td></tr></table>	Property Table	Channel Number	2	1	2	2	5	3		
Property Table	Channel Number											
2	1											
2	2											
5	3											
#16	IFUELB	Table number of property value to be used from Block 13 for axial blanket fuel. <table><tr><td>Property Table</td><td>Channel Number</td></tr><tr><td>4</td><td>1</td></tr><tr><td>4</td><td>2</td></tr><tr><td>5</td><td>3</td></tr></table>	Property Table	Channel Number	4	1	4	2	5	3		
Property Table	Channel Number											
4	1											
4	2											
5	3											
#17	ICLADV	Table number of property value to be used from Block 13 for cladding table. This is the same for all channels. = 1										
#25	NPIN	Number of pins per subassembly. = 271 for driver assemblies. = 169 for blanket assemblies.										

- #26 NSUBAS Number of subassemblies in the channel.
Channel	Subassemblies
1	12
2	390
3	157
- #27 MZUB Number of segments in the upper blanket. This must be consistent with the axial nodes used for generating the neutronics data. = 5
- #28 MZLB Number of segments in the lower blanket. This must be consistent with the axial nodes used for generating the neutronics data. = 4.
- The following 5 values represent the axial nodes for which output is sent to the plotting file for post processing. The values do not have any effect on the calculation, or on the values typically plotted from File 11, the standard plot file.
- #90 JCLN Axial heat-transfer segment for fuel temperature information. This is usually taken as the top node in the core fuel region.
= 12 for the oxide core.
- #91 JNEN Axial coolant node for clad temperature. This node is selected as the same node for JCLN, Loc. 90. The value is different since these values are calculated on the coolant nodes.
= 29 for the oxide core.
- #92 JHCN Axial coolant node for coolant temperature. This node is selected as the same node for JCLN, Loc. 90. The value is different since these values are calculated on the coolant nodes.
= 29 for the oxide core.
- #93 JNSN Axial coolant node for structure temperature. This node is selected as the same node for JCLN, Loc. 90. The value is different since these values are calculated on the coolant nodes.
= 29 for the oxide core.
- #94 JRPRO Axial heat transfer segment for the total radial profile for the mass averaged fuel temperature. This node is usually the same as JCLN, Loc. 90.
= 12 for the oxide core.
- #118 IEQMAS Radial fuel nodes set up on an equal mass basis:
= 1

- #181 IAXEXP Calculate axial expansion feedback in each channel:
= 1
- #203 IDKCRV Decay heat curve is set the same for all channels.
= 1

III. Core Geometry

The oxide core designs were developed based on the general reactor specifications and constraints summarized in Ref. [A2]. In addition, the core designs were required to be interchangeable in the following sense. The designs have the same assembly pitch, the same number and location of control rods, the same overall assembly length, and the same radially heterogeneous core layout with only minor adjustments. A description of the oxide core layout is given in Fig. A-1. The radially heterogeneous layout has 402 drivers, 157 internal blankets, 90 radial blankets, and 36 control rods (24 primary and 12 secondary).

The assembly and pin design parameters for the oxide core are given in Table I in the main report. The mixed-oxide fueled driver pin has a 0.285 in. od, a 0.022 in. thick HT-9 cladding, and a 82.5% smeared fuel density. The UO_2 fueled blanket pin has a 0.392 in. od, a 0.022 in. thick HT-9 cladding, and a 93.3 % smeared fuel density. The oxide core subassembly axial geometry is described in Table II of the main report. The driver pin has a 40 in. high active fuel region with 14 in. upper and lower axial blankets. Below the lower axial blanket is a 40 in. lower fission gas plenum and above the upper axial blanket is a 16 in. gas plenum. The overall assembly length is 185 in.

The SAS4A code use a multi-channel treatment where each channel represents a fuel pin, its associated coolant, and a fraction of the subassembly duct wall. Each SAS4A channel is used to represent an average pin in a group of similar subassemblies. For this analysis a three channel assignment is used. Channel one represents the "lead" assembly - the one that will exhibit the highest temperature during any transient. Channel two includes all the remaining driver assemblies, and channel three includes all internal blanket assemblies. Figure A-2 shows the SAS4A channel axial description. The oxide core has four axial zones: a lower reflector with 3 axial nodes; a fuel pin zone with 6 nodes for the lower plenum, 4 nodes for the lower axial blanket, 15 nodes for the active core, and 5 nodes for the upper axial blanket; an upper plenum zone with 2 nodes; and an upper reflector zone with 4 nodes. The internal blankets have the same zones and noding as the drivers.

A. Block 61, GEOMIN

This input block contains all of the geometrical data required by the channel models in SAS4A. The input data are based on the information in Tables I and II in the main report. The channel models assume a single pin geometry for the calculation, which extends from the bottom to the top of the subassembly. The single pin is divided into a number of axial zones, where

one zone represents the fuel, axial blankets, and fission gas plenum. The data defining the zones in the channel is in Block 51. The following calculations define the input variables for the oxide fuel core: (NOTE that sqr = square root, sq = square).

Location, Input Variable

#1-4 ACCZ(KZ) The coolant flow area per fuel pin is the total flow area within the subassembly divided by the number of fuel pins. The total flow area is the cross-sectional area within the hexcan, minus the cross-sectional area occupied by the fuel pins and the spacer wire. The number of axial zones KZ is 4 for the oxide core, as shown in Fig. A-2. The same flow area is used in each zone. The calculation is identical for both driver and blanket pins, although different dimensions are used.

Driver Fuel Pins

$$\begin{aligned}\text{ACCZ(KZ)} &= (1/271) * \text{sqr}(3.0) / 2 * \text{sq}(\text{duct id}) - \text{pi} * \text{sq}(\text{pin od}/2) \\ &\quad - \text{pi} * \text{sq}(\text{wire od}/2) \\ &= (1/271) * \text{sqr}(3.0) / 2 * \text{sq}(5.631) - \text{pi} * \text{sq}(0.285/2) \\ &\quad - \text{pi} * \text{sq}(0.049/2) \\ &= 3.56490 \text{ d-02 sq in.} \\ &= 2.29993 \text{ d-05 sq m}\end{aligned}$$

Blanket Fuel Pin

$$\begin{aligned}\text{ACCZ(KZ)} &= (1/169) * \text{sqr}(3) / 2 * \text{sq}(\text{duct id}) - \text{pi} * \text{sq}(\text{pin od}/2) \\ &\quad - \text{pi} * \text{sq}(\text{wire od}/2) \\ &= (1/169) * \text{sqr}(3) / 2 * \text{sq}(5.631) - \text{pi} * \text{sq}(0.392/2) \\ &\quad - \text{pi} * \text{sq}(0.034/2) \\ &= 4.08903 \text{ d-02 sq in.} \\ &= 2.63808 \text{ d-05 sq m}\end{aligned}$$

#8-31 AHXI(J) The length of each axial node in the zone representing the fuel and blankets. This is the same axial node structure used for generating the neutronics data. According to Block 51, Loc. 4 and 7, there are 24 axial nodes for the fuel and the axial blankets. The node structure selected based on neutronics considerations is as follows, with all dimensions in m, starting from the bottom of the lower axial blanket:

$$\text{AHXI(J)} = 2 @ 1.27000\text{d-01}, 2 @ 5.08000\text{d-02}, 15 @ 6.77335\text{d-02}, 2 @ 5.08000\text{d-02}, \text{ and } 3 @ 8.46668\text{d-02}.$$

#32-35 DHZ(KZ) The hydraulic diameter in each zone is calculated using the following equation, with the same value in each zone:

$$\text{DHZ(KZ)} = 4 * (\text{Coolant Flow Area/Pin}) / (\text{Wetted Perimeter})$$

Driver Fuel Pins

$$\begin{aligned}
 P &= (\text{Wetted Perimeter}) \\
 &= 6 * (\text{duct id}) / 271 * \text{sqr}(3.0) + \pi * (\text{pin od}) + \pi * (\text{wire od}) \\
 &= 6 * (5.631) / 271 * \text{sqr}(3.0) + \pi * (0.285) + \pi * (0.049) \\
 &= 1.12127 \text{ in.} \\
 &= 2.84803 \text{ d-02 m}
 \end{aligned}$$

$$\begin{aligned}
 \text{DHZ(KZ)} &= 4 * (2.29993 \text{ d-05}) / (2.84803 \text{ d-02}) \\
 &= 3.23022 \text{ d-03 m}
 \end{aligned}$$

Blanket Fuel Pins

$$\begin{aligned}
 P &= (\text{Wetted Perimeter}) \\
 &= 6 * (\text{duct id}) / 169 * \text{sqr}(3.0) + \pi * (\text{pin od}) + \pi * (\text{wire od}) \\
 &= 6 * (5.631) / 169 * \text{sqr}(3.0) + \pi * (0.392) + \pi * (0.034) \\
 &= 1.45374 \text{ in.} \\
 &= 3.69250 \text{ d-02 m}
 \end{aligned}$$

$$\begin{aligned}
 \text{DHZ(KZ)} &= 4 * (2.63808 \text{ d-05}) / (3.69250 \text{ d-02}) \\
 &= 2.85779 \text{ d-03 m}
 \end{aligned}$$

#39-42 DSTIZ(KZ) The structure in the channel model represents the hexcan duct wall, the sodium in the gaps between the assemblies, and the spacer wire. The structure thickness is set to simulate the correct thermal time response for the structure. This is accomplished by first calculating an equivalent steel wall cross-sectional area representing the steel and sodium, according to the equation:

Effective wall area (assuming HT9 material properties at 400 deg C):

$$\begin{aligned}
 A'_{\text{wall}} &= A_{\text{wall}} \\
 &\quad + \text{Agap} * [\text{Cp}(\text{sodium}) * \rho(\text{sodium})] / [\text{Cp}(\text{HT9}) * \rho(\text{HT9})]
 \end{aligned}$$

The areas are calculated as follows, with id and od referring to the inside and outside flat-to-flat dimensions, respectively:

$$\begin{aligned}
 A_{\text{wall}} &= \text{sqr}(3.0) / 2 * [\text{sq}(\text{duct od}) - \text{sq}(\text{duct id})] \\
 &= \text{sqr}(3.0) / 2 * [\text{sq}(5.931) - \text{sq}(5.631)] \\
 &= 3.00390 \text{ sq in.} \\
 &= 1.93799 \text{ d-03 sq m} \\
 \text{Agap} &= \text{sqr}(3.0) / 2 * [\text{sq}(\text{lattice od}) - \text{sq}(\text{duct od})] \\
 &= \text{sqr}(3.0) / 2 * [\text{sq}(6.131) - \text{sq}(5.931)] \\
 &= 2.08920 \text{ sq in.} \\
 &= 1.34787 \text{ d-03 sq m} \\
 A'_{\text{wall}} &= 1.93799 \text{ d-03} \\
 &\quad + 1.34787 \text{ d-03} * [(855)(1275)] / [(615.4)(7656.7)] \\
 &= 1.93799 \text{ d-03} + 3.11835 \text{ d-04} \\
 &= 2.24985 \text{ d-03 sq m}
 \end{aligned}$$

The structure thickness is determined by using the equivalent area and the hexcan inside flat-to-flat distance:

$$\begin{aligned}
 &= (\text{sqr}[(2.24984\text{d-}03 \times 100^2)/(2.54)^2] \times 2 / \text{sqr}(3.0) \\
 &\quad + (5.631)^2) - 5.631) / 2 \\
 &= 1.73434 \text{ d-}01 \text{ in} \\
 &= 4.40522 \text{ d-}03 \text{ m}
 \end{aligned}$$

The structure thickness is divided into two radial nodes, and for the present study, half of the thickness is used for each node. Therefore:

$$\begin{aligned}
 \text{DSTIZ(KZ)} &= 1/2 \text{ Structure Thickness} \\
 &= 2.20261 \text{ d-}03 \text{ m}
 \end{aligned}$$

#46-49 DSTOZ(KZ) The thickness of the outer structure node is the same as the inner structure node, half of the thickness in each node, so that:

$$\begin{aligned}
 \text{DSTOZ(KZ)} &= 1/2 \text{ Structure Thickness} \\
 &= 2.20261 \text{ d-}03 \text{ m}
 \end{aligned}$$

#53 PLENL The length of the lower fission gas plenum is given in Table II in the main report as 1.016 m, so
PLENL = 1.01600 d+00 m (40.0 in)

#54-77 RBR(J) The clad inner radius for each axial segment in the fuel and axial blanket region is the same, as given by the cladding outer diameter and cladding thickness from Table I in the main report. The result is
= 3.06070 d-03 m for drivers
= 4.41960 d-03 m for blankets

#78-101 RER(J) The clad outer radius for each axial segment in the fuel and axial blanket region is the same, as given by the cladding outer diameter in Table I in the main report. The result is
= 3.61950 d-03 m for drivers
= 4.97840 d-03 m for blankets

#102 RBRPL The clad inner radius in the gas plenum is the same as for the fuel region;
= 3.06070 d-03 m for drivers
= 4.41960 d-03 m for blankets

#103 RERPL The clad outer radius in the gas plenum is the same as for the fuel region;
= 3.61950 d-03 m for drivers
= 4.97840 d-03 m for blankets

#104-127 RINFP(J) The fuel inner radius for each axial segment in the fuel region is set to 1.0×10^{-6} m, to accomodate the use of DEFORM-4.
= 1.00000 d-06 m for drivers
= 1.00000 d-06 m for blankets

#128 ROUTFP(J) The fuel outer radius for each axial segment in the fuel region is set to the clad inner radius, Loc. 54-77, when DEFORM-4 is not being used;

= 3.06070 d-03 m for drivers
 = 4.41960 d-03 m for blankets
 When DEFORM-4 is used, the fuel outer radius is calculated differently, as given in Appendix E.

#152-155 ZONEL(KZ) The length of each zone KZ in the single pin model is shown in Fig. A-2. Starting from the bottom of the assembly, with all lengths in m;

KZ	Oxide
1	6.85800 d-01
2	2.74320 d+00
3	4.06400 d-01
4	8.63600 d-01

#159-162 SRFSTZ(KZ) The structure perimeter is used in conjunction with the structure thickness, Loc. 39-52, to simulate the proper thermal time response of the structure.

Structure perimeter per pin =
 $\{A' \text{ wall}/\# \text{pins} + \pi/4 * (\text{wire od})^2\} / \text{wall thickness}$

SRFSTZ(KZ) = $\{2.24985 \text{ d-03}/271 + \pi/4 * (1.2446 \text{ od-03})^2\} / 4.40500 \text{ d-03}$
 = 2.16085 d-03 m for drivers

SRFSTZ(KZ) = $\{2.24985 \text{ d-03}/169 + \pi/4 * (8.63600 \text{ d-04})^2\} / 4.40500 \text{ d-03}$
 = 3.15494 D-03 m for blankets

#166 AREAPC The coolant plus pin area per pin, in the pin section, is determined by total cross-sectional area per pin and subtracting the cross-sectional area of the spacer wire:

= $(\text{sqr}(3.0)/2 * \# \text{pins}) * \text{sq}(\text{duct id}) - \pi * \text{sq}(\text{wire od}/2)$

AREAPC = $(\text{sqr}(3.0)/2 * 271) * \text{sq}(5.631) - \pi * \text{sq}(0.049/2)$ for drivers
 = 9.94429 d-02 sq in
 = 6.41566 d-05 sq m

AREAPC = $(\text{sqr}(3.0)/2 * 169) * \text{sq}(5.631) - \pi * \text{sq}(0.034/2)$ for blankets
 = 1.61578 d-01 sq in
 = 1.04244 d-04 sq m

#169-172 DRFO(KZ)
 #189-192 DRFI(KZ) The reflector zones above and below the fuel pin section are treated in a different manner. The thickness of the reflector zones is based on a heat transfer model which represents the reflector zones in a slab geometry. The reflector is assumed to be a solid pin with the same area and perimeter as the driver or blanket fuel pins. A two node slab geometry is used for the heat transfer to the to the reflector, where the effective reflector thickness can be obtained from:

$$\pi * R(\text{radius})^{**2} = P(\text{perimeter}) * t(\text{thickness})$$

where $P = 2 * \pi * R$

In order to preserve both the perimeter and the cross-sectional area,

$$t = R/2 \quad \text{for a slab geometry.}$$

$t = 1.80975 \text{ d-03}$ for the driver fuel pins

$t = 2.48920 \text{ d-03}$ for the blanket fuel pins

These thicknesses are divided equally between the inner and outer reflector nodes. The clad thickness for the fuel and fission gas plenum region is $5.58800 \times 10^{-4} \text{ m}$, which is used for the reflector thickness in this zone. For the reflector zone which simulates the upper fission gas plenum, this thickness is split equally between the inner and outer nodes with a value of $2.79400 \times 10^{-4} \text{ m}$ for each. The values for each zone in the drivers and blankets are:

Drivers:

KZ	reflector thickness		
	inner node	outer node	
1	9.04875 d-04	9.04875 d-04	reflector
2		5.58800 d-04	fuel pin
3	2.79400 d-04	2.79400 d-04	reflector
4	9.04875 d-04	9.04875 d-04	reflector

Blankets:

KZ	reflector thickness		
	inner node	outer node	
1	1.24460 d-03	1.24460 d-04	reflector
2		5.58800 d-04	fuel pin
3	2.79400 d-04	2.79400 d-04	reflector
4	1.24460 d-03	1.24460 d-03	reflector

#180 RBRO

The nominal cladding inner radius is the same as the cladding inner radius listed in Loc. 54-77;

= 3.06070 d-03 m for drivers

= 4.41960 d-03 m for blankets

#181 RERO

The nominal cladding outer radius is the same as the cladding outer radius listed in Loc. 78-101;

= 3.61950 d-03 m for drivers

= 4.97840 d-03 m for blankets

#182-185 SER(KZ)

The reflector perimeter per pin in each zone is the nominal pin perimeter, consistent with the cross-sectional area used to determine the reflector thickness in Loc. 169-175 and Loc. 189-195. The pin perimeter is also used in the gas plenum region;

= $2\pi(\text{Pin radius})$ for both driver and blanket assemblies.

= 2.27420 d-02 for drivers.

= 3.12802 d-02 for blankets.

IV. SAS4A Reactivity Worth Tables

The neutronics information used in the SAS4A codes is generated using a separate set of computer codes [A2]. The discussion in Ref. A2 can be summarized as follows, highlighting the overall features of the process:

The neutronics analysis on the oxide heterogeneous core design established the performance parameters needed for safety evaluations. ENDF/B-V.2 basic cross section data was used and processed through the MC2-2/SDX code system to generate broad-group libraries. A nine group cross section library was generated for use in depletion and rod worth calculations and a 21 group library was generated for determining reactivity coefficients. Global reactivity coefficients were evaluated for end-of-equilibrium cycle conditions.

The prompt neutron lifetime, beta-effective, and delayed neutron fractions were calculated using ENDF/B-V.2 delayed neutron data with 21 group flooded real and adjoint fluxes from a 3D hex-z finite difference model. All reactivity worth coefficients were calculated using 21 group cross-section libraries, a 3D hex-z finite difference model, and appropriate flooded or voided real and adjoint fluxes.

The sodium density worth was calculated using first order perturbation theory with flooded real and adjoint fluxes and flooded microscopic cross sections. The sodium void worth coefficient was calculated with the perturbation defined by voiding the flowing sodium from the core and axial blankets. Both the sodium number densities and the cross sections were changed to reflect changes in spectrum and self-shielding. The leakage term was treated rigorously.

The Doppler reactivity worths were calculated for both flooded and voided conditions. The flooded Doppler uses flooded real and adjoint fluxes and cross-sections reflecting full sodium density; while the voided Doppler used fluxes and cross-sections reflecting the voiding of flowing sodium in the core and axial blankets. The base case temperatures for the fuel isotopes was 1300 K for the oxide core. The Doppler reactivity worths were determined by perturbing the cross-sections to reflect fuel temperatures of 2600 K for the oxide core. The Doppler reactivity worths for the fuel pin clad and structural duct wall were also calculated. The base case temperature for the structure was 750 K and the perturbation temperature was 1500 K.

The fuel and clad material worths were also calculated using first order perturbation theory for both flooded and voided conditions. Appropriate fluxes and cross-sections were used with uniform changes in associated material densities. Radial and axial expansion reactivity loss coefficients were computed by eigenvalue differences using a 9 group cross-section library and a hex-z nodal model. The radial expansion reactivity loss was calculated for a uniform radial dilation of all core materials, then was normalized by

dividing by the change in radius of the core/radial blanket interface. The axial expansion reactivity loss was calculated for a uniform axial dilation of the driver, then normalized by dividing by the change in core fuel height.

Control rod bank differential worths were calculated by eigenvalue difference at their BOEC and EOEC full power operating positions. A special hex-z model with a fine axial mesh near rod tips was used with 9-group cross section libraries. These coefficients account for the control rod driveline expansion in ATWS accidents.

The information created by these calculations is converted into a form suitable for input to SAS4A for each subassembly [A2]. The data from this process is then grouped into the channel format as described in Section I.

A. Block 12, POWINA

The input data for input block 12 is summarized in this section. The input data support the discussion in Section II.A.1 on establishing the core modelling. Most of the data comes from Ref. A2, and was included in Ref. 5 of the main report, both of which contain descriptions of the neutronics calculations and results for the metallic and oxide fuel designs. The resulting input for block 12 is as follows, where only non-zero entries are indicated unless a zero value has a particular meaning:

Location, Input Variable

#2	GENTIM	The prompt neutron lifetime is obtained from Ref. A3, as follows; = 4.05400d-07 sec. for the oxide fuel core.
#3	POWTOT	The total reactor power is the same for both reactors at 3500 MWth; = 3.50000d+09, W
#4-9	BETADN(L)	The effective delayed neutron fraction for the delayed neutron precursor families. For this study, six delayed neutron precursor families were used, with the data obtained from Ref. A2 for each family with the following delayed neutron fractions;
	Family	Oxide Core
	1	7.76170d-05
	2	7.06290d-04
	3	6.13530d-04
	4	1.19000d-03
	5	5.44450d-04
	6	1.69620d-04
#10-15	DECCON(L)	The decay constant for each delayed neutron precursor family, obtained from Ref. A2, with units of 1/sec.

Family	Oxide Core
--------	------------

1	1.29750d-02
2	3.13260d-02
3	1.35070d-01
4	3.46130d-01
5	1.37920d+00
6	3.74970d+00

- #69 FRPR The fraction of total reactor power represented by the sum of all SAS4A channels. For the oxide core this is the fraction of the total power contained in the driver and internal blanket subassemblies. The data for performing this calculation were also obtained from Ref. A2. The result is FRPR = 9.68323d-01 for the oxide fuel core.
- #70 FRFLOW The fraction of the total reactor flow represented by the sum of all SAS4A channels was calculated from data contained in Ref. A3, which discusses the core orificing in detail; see Blocks 18 and 64. Again, this represents the fraction of the total flow that goes through the driver and internal blanket subassemblies, as follows;
= 9.15444d-01 for the oxide fuel core.
- #71 CRDLEN The length of the control rod drivelines washed by sodium leaving the active core region. As described in Section II.A.3.c, this length is measured from the top of the active fuel region in the core to the free surface in the outlet plenum, with the following result;
= 6.2992 m for the oxide fuel core.
- #72 CRDEXP The thermal expansion coefficient of the control rod drivelines is the suggested value which assumes that the drivelines are made of 316 stainless-steel;
= 2.00000d-05 per K.
- #73 ACRDEX The control rod driveline expansion reactivity feedback coefficient, where reactivity change = ACRDEX * (change in length + BCRDEX * (change in length)². The coefficients are listed in Table III in the main report, from Ref. A2, as follows:
= - 7.12 \$/m for the oxide fuel core.
- #74 BCRDEX Not used in the current model.
- #75 CRDMC The control rod driveline mass times specific heat, in J/K, is obtained from the suggested value in SAS4A, 5.6×10^4 for a control rod driveline 6.0 m in length. Assuming that the mass per unit length is the same, this value is factored according to the length given in Loc. 71 for the oxide core design:
 $(6.2992/6.0) \times 5.6 \times 10^4 = 5.87927 \times 10^4$, oxide core.

- #76 CRDHA The control rod driveline surface area times the average heat-transfer coefficient, J/deg K, is obtained from the suggested value in SAS4A of 2300.0 for a control rod driveline 6.0 m in length. This value is factored according to the driveline length in Loc. 71:
 $(6.2992/6.0) * 2300.0 = 2414.70$ for the oxide core.
- #77 UIVOL The coolant volume in the upper internal structure region is determined from the suggested value of 25.0 m³ given for a control rod driveline 6.0 m long and SAS4A. Factoring according to the control rod driveline length in Loc. 71,
 $(6.2992/6.0) * 25.0 = 26.2467$ for the oxide core.
- #260-265 BETADK (L,IPW) The decay heat precursor yield for each group used to represent the decay heat curve is taken from previous work on a similar fuel, as follows:
L	BETADK
1	2.20054d-02
2	1.90590d-02
3	9.93659d-03
4	5.25396d-03
5	1.93225d-03
6	1.67409d-03
- #290-295 DKLAM (L,IPW) The decay heat group decay constant for each group in the decay heat curve is also taken from previous work on a similar fuel, with the same values for both fuel types as follows:
L	DKLAM
1	6.56251d-02
2	1.77911d-03
3	1.37667d-04
4	6.08268d-06
5	2.25624d-07
6	1.89684d-08
- #320 BETAHT (IPW) The total decay heat precursor yield, which is simply the sum of the values for BETADK, Loc. 260-265:
 $= 6.22500d-02$
- #325 POWLVL (L,IPW) The normalized power history for initializing the decay heat is assumed to be constant operation at nominal conditions for one cycle. There is only one entry in the table, $= 1.00000d+00$
- #365 POWTIM (L,IPW) The time at power level POWLVL given in Loc. 325-364 is the time for one complete cycle, 80% of 365 days, or 292 days. Converting this to seconds,
 $= 2.52288 \times 10^7$ sec.

#408	SLLMAX	The maximum allowable slope of subassembly at grid plate with respect to vertical based on subassembly nozzle/grid plate clearances = 1.00. See App. C.
#409	PITCHG	The subassembly pitch at the grid plate at the reference temperature = 0.154827 m. See App. C.
#410	PITCHA	The flat-to-flat dimension across the above-core load pad at the reference temperature = 0.154887 m. See App. C.
#411	PITCHT	The flat-to-flat dimension across the top load pad at the reference temperature = 0.154887 m. See App. C.
#412	RDEXCF	The radial expansion reactivity feedback coefficient for a uniform core dilation, $\$/m$. See App. C. = -64.0 $\$/m$ for the oxide fuel core.
#413	TLPRRC	The clearance between the top load pads and the restraint ring. See App. C. = - 7.00000d-05 m for the oxide fuel core.
#414	BNDMM1	The applied bending moment at the top of the core region, representing the flat-to-flat temperature difference in this region for subassemblies at the outer edge of the active core. See App. C. = 1.00000d-03
#415	BNDMM2	The applied bending moment in the region above the core, representing the flat-to-flat temperature difference in this region for subassemblies at the outer edge of the active core. See App. C. = 1.00000d-03
#423	FCDTR2	The nominal steady-state top restraint ring temperature, expressed as a fraction of the average coolant temperature rise through the core. See App. C. = 1.00000d-01
#424	FCDTRF	The nominal steady-state reflector load pad temperature, expressed as a fraction of the coolant temperature rise through the core. See App. C. = 3.50000d-01
#426	CRSAC	Additional clearance in the interior of the core, or the difference between the actual core radius and the ideal core radius. See App. C. = 1.62400d-03
#428	RR2TC	The thermal response time constant for the top restraint ring. See App. C. = 5.00000d+02

B. Block 62, POWINC

This input block contains all of the channel dependent neutronics input data. The majority of the data comes from the combination of the subassembly-by-subassembly data for the core, according to the channels as described in Section I. This block also includes other data to account for differences in irradiation between the channels, but for this study, all assemblies were assumed to have an irradiation of 1 cycle. The data is originally generated for 63 subassemblies in a 1/12 symmetry representation of the reactor core [A2]. Each core position then represents a number of identical subassemblies around the core, which in this case is usually either 6 or 12, forming a "channel", (not to be confused with a SAS4A channel), as shown in Fig. A-1.

Location, Input Variable

#2	GAMSS	The fraction of total power used in direct heating of the structure. A representative value taken from previous work is used; = 7.8000 d-03
#4	GAMTNC	The fraction of total power used in direct heating of the coolant. A representative value taken from previous work is used; = 4.8000 d-03
#5	GAMTNE	The fraction of total power used in direct heating of the cladding. A representative value taken from previous work is used; = 1.14000 d-02
#6-29	PSHAPE(I)	The ratio of pin power at axial node J to the power in the peak axial fuel pin segment for all nodes in the core. The values are calculated from the combination of the data for the 63 individual "channels" into the 3 SAS4A channels. Ref. A2 contains the discussion on generating the assembly data. The values are in the listing of the input data in Appendix F for the oxide fuel core.
#30-40	PSHAPR(I)	The normalized radial power shape within the pin by radial node, I. A flat power profile is assumed for the oxide fuel pin, so a value of 1.0 is used for all nodes; = 1.00000 d+00 for all 11 radial nodes.
#45	PLIN(M)	The relative power level at the end of the M steady-state division. This input is used by DEFORM-4 to calculate the appropriate conditions in the pin at the start of the transient. For cases without DEFORM-4, one value of 1.0 is sufficient. The input required for using DEFORM-4 is discussed in Appendix E. = 1.00000 d+00

#62 ADOP The Doppler coefficient for the channel when the channel is not voided. This is also obtained from the combining of the "channel" data [A2], with the totals as listed in Table III in the main report. The units are ($\Delta k/k$).

Channel	Oxide Fuel
1	-1.96886d-04
2	-4.37996d-03
3	-4.33673d-03

#63 BDOP The Doppler coefficient for the channel when the channel is fully voided. The data is obtained from the combining of the "channel" data, as with ADOP, Loc. 62. The units are ($\Delta k/k$).

Channel	Oxide Fuel
1	-1.52839d-04
2	-3.28633d-03
3	-3.51787d-03

#64-87 WDOPA(J) The Doppler weighting factor for the case where the sodium coolant is still present in the subassembly. This factor is generated for each axial node within the fueled region of the pin. The values are calculated from the combining of the data for the 63 "channels" into 3 channels. The result is listed in Appendix F as part of the input deck.

#112-135 VOIDRA(J) The first-order coolant void reactivity worth is calculated by the combination of the data from the 63 "channels" into the 3 SAS4A channels. The result is listed in Appendix F.

#160-183 CLADRA(J) The first-order cladding reactivity worth is calculated by the combination of the data from the 63 "channels" into the 3 SAS4A channels. The result is listed in Appendix F.

#208-231 FUELRA(J) The first-order core fuel reactivity worth is calculated by the combination of the data from the 63 "channels" into the 3 SAS4A channels. The result is listed in Appendix F.

#256 PRSHAP The ratio of the power per subassembly averaged over this channel to the power per subassembly averaged over all channels is calculated as part of the process combining the data from the 63 "channels" into the 3 SAS4A channels. The value is normalized to give a value of 1.0 when summed over the three SAS4A channels weighted by the number of subassemblies in each channel.

V. Thermophysical Properties

The input data for the thermophysical properties of the core materials is contained in two input data blocks, Block 13 for the basic data, and Block 63 for any additional data specific to each channel and not covered in Block 13. Usually this is applied to certain model options, such as the force balance model for calculating the axial expansion of the fuel and cladding. Some of the properties are modified before being used as input data in order to be consistent with the models in SAS4A. The original data and any modifications are covered in detail for all of the properties needed for the oxide fuel core model.

A. Block 13, PMATCM

Block 13 contains general channel independent fuel and cladding properties. The oxide fuel properties were taken from the Nuclear Systems Materials Handbook [A4]. The cladding properties for HT-9 are also taken from the Nuclear Systems Materials Handbook.

Cladding Tables

- # 11-18 Table 1: HT-9 Cladding Thermal Conductivity
- # 71-78 Temperatures for Cladding Table 1, deg K.

Fuel Density Tables

- # 91-105 Table 1: Mixed Oxide, Theoretical Density
- #111-125 Table 2: Mixed Oxide, Fuel Pin Smear Density (82.5%)
- #131-143 Table 3: UO₂, Theoretical Density
- #151-163 Table 4: UO₂, Axial Blanket Pin Smear Density (82.5%)
- #171-183 Table 5: UO₂, Internal and Radial Blanket Smear Density (93.3%)
- #251-265 Temperatures for Fuel Density Table 1, deg K
- #271-285 Temperatures for Fuel Density Table 2, deg K
- #291-303 Temperatures for Fuel Density Table 3, deg K
- #311-323 Temperatures for Fuel Density Table 4, deg K
- #331-343 Temperatures for Fuel Density Table 5, deg K

Fuel Conductivity Tables

- #420-434 Table 1: Mixed Oxide, Fuel Conductivity
- #440-454 Table 2: Mixed Oxide, Fuel Conductivity
- #460-474 Table 3: UO₂, Fuel Conductivity
- #480-494 Table 4: UO₂, Axial Blanket Fuel Conductivity
- #500-514 Table 5: UO₂, Internal and Radial Blanket Fuel Conductivity
- #580-594 Temperatures for Fuel Conductivities, deg K

Fuel Specific Heat Tables

- #606-620 Table 1: Mixed Oxide 100.0% TD, Fuel Specific Heat
- #626-640 Table 2: Mixed Oxide 82.5% TD, Fuel Specific Heat

- #646-660 Table 3: UO₂ 100.0% TD, Fuel Specific Heat
 #666-680 Table 4: UO₂ 82.5% TD, Axial Blanket Fuel Specific Heat
 #686-700 Table 5: UO₂ 93.3% TD, Internal and Radial Blanket Fuel Specific Heat
 #766-780 Temperatures for Fuel Specific Heat, deg K

Cladding Property Tables

- #819-830 HT-9 Cladding Specific Heat
 #879-890 Temperatures for Cladding Specific Heat, deg K
 #990-1001 HT-9 Cladding Specific Heat * Density
 #1050-1061 Temperatures for Cladding Specific Heat * Density, deg K

Location, Input Variable

#11-18 EXKTAB (I,ICLAD) Thermal conductivity of HT-9 cladding, W/m-K. Cladding properties taken from Nuclear Systems Materials Handbook [A4].

#71-78 EXKTM (L) Cladding property temperatures, deg K.

Temperature deg K	HT-9 Thermal Conductivity W/m-K
373.00	24.9
473.00	25.4
573.00	25.9
673.00	26.3
773.00	26.8
873.00	27.2
973.00	27.6
1073.00	28.0

#91-183 RHOTAB (L,IFUEL) Theoretical fuel density, Kg/m³. Oxide fuel properties taken from Nuclear Systems Materials Handbook [A4], The mixed oxide theoretical densities were adjusted to 85.5% to account for fuel porosity, for Table 2. The UO₂ theoretical densities were adjusted to 82.5% for Table 4, and to 93.3% for Table 5.

#251-343 RHOTEM (L,IFUEL) Temperatures for fuel density tables, deg K.

Fuel Densities Kg/m³

Temperature deg K	Mixed Oxide		UO ₂		
	100% TD	82.5 TD	100% TD	82.5 TD	93.3 TD
373.15	10960.6	9042.5	10938.7	9024.4	10205.8
673.15	10878.2	8974.5	10856.9	8956.9	10129.5
973.15	10780.5	8893.9	10759.6	8876.7	10038.7
1273.15	10667.4	8800.6	10647.0	8783.8	9933.7
1573.15	10540.1	8695.6	10519.4	8678.5	9814.6

1873.15	10397.4	8577.8	10377.4	8561.4	9682.1
2173.15	10241.5	8449.2	10221.1	8432.4	9536.3
2473.15	10070.2	8307.9	10051.0	8292.1	9377.6
2773.15	9886.9	8156.7	9867.5	8140.7	9206.4
2973.15	9756.3	8048.9			
3000.00	9294.9	7668.3			
3025.15	8929.2	7366.6			
3073.15	8880.9	7326.7	9670.8	7978.4	9022.9
3120.15			9638.8	7952.0	8993.0
3120.15			8860.0	8860.0	8860.0
3250.15			8740.9	8740.9	8740.9
3673.15	8351.8	7173.9			

- #419 TR Reference design point temperature at which pin dimensions are measured, deg K.
= 300.0
- #420-509 XKTEB Fuel thermal conductivity, W/m-K. The mixed oxide
(L,IFUEL) and UO2 thermal conductivities for theoretical density were reduced to 82.1% for Tables 1-4 for an assumed fractional porosity of 10%. The tables were also reduced for use with DEFORM-4 which does not automatically correct thermal conductivities. The UO2 internal and radial blanket thermal conductivities were reduced to 91.7% for an assumed fractional porosity of 5%.
- #580-594 XKTEM
(L)

Temperature deg K	Mixed Oxide		UO2		
	100% TD	82.5% TD	100% TD	82.5% TD	93.3% TD
273.15	10.746	10.746	7.060	7.060	7.881
473.15	6.189	6.189	5.316	5.316	5.935
673.15	4.360	4.360	4.208	4.208	4.697
873.15	3.388	3.388	3.444	3.444	3.845
1073.15	2.801	2.801	2.890	2.890	3.226
1273.15	2.428	2.428	2.472	2.472	2.760
1473.15	2.192	2.192	2.158	2.158	2.409
1673.15	2.054	2.054	1.935	1.935	2.160
1873.15	1.994	1.994	1.813	1.813	2.024
2073.15	1.999	1.999	1.816	1.816	2.027
2273.15	2.064	2.064	1.976	1.976	2.206
2473.15	2.185	2.185	2.331	2.331	2.602
2673.15	2.361	2.361	2.920	2.920	3.260
2873.15	2.593	2.593	2.870	2.870	3.204
3073.15	2.880	2.880	2.833	2.833	3.163

- #606-701 CPFTAB Fuel specific heat, J/Kg-K. Fuel specific heat properties
(L,IFUEL) taken from Nuclear Systems Materials Handbook [A4],
- #766-785 CPFTAB Temperatures for fuel specific heat tables, deg K.
(L) Oxide Fuel Specific Heat, J/Kg-K.

Temperature	Specific Heat
deg K	J/Kg-K
273.15	224.69
473.15	276.10
673.15	295.14
873.15	306.36
1073.15	314.85
1273.15	322.21
1473.15	329.37
1673.15	337.97
1873.15	351.53
2073.15	376.34
2273.15	421.48
2473.15	498.14
2673.15	618.44
3120.00	618.67
3120.00	485.00

#786-791	TFSOL	Fuel solidus temperature, deg K. 2975.15 for mixed oxide (PuO_2 , UO_2) 3110.15 for UO_2
#794-799	TFLIQ	Fuel liquidus temperature, deg K. 3021.15 for mixed oxide (PuO_2 , UO_2) 3130.15 for UO_2
#802-807	UFMELT (IFUEL)	Fuel Heat of fusion, J/Kg. 2.64896D+05 for mixed oxide (PuO_2 , UO_2) 2.77037D+05 for UO_2
#810	TESOL (ICLAD)	Cladding solidus temperature, deg K. 1690.15 for HT-9
#813	TELIQ (ICLAD)	Cladding liquidus temperature, deg K. 1710.15 for HT-9
#816	UEMELT (ICLAD)	Cladding heat of fusion, J/Kg. 2.68300D+05 for HT-9
#819-830	CPCTAB	Cladding specific heat, J/Kg-K.
#879-890	CPCTAM	Temperatures for cladding specific heat tables, deg K.

Temperature	HT-9 Specific Heat
deg K	J/Kg-K
373.0	482.50
473.0	521.48
573.0	563.37
673.0	615.38
773.0	684.68
873.0	784.96
923.0	866.08
973.0	1021.50

998.0	1021.00
1023.0	868.34
1073.0	747.97
1088.6	762.14

#990-1002 CROETB (L) Specific heat * density for HT-9 cladding, J/m**3-K.

#1050-1061 CROETM (L) Temperatures, deg K.

Temperature deg K	HT-9 Specific Heat * Density J/m**3-K
373.0	3.73566D+6
473.0	4.02379D+6
573.0	4.33085D+6
673.0	4.71178D+6
773.0	5.22021D+6
873.0	5.95863D+6
923.0	6.55969D+6
973.0	7.71948D+6
998.0	9.13369D+6
1023.0	6.54728D+6
1073.0	5.62698D+6
1088.6	5.72962D+6

#1086 RGASSI Ideal gas constant, J/gram-mole.

= 8.31434

B. Block 63, PMATCH

This input block contains thermophysical properties for the subassemblies represented by each channel that are not already input in Block 13, PMATCH. Some of the input also relates to DEFORM-4, and the generation of that input data is covered in Appendix E. Only non-zero values are reviewed, unless a value of zero has a special meaning.

Location, Input Variable

#2	AHBPAR	The coefficient in the bond gap conductance correlation; from previous work, AHBPAR = 1800. Not used with DEFORM-4.
#3	BHBPAR	From previous work, BHBPAR = 6.1×10^{-5} . Not used with DEFORM-4.
#4	CHBPAR	From previous work, CHBPAR = 1.32×10^{-6} . Not used with DEFORM-4.
#5	HBMAX	The maximum bond conductance when a gap exists, and the minimum when a gap does not exist is set to 8000.0 based on DEFORM-4 results; HBMAX = 8000.0 for cases not using DEFORM-4. Also see Appendix E.

#6	HBMIN	The minimum value of bond conductance at any time is set to 8000.0; HBMIN = 8000.0 for cases not using DEFORM-4. Also see Appendix E.
#7	HBPAP	Set to 0.0286 based on previous results; HBPAP = 0.0286.
#11-14	XKSTIZ	The thermal conductivity of the inner structure node by axial zone is set to 27.2 W/mK based on the properties of HT-9 at the operating conditions.
#18-21	XKSTOZ	The thermal conductivity of the outer structure node by axial zone is set to 27.2 W/mK based on the properties of HT-9 at the operating conditions.
#25	DEL	The product of Stefan-Boltzmann's constant and the emissivity of the cladding is 4.818×10^{-8} based on Block 13, Loc. 1107 and 1109; DEL = 4.818×10^{-8} .
#27	POGAS	The initial gas pressure in the fuel pin at the reference temperature TR (Block 13, Loc. 419) is assumed to be atmospheric pressure; POGAS = 1.01325×10^5 .
#28-31	XKRF	The thermal conductivity of the reflector by axial zones is set to 27.2 W/mK based on the properties of HT-9 at the operating conditions.
#35	DENSS	The density of the cladding at the reference temperature TR is 7760 kg/m ³ based on the properties of HT-9.
#36	COOLDN	The coolant density used in the calculation of the coolant void reactivity worth is set to 845 kg/m ³ based on comparison with the void worth listed in Table III in the main report.
#37-40	RHOCSI	The product of density and heat capacity for the inner structure is 5.9856×10^6 based on the properties of HT-9 at the operating temperature.
#44-47	RHOCSO	The product of density and heat capacity for the outer structure is 5.9856×10^6 based on the properties of HT-9 at the operating temperature.
#51-54	RHOCCR	The product of density and heat capacity for the reflector is 5.9856×10^6 based on the properties of HT-9 at the operating temperature.
#58	RHO CG	The product of density and heat capacity for gas in the fission gas plenum is 5.57×10^5 based on previous work.
#59	RG	The thermal resistance of the gas in the fission gas plenum is set to 0.03 from previous work.
#73	FUELEX	The axial expansion coefficient of the fuel is set to 1.10×10^{-5} per K based on the available data for oxide fuel at operating temperature.

#74	CLADEx	The axial expansion coefficient of the cladding is set to 1.40×10^{-5} per K based on the available data for HT-9 at operating temperature.
#75	YFUEL	The Young's modulus of the oxide fuel is estimated at $1.5 \times 10^{+11}$ based on available data at operating temperature.
#76	YCLAD	The Young's modulus of the cladding is estimated at $1.52056 \times 10^{+11}$ based on available data.
#77	FULREX	The fuel linear expansion coefficient is the same as FUELEX, Loc. 73; $FULREX = 1.10 \times 10^{-5}$.
#78	CLDREX	The cladding linear expansion coefficient is the same as CLADEx, Loc. 74; $CLDREX = 1.40 \times 10^{-5}$.
#79	EXPCFF	The effective multiplier for axial fuel expansion is set to 1.0 for nominal fuel expansion.

C. Block 65, FUELIN

The only input data in this block is one entry, Loc. 2, which must be input to the code. The suggested value of 0.2 is input. This value is only used for avoiding difficulties in job execution, but the calculations do not use this value.

VI. Coolant Flow

The thermal-hydraulic analysis for the oxide core design was performed using the multi-assembly, steady-state thermal-hydraulic code SUPERENERGY-2. [A3] Hot channel subfactor data was taken from current work with appropriate changes made to account for the sodium bond used in the metal fuel. A separate thermal-hydraulic analysis was performed for each design, and no driver or blanket shuffling was employed [A3].

Nozzle orificing was used with the number of discriminator zones fixed at 10. Orificing zones were defined by grouping assemblies with respect to maximum value of power times skew over lifetime. Assemblies with similar values were placed in the same orifice zone, and the mixing of driver and blanket assemblies within a zone was not permitted. The core fuel-cycle calculations were performed using the equilibrium-cycle approach with the assembly powers adjusted by stage factors to obtain the actual power at a specific point in assembly life. Explicit gamma transport and gamma heating rates are included in the spatial power profiles used to predict temperatures. Thus the heating rate in reflector and control assemblies where gamma heating is an important contribution to assembly power is properly predicted.

A fixed percentage of the total available flow is allocated for other than fuel and blanket assemblies. Reflector assemblies received 1.4% of the flow, control assemblies 0.5%, and 1.4% was allocated for cold bypass flow. The remainder of the total flow is allocated among driver and blanket assemblies so as to minimize the peak cladding midwall temperature for all fueled orifice zones over all life. This orificing strategy yielded a peak two-sigma cladding midwall temperature of 937 K (1227 deg F) for the oxide core.

A. Block 14, PRIMIN

This input block contains data on the coolant flow and temperatures in the primary loop. Most of the data is used by PRIMAR-4, while the rest involves the boiling model. Only nonzero values are discussed.

Location, Input Variable

#1	PX	The coolant exit pressure at ZPLENU, Loc. 88, is calculated based on atmospheric pressure at the surface of the outlet plenum, with the elevations as given in Appendix D. The pressure and elevation ZPLENU must be consistent with the PRIMAR-4 pressure and elevation in Block 18. For the current calculation, the value is 1.52599×10^5 Pa; $PX = 1.01325 \times 10^5 + (830 \text{ kg/m}^3)(9.807 \text{ m/sec}^2)(6.2992 \text{ m})$ $= 1.52599 \times 10^5 \text{ at the subassembly exit.}$
#45	TOTAB	The core inlet temperature as a function of time: for this study, PRIMAR-4 calculates the transient coolant temperatures, so only the steady-state value is input. TOTAB = 630.15.
#65	TOTME	The time for the input in TOTAB is only 0.0, to represent steady-state; TOTME = 0.0.
#85	TNTRY	This is the temperature of the sodium reentering the subassembly from the top. It is set to the steady-state outlet plenum temperature; TNTRY = 783.0.
#87	ZPLENL	The reference elevation for the core inlet plenum is set to the elevation of the bottom of the subassembly, ZPLENL = -1.7018 m.
#88	ZPLENU	The reference elevation for the core outlet plenum is set to the elevation of the top of the subassembly, ZPLENU = 2.9972 m.
#90	DZBGCL	The minimum distance for bubble breakaway above the subassembly has been set to the suggested value, DZBGCL = 0.10.
#91	DZBCGU	The maximum distance for bubble breakaway above the subassembly has been set to the suggested value, DZBCGU = 0.25.

B. Block 64, COOLIN

This input block contains all of the coolant flow information required by each channel. Most of the data is based on previous work, while the rest is from the current core orificing [A3].

Location, Input Variable

#1	AFR	The liquid slug friction factor coefficient is set to the current recommended value, 0.4089.												
#2	BFR	The liquid slug friction factor coefficient is set to the current recommended value, -0.25.												
#3-5	C1,C2,C3	The coefficients in the convective heat transfer correlation (Lyon-Martinelli) are set to the current recommended values: C1 = 0.0144 C2 = 0.86 C3 = 4.480												
#7	RELAM	The Reynolds number for the transition from the laminar friction factor correlation to the turbulent friction factor correlation is set to the current recommended value of 1509.												
#8	AFLAM	The coefficient in the laminar friction factor correlation is set to the current recommended value of 99.												
#47	W0	The steady-state coolant flow rate per pin in the channel is obtained from the process of grouping the channels, and is the average of the flow rates of the assemblies in each channel, weighted by the number of assemblies in each orifice zone; <table data-bbox="580 1284 1128 1419"> <thead> <tr> <th>Channel</th><th>W0</th><th></th></tr> </thead> <tbody> <tr> <td>1</td><td>0.126907</td><td>in kg/sec/pin</td></tr> <tr> <td>2</td><td>0.118135</td><td>"</td></tr> <tr> <td>3</td><td>0.133845</td><td>"</td></tr> </tbody> </table>	Channel	W0		1	0.126907	in kg/sec/pin	2	0.118135	"	3	0.133845	"
Channel	W0													
1	0.126907	in kg/sec/pin												
2	0.118135	"												
3	0.133845	"												
#67-68	THETA1,2	The coefficients in the numerical scheme for determining the degree of implicitness. Both are set to 0.5.												
#69	DTLMAX	The maximum temperature change of the liquid coolant per time step is set to the suggested value of 15.												
#70	DTVMAX	The maximum temperature change of the coolant vapor per time step is set to the suggested value of 50.												
#71	DZIMAX	The maximum motion of the liquid/vapor interface per time step is set to the suggested value of 0.10.												

#72	HCOND	The condensation heat transfer conductance of the coolant is set to 63000 based on previous work.
#73	SLMIN	The minimum initial length of a liquid slug is set to the suggested value of 0.02.
#75	WFMIN	The minimum thickness of a coolant film on the cladding is set to the default value of 1.00×10^{-7} .
#76	WFMINs	The same as WFMIN, 1.00×10^{-7} .
#77	WFS00	The same as WF0, Loc. 84; 1.427×10^{-4} .
#84	WF0	The initial film thickness is set to 0.15 of the total liquid volume fraction = 1.427×10^{-4} .
#165	DTS	The degrees of superheat required before any bubble formation is set to 10 K.

REFERENCES

- A-1 G. Birgersson et al., "The SAS4A LMFBR Accident Analysis Code System," ANL/RAS 83-38, Revision 2 (February 1988).
- A-2 E. K. Fujita and D. C. Wade, "The Neutronic and Fuel Cycle Performance of Interchangeable 3500 MWth Metal and Oxide Fueled LMRs," ANL-FRA-163 (March 1989).
- A-3 R. B. Vilim and R. N. Hill, "The Steady-State Thermal-Hydraulic Performance of Interchangeable 3500 MWth Metal and Oxide Fueled LMRs," ANL-FRA-164 (March 1989).
- A-4 Nuclear Systems Materials Handbook, TID-26666, Hanford Engineering Development Laboratory.

APPENDIX B. SAS4A MODELLING OF THE METALLIC FUEL CORE

The modelling of the metallic fuel core is discussed in detail in this appendix. The first topic covered is the combining of the subassemblies into appropriate SAS4A channels [B1], including the rationale for selecting the combination used for this study. The generation of the SAS4A input data deck is covered next. This is done block-by-block for every input block. The basis for each value is given, along with any calculations required to arrive at the input value. Background information which was not included in Ref. 5 is also provided.

I. Channel Modelling of the Core

The neutronics analysis for the interchangeable metal and oxide core design specified each assembly uniquely [B2]. The assembly assignment is shown in Fig. B-1 for the metallic fuel core and represents 63 different individual assembly locations. In order to model this core with SAS4A, the assemblies must be grouped into a relatively small number of channels. The grouping of the subassemblies into channels is done according to the following general guidelines:

1. Assembly with the highest average power-to-flow ratio with a respectively high average power.
2. All remaining driver assemblies.
3. All remaining internal blanket assemblies.

The reasons for this arrangement are briefly discussed in Section II.A.2. The radial blanket assemblies are each treated as a bypass channel, as are the control rod assemblies and shield assemblies, and true bypass flow. This is covered in detail in Appendix D.

A detailed description of the assembly assignment number, orifice zone, relative EOEC average power per assembly, relative flow per assembly, and relative power-to-flow ratio is given in Table B-1 for the metallic fuel core. For the metallic fuel core, assembly location 5 has the highest average power per assembly and the highest power-to-flow ratio for the high power assemblies and therefore is the best choice for the hottest channel assembly.

II. General Integer Input and Options

The integer input for the core models is contained in 2 input blocks. Block 1 contains the integer input which applies on a core-wide basis, while Block 51 integer input data applies to each channel. General input for running the codes is contained in Block 11. The selection of model options is discussed, along with the interactions with other input data blocks. The final result is listed in Appendix G which contains the metal fuel core input data decks.

Fig. B-1. Assembly Assignment for the 3500 MWth Metallic Fuel Heterogeneous Core.

TABLE B-1. SAS4A Channel/Assembly Parameters at EOEC For Metal Core

Asm Num	Asm Type	SAS Channel	Orifice Zone	Total Asm	Pwr/Asm (Rel)	Flow/Asm (Rel)	Pwr/Flow (Rel)
6	Dr	2	1	12	1.60320	1.31530	1.219
8	Dr	2	1	12	1.57543	1.31530	1.198
9	Dr	2	1	12	1.59255	1.31530	1.211
11	Dr	2	1	6	1.48581	1.31530	1.130
12	Dr	2	1	12	1.47331	1.31530	1.120
16	Dr	2	1	6	1.38346	1.31530	1.052
17	Dr	2	1	12	1.41342	1.31530	1.075
5	Dr	1	2	6	1.61197	1.29829	1.242
7	Dr	2	2	12	1.54523	1.29829	1.190
14	Dr	2	2	12	1.49937	1.29829	1.155
15	Dr	2	2	12	1.52257	1.29829	1.173
18	Dr	2	2	12	1.45651	1.29829	1.122
10	Dr	2	3	12	1.55889	1.25181	1.245
19	Dr	2	3	6	1.25885	1.25181	1.006
20	Dr	2	3	12	1.32810	1.25181	1.061
22	Dr	2	3	12	1.38688	1.25181	1.108
13	Dr	2	4	12	1.46411	1.23140	1.189
21	Dr	2	4	12	1.31633	1.23140	1.069
24	Dr	2	4	12	1.16260	1.23140	0.944
25	Dr	2	4	12	1.22182	1.23140	0.992
26	Dr	2	4	12	1.24384	1.23140	1.010
27	Dr	2	4	12	1.27165	1.23140	1.033
1	Dr	2	5	12	1.29780	1.16110	1.118
2	Dr	2	5	6	1.47202	1.16110	1.268
3	Dr	2	5	12	1.42741	1.16110	1.229
4	Dr	2	5	6	1.42283	1.16110	1.225
23	Dr	2	5	6	1.04502	1.16110	0.900
29	Dr	2	5	12	1.01241	1.16110	0.872
30	Dr	2	5	12	1.09107	1.16110	0.940
31	Dr	2	5	12	1.10283	1.16110	0.950
32	Dr	2	5	12	1.14072	1.16110	0.982
28	Dr	2	6	12	0.86528	0.95303	0.908
33	Dr	2	6	12	0.78548	0.95303	0.824
34	Dr	2	6	12	0.85639	0.95303	0.899
35	Dr	2	6	12	0.86135	0.95303	0.904
36	Dr	2	6	12	0.87764	0.95303	0.921
37	Dr	2	6	6	0.90427	0.95303	0.949
43	IB	3	7	12	0.82490	0.95949	0.860
45	IB	3	7	6	0.82755	0.95949	0.862

TABLE B-1. SAS4A Channel/Assembly Parameters at EOEC For Metal Core (Cont'd)

Asm Num	Asm Type	SAS Channel	Orifice Zone	Total Asm	Pwr/Asm (Rel)	Flow/Asm (Rel)	Pwr/Flow (Rel)
46	IB	3	7	6	0.87338	0.95949	0.910
47	IB	3	7	12	0.85198	0.95949	0.888
48	IB	3	7	12	0.86561	0.95949	0.902
49	IB	3	7	12	0.87712	0.95949	0.914
53	IB	3	7	6	0.83387	0.95949	0.869
42	IB	3	8	6	0.62234	0.85211	0.730
44	IB	3	8	12	0.77975	0.85211	0.915
50	IB	3	8	12	0.78283	0.85211	0.919
51	IB	3	8	12	0.77837	0.85211	0.913
52	IB	3	8	12	0.74977	0.85211	0.880
54	IB	3	8	12	0.72084	0.85211	0.846
55	IB	3	8	12	0.54116	0.85211	0.635
40	IB	3	9	6	0.48017	0.54313	0.884
41	IB	3	9	6	0.45137	0.54313	0.831
56	RB	BP	9	6	0.31278	0.54313	0.576
58	RB	BP	9	12	0.29578	0.54313	0.545
38	IB	3	10	1	0.31421	0.37305	0.842
39	IB	3	10	6	0.35247	0.37305	0.945
57	RB	BP	10	12	0.21324	0.37305	0.572
59	RB	BP	10	12	0.18413	0.37305	0.494
60	RB	BP	10	12	0.22574	0.37305	0.605
61	RB	BP	10	12	0.24201	0.37305	0.649
62	RB	BP	10	12	0.24745	0.37305	0.663
63	RB	BP	10	12	0.25321	0.37305	0.679

A. Block 1, INPCOM

Block 1 contains general integer input data for the SAS4A code and models. It includes time step controls for printing, restart files and problem time. Many selection flags for model options are also set in this input block. Only nonzero values are discussed unless a value of zero has a particular meaning.

Location, Input Variable

#1	NCHAN	Same as oxide; NCHAN = 3.								
#3	IFUEL1	The number of fuel types used in the channel models. This is used to set the number of property tables in Block 13. For this study, IFUEL1 = 5.								
#4	ICLAD1	Same as oxide; ICLAD1 = 1.								
#5	IPLUP	This flag selects the location of the fission gas plenum, which for the metallic fuel core is above the core region. The appropriate value is then IPLUP = 0.								
#8	IPOWER	Same as oxide; IPOWER = 0.								
#11	MAXSTP	Same as oxide. <table><tr><td>Accident</td><td>MAXSTP</td></tr><tr><td>LOF</td><td>400</td></tr><tr><td>TOP</td><td>800</td></tr><tr><td>LOHS</td><td>3000</td></tr></table>	Accident	MAXSTP	LOF	400	TOP	800	LOHS	3000
Accident	MAXSTP									
LOF	400									
TOP	800									
LOHS	3000									
#12	IPO	Same as oxide. <table><tr><td>Accident</td><td>IPO</td></tr><tr><td>LOF</td><td>20</td></tr><tr><td>TOP</td><td>50</td></tr><tr><td>LOHS</td><td>50</td></tr></table>	Accident	IPO	LOF	20	TOP	50	LOHS	50
Accident	IPO									
LOF	20									
TOP	50									
LOHS	50									
#13	IPOBOI	The same as Loc. 12, but for printing after the initiation of boiling. The values in Loc. 12 are repeated for this study.								
#15	NSTEP	Same as oxide; NSTEP = 100.								
#16	NDELAY	The number of delayed neutron precursor families is 6, as obtained from Ref. 6 and used in Block 12 in several locations; NDELAY = 6.								
#17	NDKGRP	The number of decay heat groups is 6, as used in Block 12; NDKGRP = 6.								
#22	NTOTAB	Same as oxide; NTOTAB = 1.								
#23	MULSTR	Same as oxide; MULSTR = 0.								
#24	ICLCMP	Same as oxide; ICLCMP = 11.								

#25	INEDIT	Same as oxide; INEDIT = 0.
#27	IPRION	Same as oxide; IPRION = 4.
#31	ICREXP	Same as oxide; ICREXP = 2.
#36	IRADEX	Same as oxide; IRADEX = -4.
#39	NFUELD	Same as oxide; NFUELD = -5.
#45	NPOWDK	The number of decay heat curves used in the modelling of the core. In this study, each channel is assumed to have the same decay heat curve; NPOWDK = 1.
#46	NPKST	The number of time periods used for initializing the decay heat curves is 1, with the time set in Block 12; NPKST = 1.
#51	NSUBTC	NSUBTC = 595; see Appendix C.
#52	MTGRD	MTGRD = 1; see Appendix C.
#53	MTACLP	MTACLP = 2; see Appendix C.
#54	MTTLP	MTTLP = 2; see Appendix C.
#55	MODEEX	This integer selects the model used for calculating the reactivity feedback from axial expansion of the fuel and cladding. This is not used with DEFORM-4; when DEFORM-4 is not used, the value of 0 is used for the option of a force balance (or free expansion, if there is a fuel-to-cladding gap) between the fuel and the cladding. The model requires additional input on the relative strength and thermal expansion characteristics, which is in Block 63 for each channel; MODEEX = 0.
#56	JREEXT	Same as oxide; JREEXT = 1.
#59	NSUBTR	NSUBTR = 991; see Appendix C.
#60	NRRNGS	NRRNGS = 1; see Appendix C.
#61	MTRRAC	MTRRAC = 0; see Appendix C.
#62	MTRRT	MTRRT = 1; see Appendix C.
#63	MTRFAC	MTRFAC = 2; see Appendix C.
#64	MTRFT	MTRFT = 2; see Appendix C.
#65	IROPT	IROPT = 1; see Appendix C.

B. Block 11, OPCIN

This input block contains controls for time step size and for convergence criteria for various models. Many of these values have default values, as indicated.

Location, Input Variable

#1	EPSTEM	The steady-state temperature convergence criterion for fuel-pin temperatures. The recommended value is 0.1 K, so EPSTEM = 0.1.								
#3	EPSPOW	The neutron flux amplitude convergence criterion is based on typical values for previous studies; EPSPOW = 0.00001.								
#5	DT0	The initial and maximum main time step length depends on the transient being analyzed, as follows: <table border="0"> <tr> <td>Accident</td> <td>DT0</td> </tr> <tr> <td>LOF</td> <td>0.25</td> </tr> <tr> <td>TOP</td> <td>0.50</td> </tr> <tr> <td>LOHS</td> <td>0.50</td> </tr> </table> These values can be increased during stages of the transient when changes are slow.	Accident	DT0	LOF	0.25	TOP	0.50	LOHS	0.50
Accident	DT0									
LOF	0.25									
TOP	0.50									
LOHS	0.50									
#6	DTMXB	The maximum heat transfer time step is set to the current recommended value of 0.01: DTMXB = 0.01.								
#7	TIMAX	The maximum problem time is arbitrary, depending on the length of calculation desired. This value is used in conjunction with MAXSTP, Block 1, Loc. 11, to control the length of the calculation. For this study, <table border="0"> <tr> <td>Accident</td> <td>TIMAX</td> </tr> <tr> <td>LOF</td> <td>100.0</td> </tr> <tr> <td>TOP</td> <td>200.0</td> </tr> <tr> <td>LOHS</td> <td>3000.0, with restarts</td> </tr> </table>	Accident	TIMAX	LOF	100.0	TOP	200.0	LOHS	3000.0, with restarts
Accident	TIMAX									
LOF	100.0									
TOP	200.0									
LOHS	3000.0, with restarts									
#9	TCOSTP	The number of CPU seconds at the end of the calculation reserved for writing the restart file is set to 10.0, which is sufficient for operations at ANL; TCOSTP = 10.0.								
#10	DTFUEL	The maximum allowable fuel temperature change per heat transfer time step is set to the recommended value of 50 K; DTFUEL = 50.0.								
#11	DTCLAD	The maximum allowable clad temperature change per heat transfer time step is set to the recommended value of 30 K; DTCLAD = 30.0.								
#13	DTP0	The initial PRIMAR-4 time step size is not input, which forces it to be the same as the initial main time step size, if possible; DTP0 = 0.0.								

- #26 DTSSCP The time step size for the "null transient", which is a calculation to determine the initial temperatures throughout the reactor system modelled by PRIMAR-4 when the component-to-component heat transfer option is used. (See also Block 3, Loc. 1288). The value selected is 20.0 seconds, based on the typical reactor system design; DTSSCP = 20.0.
- #27 EPSSCP The convergence criterion for component temperatures in the "null transient". For this study, 0.001 K is used; EPSSCP = 0.001.

C. Block 51, INPCHN

Block 51 contains the general integer input data for selecting general channel-dependent input options. This includes the number of segments and nodes per axial zone, correspondence of material properties with channel assignment, and selection of appropriate temperature nodes for plotting output. Much of the input is generated in connection with the channel geometry, which is covered in Section III.A, with reference to Fig. B-2.

Location, Input Variable

- #4 NPLN Number of segments in the fission gas plenum.
= 6
- #5 NREFB Number of reflector zones below the pin section.
= 3
- #6 NREFT Number of reflector zones above the pin section.
= 1
- #7-11 NZNODE(KZ) Number of segments in zone KZ. (KZ=5 for metal core)
KZ	Segments
1	2
2	2
3	3
4	30
5	4
- #14 NT Number of radial temperature nodes within the fuel.
(4 ≤ NT ≤ 11). Same as oxide;
= 11
- #15 IFUELV Table number of property table to be used from Block 13 for core fuel.

Not to Scale

Fig. B-2. SAS4A Axial Zones and Fuel Nodes for the Metallic Fuel Core.

		Property Table	Channel Number
		2	1
		2	2
		4	3
#16	IFUELB	Table number of property table to be used from Block 13 for axial blanket fuel.	
		Property Table	Channel Number
		5	1
		5	2
		5	3
#17	ICLADV	Table number of property table to be used from Block 13 for cladding table. This is the same for all channels. = 1	
#25	NPIN	Number of pins per subassembly. = 271 for driver assemblies. = 169 for blanket assemblies.	
#26	NSUBAS	Number of subassemblies in the channel.	
		Channel	Subassemblies
		1	6
		2	390
		3	163
#27	MZUB	Number of segments in upper blanket. = 2	
#28	MZLB	Number of segments in lower blanket. = 2	
		The following 5 values represent the axial nodes for which output is sent to the plotting file for post processing. The values do not have any effect on the calculation, or on the values typically plotted from File 11, the standard plot file.	
#90	JCLN	Axial heat-transfer segment for fuel temperature information. This is usually taken as the top node in the core fuel region. = 21 for the metallic core.	
#91	JNEN	Axial coolant node for clad temperature. This node is selected as the same node as JCLN, Loc. 90, but the node number is based on the coolant node structure. = 28 for the metallic core.	

#92	JHCN	Axial coolant node for coolant temperature. This node is selected as the same node as JCLN, Loc. 90, but the node number is based on the coolant node structure. = 28 for the metallic core.
#93	JNSN	Axial coolant node for structure temperature. This node is selected as the same node as JCLN, Loc. 90, but the node number is based on the coolant node structure. = 28 for the metallic core.
#94	JRPRO	Axial heat transfer segment for the total radial profile for the mass averaged fuel temperature. This is selected as the same node as JCLN, Loc. 90. = 21 for the metallic core.
#118	IEQMAS	Radial fuel nodes set up on an equal mass basis. = 1
#181	IAXEXP	Calculate axial expansion feedback in each channel. = 1
#203	IDKCRV	Decay heat curve is set the same for all channels. = 1

III. Core Geometry

The metal core designs were developed based on the general reactor specifications and constraints summarized in Ref [B2]. In addition, the core designs were required to be interchangeable in the following sense. The designs have the same assembly pitch, the same number and location of control rods, the same overall assembly length, and the same radially heterogeneous core layout with only minor adjustments. A description of the metal core layout is given in Fig. B-1. The radially heterogeneous layout has 396 drivers, 163 internal blankets, 90 radial blankets, and 36 control rods (24 primary and 12 secondary).

The assembly and pin design parameters for the metal core are given in Table I in the main report. The U-15Pu-10Zr fueled driver pin has a 0.285 in. od, a 0.022 in. thick HT-9 cladding, and a 75.0% smeared fuel density. The U-10Zr fueled blanket pin has a 0.392 in. od, also a 0.022 in. thick HT-9 cladding, and a 85.0% smeared fuel density. The metal core subassembly axial geometry is described in Table II of the main report. The driver pin has a 36 in. high active fuel region with a 52 in. upper fission gas plenum, and the overall assembly length is 185 in.

The metal core SAS4A analysis uses the same three channel assignment strategy as described for the oxide core. Channel one represents the "lead" assembly - the one that will exhibit the highest temperature during any transient. Channel two includes all the remaining driver assemblies, and channel three includes all internal blanket assemblies. Figure B-2 shows the SASSYS channel axial description. The metal core has five axial zones: a lower reflector with 2 axial nodes; a lower space zone with 2 axial nodes; a fuel pin zone with 6 nodes for the upper plenum, 2 nodes for the lower fuel expansion

space, 20 nodes for the active core, and 2 nodes for the upper fuel expansion space; and an upper reflector zone with 4 nodes. The internal blankets have the same zoning and noding as the drivers.

A. Block 61, GEOMIN

This input block contains all of the geometrical data required by the channel models in SAS4A. The input data are based on the information in Tables I and II in the main report. The channel models assume a single pin geometry for the calculation, which extends from the bottom to the top of the subassembly. The single pin is divided into a number of axial zones, where one zone represents the fuel and fission gas plenum. The data defining the zones in the channels is in Block 51. Many of the calculations are identical to the ones for the oxide core, and these are indicated. The other calculations are as follows:

Location, Input Variable

#1-5	ACCZ(KZ)	Same as the oxide fuel design, although there are 5 axial zones in the metallic fuel core.
#8-31	AHXI(J)	The length of each axial node in the zone representing the fuel and blankets. This is the same axial node structure used for generating the neutronics data. According to Block 51, Loc. 4 and 10, there are 24 axial nodes for the fuel. In the metallic core, this extends beyond the fuel both above and below, providing a region for axial fuel expansion which is included in the neutronics calculation. The lower 4 nodes and the upper 5 nodes are not real axial blankets, but are modelled using sodium to provide heat capacity.
	AHXI(J)	= 2 @ 5.08000d-02, 20 @ 4.57200d-2, and 2 @ 5.08000d-02
#32-36	DHZ(KZ)	Same as for the oxide fuel design.
#39-43	DSTIZ(KZ)	Same as for the oxide fuel design.
#46-50	DSTOZ(KZ)	Same as for the oxide fuel design.
#53	PLENL	The length of the fission gas plenum is given in Table II in the main report. 52.0 in. but 4 in. was taken to provide a fuel expansion region for the neutronics calculations; see Section II.B in the main report. Therefore, PLENL = 1.21920 d+00 m (48.0 in) Metal
#54	RBR(J)	Same as for the oxide fuel design.
#78	RER(J)	Same as for the oxide fuel design.
#102	RBRPL	Same as for the oxide fuel design.
#103	RERPL	Same as for the oxide fuel design.

#104 RINFP(J) Same as for the oxide fuel design.

#128 ROUTFP(J) Same as for the oxide fuel design.

#152-156 ZONEL(KZ) The length of each zone KZ in the single pin model is shown in Fig. B-2. Starting from the bottom of the assembly, with all lengths in m;

KZ	Metal
1	4.44500 d-01
2	4.57200 d-01
3	5.96900 d-01
4	2.33680 d+00
5	8.63600 d-01

#159-164 SRFSTZ(KZ) Same as for the oxide fuel design.

#166 AREAPC Same as for the oxide fuel design.

#169-174 DRFO(KZ) Same values as for the oxide fuel design;

#189-194 DRFI(KZ)

Drivers:

KZ	reflector thickness		
	inner node	outer node	
1	9.04875 d-04	9.04875 d-04	reflector
2	2.79400 d-04	2.79400 d-04	space
3	9.04875 d-04	9.04875 d-04	reflector
4		5.58800 d-04	fuel pin
5	9.04875 d-04	9.04875 d-04	reflector

Blankets:

KZ	reflector thickness		
	inner node	outer node	
1	1.24460 d-03	1.24460 d-04	reflector
2	2.79400 d-04	2.79400 d-04	space
3	1.24460 d-03	1.24460 d-03	reflector
4		5.58800 d-04	fuel pin
5	1.24460 d-03	1.24460 d-03	reflector

#180 RBRO Same as for the oxide fuel design.

#181 RERO Same as for the oxide fuel design.

#182-186 SER(KZ) Same as for the oxide fuel design.

IV. SAS4A Reactivity Worth Tables

The neutronics information used in the SAS4A codes is generated using a separate set of computer codes [B2]. The discussion in Ref. B2 can be summarized as follows, highlighting the overall features of the process.

The neutronics analysis on the interchangeable metallic and oxide heterogeneous core designs established the performance parameters needed for safety evaluations. ENDF/B-V.2 basic cross section data was used and processed through the MC2-2/SDX code system to generate broad-group libraries. A nine group cross section library was generated for use in depletion and rod worth calculations and a 21 group library was generated for determining reactivity coefficients. Global reactivity coefficients were evaluated for end-of-equilibrium cycle conditions.

The prompt neutron lifetime, beta-effective, and delayed neutron fractions were calculated using ENDF/B-V.2 delayed neutron data with 21 group flooded real and adjoint fluxes from a 3D hex-z finite difference model. All reactivity worth coefficients were calculated using 21 group cross-section libraries, a 3D hex-z finite difference model, and appropriate flooded or voided real and adjoint fluxes.

The sodium density worth was calculated using first order perturbation theory with flooded real and adjoint fluxes and flooded microscopic cross sections. The sodium void worth coefficient was calculated with the perturbation defined by voiding the flowing sodium from the core and axial blankets. Both the sodium number densities and the cross sections were changed to reflect changes in spectrum and self-shielding. The leakage term was treated rigorously.

The Doppler reactivity worths were calculated for both flooded and voided conditions. The flooded Doppler uses flooded real and adjoint fluxes and cross-sections reflecting full sodium density; while the voided Doppler used fluxes and cross-sections reflecting the voiding of flowing sodium in the core and axial blankets. The base case temperatures for the fuel isotopes was 850 K for the metallic core. The Doppler reactivity worths were determined by perturbing the cross-sections to reflect fuel temperatures of 1700 K for the metallic fuel. The Doppler reactivity worths for the fuel pin clad and structural duct wall were also calculated. The base case temperature for the structure was 750 K for the metallic fuel design and the perturbation temperature was 1500 K.

The fuel and clad material worths were also calculated using first order perturbation theory for both flooded and voided conditions. Appropriate fluxes and cross-sections were used with uniform changes in associated material densities. Radial and axial expansion reactivity loss coefficients were computed by eigenvalue differences using a 9 group cross-section library and a hex-z nodal model. The radial expansion reactivity loss was calculated for a uniform radial dilation of all core materials, then was normalized by dividing by the change in radius of the core/radial blanket interface. The axial expansion reactivity loss was calculated for a uniform axial dilation of the driver, then normalized by dividing by the change in core fuel height.

Control rod bank differential worths were calculated by eigenvalue difference at their BOEC and EOEC full power operating positions. A special

hex-z model with a fine axial mesh near rod tips was used with 9-group cross section libraries. These coefficients account for the control rod driveline expansion in ATWS accidents.

The information created by these calculations is converted into a form suitable for input to SAS4A [B2]. The data from this process is then grouped into the channel format as described in Section I.

A. Block 12, POWINA

The input data for input block 12 is summarized in this section. The input data support the discussion in Section II.A.2 on establishing the core modelling. Most of the data comes from Ref. B2, and was included in Ref. 5 of the main report which contains a complete description of the neutronics calculations and results for both metallic and oxide fuel designs. The resulting input for Block 12 is as follows, where only non-zero entries are indicated unless a zero value has a particular meaning:

Location, Input Variable

#2 GENTIM The prompt neutron lifetime is obtained from Ref. B2, as follows;
= 3.19000d-07 sec

#3 POWTOT The total reactor power is the same for both reactors at 3500 MWth;
= 3.50000d+09, W

#4-9 BETADN(L) The effective delayed neutron fraction for the delayed neutron precursor families. For this study, six delayed neutron precursor families were used, with the data obtained from Ref. B2 for each family with the following delayed neutron fractions;

Family Metallic Core

1	7.81930d-05
2	7.27030d-04
3	6.42330d-04
4	1.27560d-03
5	6.00940d-04
6	1.91170d-04

#10-15 DECCON(L) The decay constant for each delayed neutron precursor family, obtained from Ref. B2, with units of 1/sec.

Family Metallic Core

1	1.29920d-02
2	3.14110d-02
3	1.35590d-01
4	3.47520d-01
5	1.38070d+00
6	3.79190d+00

- #69 FRPR The fraction of total reactor power represented by the sum of all SAS4A channels. For each reactor, this is the fraction of the total power contained in the driver and internal blanket subassemblies. The data for performing this calculation were also obtained from Ref. B2. The result is $9.66367\text{d-}01$ for the metallic fuel core.
- #70 FRFLOW The fraction of the total reactor flow represented by the sum of all SAS4A channels was calculated from data contained in Ref. B3, which discusses the core orificing in detail; see Blocks 18 and 64. Again, this represents the fraction of the total flow that goes through the driver and internal blanket subassemblies, as follows:
= $9.12206\text{d-}01$ for the metallic fuel core.
- #71 CRDLEN The length of the control rod drivelines washed by sodium leaving the active core region. As described in Section II.A.3.c, this length is measured from the top of the active fuel region in the core to the free surface in the outlet plenum, with the following results;
= 6.2992 m for the metallic fuel core.
- #72 CRDEXP The thermal expansion coefficient of the control rod drivelines is the suggested value in SAS4A, which assumes that the drivelines are made of 316 stainless-steel. The same value is used for both metallic and oxide cores;
= $2.00000\text{d-}05$ per K.
- #73 ACRDEX The control rod driveline expansion reactivity feedback coefficient, where reactivity change = $\text{ACRDEX} \times \text{change in length} + \text{BCRDEX} \times (\text{change in length})^2$. The coefficients are listed in Table III in the main report from Ref. B3, as follows:
= $-11.60\text{ } \$/\text{m}$ for the metallic fuel core.
- #74 BCRDEX Not used in the current model.
- #75 CRDMC The control rod driveline mass times specific heat, in J/K, is obtained from the suggested value in SAS4A, 5.6×10^4 for a control rod driveline 6.0 m in length. Assuming that the mass per unit length is the same, this value is factored according to the length given in Loc. 71 for the metallic core design:
 $(6.2992/6.0) \times 5.6 \times 10^4 = 5.87927 \times 10^4$, metallic core.
- #76 CRDHA The control rod driveline surface area times the average heat-transfer coefficient, J/deg K, is obtained from the suggested value in SAS4A of 2300.0 for a control rod driveline 6.0 m in length. This value is factored according to the driveline length in Loc. 71:
 $(6.2992/6.0) \times 2300.0 = 2414.70$ for the metallic core.

- #77 UIVOL The coolant volume in the upper internal structure region is determined from the suggested value of 25.0 m³ given for a control rod driveline 6.0 m long in SAS4A. Factoring according to the control rod driveline length in Loc. 71:
 $(6.2992/6.0) \times 25.0 = 26.2467$ for the metallic core.
- #260-265 BETADK
 (L,IPW) The decay heat precursor yield for each group used to represent the decay heat curve is taken from previous work on a similar fuel, with the same values for both fuel types as follows:
L	BETADK
1	2.20054d-02
2	1.90590d-02
3	9.93659d-03
4	5.25396d-03
5	1.93225d-03
6	1.67409d-03
- #290-295 DKLAM
 (L,IPW) The decay heat group decay constant for each group in the decay heat curve is also taken from previous work on a similar fuel, with the same values for both fuel types as follows:
L	DKLAM
1	6.56251d-02
2	1.77911d-03
3	1.37667d-04
4	6.08268d-06
5	2.25624d-07
6	1.89684d-08
- #320 BETAHT
 (IPW) The total decay heat precursor yield, which is simply the sum of the values for BETADK, Loc. 260-265. Since the same yields are used for both fuel types, the sum is the same for both:
 $= 6.22500d-02$
- #325 POWLVL
 (L,IPW) The normalized power history for initializing the decay heat is assumed to be constant operation at nominal conditions for one cycle, and is the same for both fuel types. There is only one entry in the table,
 $= 1.00000d+00$
- #365 POWTIM
 (L,IPW) The time at power level POWLVL given in Loc. 325-364 is the time for one complete cycle, 80% of 365 days, or 292 days. Converting this to seconds,
 $= 2.52288 \times 10^7$ sec.

#408	SLLMAX	The maximum allowable slope of subassembly at grid plate with respect to vertical based on subassembly nozzle/grid plate clearances = 1.00. See App. C.
#409	PITCHG	The subassembly pitch at the grid plate at the reference temperature = 0.154827 m. See App. C.
#410	PITCHA	The flat-to-flat dimension across the above-core load pad at the reference temperature = 0.154887 m. See App. C.
#411	PITCHT	The flat-to-flat dimension across the top load pad at the reference temperature = 0.154887 m. See App. C.
#412	RDEXCF	The radial expansion reactivity feedback coefficient for a uniform core dilation, \$/m. See App. C. = -76.5 \$/m for the metallic fuel core
#413	TLPRRC	The clearance between the top load pads and the restraint ring. See App. C. = 1.20000d-03 m for the metallic fuel core.
#414	BNDMM1	The applied bending moment at the top of the core region, representing the flat-to-flat temperature difference in this region for subassemblies at the outer edge of the active core. See App. C. = 1.00000d-03
#415	BNDMM2	The applied bending moment in the region above the core, representing the flat-to-flat temperature difference in this region for subassemblies at the outer edge of the active core. See App. C. = 1.00000d-03
#423	FCDTR2	The nominal steady-state top restraint ring temperature, expressed as a fraction of the average coolant temperature rise through the core. See App. C. = 1.00000d-01
#424	FCDTRF	The nominal steady-state reflector load pad temperature, expressed as a fraction of the coolant temperature rise through the core. See App. C. = 3.50000d-01
#426	CRSAC	Additional clearance in the interior of the core, or the difference between the actual core radius and the ideal core radius. See App. C. = 1.62400d-03
#428	RR2TC	The thermal response time constant for the top restraint ring. See App. C. = 5.00000d+02

B. Block 62, POWINC

This input block contains all of the channel dependent neutronics input data. The majority of the data comes from the combination of the subassembly-by-subassembly data for the core, according to the channels as described in Section I. This block also includes other data to account for differences in irradiation between the channels, but for this study, all assemblies were assumed to have an irradiation of 1 cycle. The data is originally generated for 63 subassemblies in a 1/12 symmetry representation of the reactor core. Each core position then represents a number of identical subassemblies around the core, which in this case is usually either 6 or 12, forming a "channel", not to be confused with a SAS4A channel.

Location, Input Variable

#2	GAMSS	The fraction of total power used in direct heating of the structure. A representative value taken from previous work is used; = 7.8000 d-03
#4	GAMTNC	The fraction of total power used in direct heating of the coolant. A representative value taken from previous work is used; = 4.8000 d-03
#5	GAMTNE	The fraction of total power used in direct heating of the cladding. A representative value taken from previous work is used; = 1.14000 d-02
#6-29	PSHAPE(I)	The ratio of pin power at axial node J to the power in the peak axial fuel pin segment for all nodes in the core. The values are calculated from the combination of the data for the 63 individual "channels" into the 3 SASSYS/SAS4A channels. Ref. B2 contains the discussion on generating the assembly data. The values are in the listing of the input data in Appendix G for the metallic fuel core.
#30-40	PSHAPR(I)	The normalized radial power shape within the pin by radial node, I. A flat power profile is assumed for the oxide fuel pin, so a value of 1.0 is used for all nodes; = 1.00000 d+00 for all 11 radial nodes.
#45	PLIN(M)	The relative power level at the end of the M steady-state division. This input is used by DEFORM-4 to calculate the appropriate conditions in the pin at the start of the transient. Since DEFORM-4 is only used with oxide fuel, a single value of 1.0 is sufficient; = 1.00000 d+00
#62	ADOP	The Doppler coefficient for the channel when the channel is not voided. This is also obtained from the combining of the "channel" data [B2], with the totals as listed in Table III In the main report. The units are ($\Delta k/k$).

		Channel	Metallic Fuel
		1	-4.77626d-05
		2	-1.88176d-03
		3	-1.72639d-03
#63	BDOP	The Doppler coefficient for the channel when the channel is fully voided. The data is obtained from the combining of the "channel" data, as with ADOP, Loc. 62. The units are (delta k/k).	
		Channel	Metallic Fuel
		1	-2.95938d-05
		2	-1.12517d-03
		3	-1.17952d-03
#64-87	WDOPA(J)	The Doppler weighting factor for the case where the sodium coolant is still present in the subassembly. This factor is generated for each axial node within the fueled region of the pin. The values are calculated from the combining of the data for the 63 "channels" into 3 channels. The result is listed in Appendix G as part of the input deck.	
#112-135	VOIDRA(J)	The first-order coolant void reactivity worth is calculated by the combination of the data from the 63 "channels" into the 3 SAS4A channels. The result is listed in Appendix G.	
#160-183	CLADRA(J)	The first-order cladding reactivity worth is calculated by the combination of the data from the 63 "channels" into the 3 SAS4A channels. The result is listed in Appendix G.	
#208-231	FUELRA(J)	The first-order core fuel reactivity worth is calculated by the combination of the data from the 63 "channels" into the 3 SAS4A channels. The result is listed in Appendix G.	
#256	PRSHAP	The ratio of the power per subassembly averaged over this channel to the power per subassembly averaged over all channels is calculated as part of the process combining the data from the 63 "channels" into the 3 SAS4A channels. The value is normalized to give a value of 1.0 when summed over the three SAS4A channels weighted by the number of subassemblies in each channel.	

V. Thermophysical Properties

The input data for the thermophysical properties of the core materials is contained in two input data blocks, Block 13 for the basic data, and Block 63 for any additional data specific to each channel and not covered in Block

13. Usually this is applied to certain model options, such as the force balance model for calculating the axial expansion of the fuel and cladding. Some of the properties are modified before being used as input data in order to be consistent with the models in SAS4A. The original data and any modifications are covered in detail for all of the properties needed for the metal fuel core model.

A. Block 13, PMATCM

Block 13 contains general channel independent fuel and cladding properties. The metallic fuel properties were taken from the Metallic Fuels Handbook, [B4]. The cladding properties for HT-9 are taken from the Nuclear Systems Materials Handbook [B5].

Cladding Tables

11-28 Table 1: HT-9 Cladding Thermal Conductivity
71-88 Temperatures for Cladding Table 1, deg K.

Fuel Density Tables

91-103 Table 1: U-15Pu-10Zr, Theoretical Density
#111-123 Table 2: U-15Pu-10Zr, Fuel Pin Smear Density (75.0%)
#131-140 Table 3: U-10Zr, Theoretical Density
#151-160 Table 4: U-10Zr, Blanket Pin Smear Density (85.0%)
#171-180 Table 5: Na, Theoretical Density

#251-263 Temperatures for Fuel Density Table 1, deg K
#271-283 Temperatures for Fuel Density Table 2, deg K
#291-300 Temperatures for Fuel Density Table 3, deg K
#311-320 Temperatures for Fuel Density Table 4, deg K
#331-340 Temperatures for Na Density Table 5, deg K

Fuel Conductivity Tables

#420-429 Table 1: U-15Pu-10Zr, Fuel Conductivity
#440-449 Table 2: U-15Pu-10Zr, Fuel Conductivity
#460-469 Table 3: U-10Zr, Blanket Fuel Conductivity
#480-489 Table 4: U-10Zr, Blanket Fuel Conductivity
#500-509 Table 5: Sodium Conductivity
#580-589 Temperatures for Fuel Conductivities, deg K

Fuel Specific Heat Tables

#606-625 Table 1: U-15Pu-10Zr, Fuel Specific Heat
#626-645 Table 2: U-15Pu-10Zr, Fuel Specific Heat
#646-665 Table 3: U-10Zr, Blanket Fuel Specific Heat
#666-685 Table 4: U-10Zr, Blanket Fuel Specific Heat
#686-705 Table 5: Liquid Sodium Specific Heat
#766-785 Temperatures for Fuel Specific Heat, deg K

Cladding Property Tables

#819-830 HT-9 Cladding Specific Heat
 #879-890 Temperatures for Cladding Specific Heat, deg K
 #990-1001 HT-9 Cladding Specific Heat * Density
 #1050-1061 Temperatures for Cladding Specific Heat * Density, deg K

Location, Input Variable

#11-28 EXKTAB Thermal conductivity of HT-9 cladding, W/m-K. Cladding
 (I,ICLAD) properties taken from Nuclear Systems Materials Handbook
 [B5].

#71-88 EXKTM(L) Cladding property temperatures, deg K.

Temperature deg K	HT-9 Thermal Conductivity W/m-K
294.27	24.5
323.00	24.7
373.00	24.9
423.00	25.2
473.00	25.4
523.00	25.6
573.00	25.9
623.00	26.1
673.00	26.3
723.00	26.5
773.00	26.8
823.00	27.0
873.00	27.2
923.00	27.4
973.00	27.6
1023.00	27.8
1073.00	28.0
1088.72	28.0

#91-180 RHOTAB Theoretical fuel density, Kg/m**3. Metal fuel properties
 (L,IFUEL) taken from Metallic Fuels Handbook, [B4], and sodium
 properties taken from Ref. B1. U-15Pu-10Zr theoretical densities
 were adjusted by 75% for Table 2 and U-10Zr densities were adjusted
 by 85% for Table 4.

#251-340 RHOTEM Temperatures for fuel density tables, deg K.
 (L,IFUEL)

Fuel Densities Kg/m**3

Temperature deg K	U-15Pu-10Zr		U-10Zr	
	100% TD	75% TD	100% TD	85% TD
293.0	15800.0	11850.0	16020.0	13617.0
400.0	15720.0	11790.0	15950.0	13558.0
500.0	15630.0	11723.0	15890.0	13439.0

600.0	15550.0	11663.0	15810.0	13439.0
700.0	15470.0	11603.0	15730.0	13371.0
800.0	15390.0	11543.0	15640.0	13294.0
868.0	15330.0	11498.0		
900.0			15540.0	13209.0
938.0	15170.0	11378.0		
1000.0	15110.0	11333.0	15220.0	12937.0
1100.0	15020.0	11265.0	15120.0	12852.0
1200.0	14930.0	11198.0	15020.0	12767.0
1300.0	14850.0	11138.0		
1378.0	14780.0	11085.0		

Temperature deg K	Sodium Density Kg/m**3
550.0	883.24
650.0	860.53
750.0	837.59
850.0	814.43
950.0	791.06
1050.0	767.46
1150.0	743.64
1250.0	719.60
1350.0	695.33
1450.0	670.85

- #419 TR Reference design point temperature at which pin dimensions are measured, deg K.
= 300.0
- #420-509 XKTEB (L,IFUEL) Fuel thermal conductivity, W/m-K. The metal fuel thermal conductivity for U-15Pu-10Zr at 100% TD is taken from Table C.1-2 and that for U-10Zr at 100% is taken from Table C.1-1 of the Metallic Fuels Handbook, [B4]. For smeared densities of 75% and 85% the thermal conductivities are adjusted using the following correction for porosity.
- $$K/K_s = 1 - P_v^{2/3} - P_{na}^{2/3} + P_{na}^{2/3} / \{ (K_s/K_{na}) * P_{na}^{1/3} + (1 - P_{na}^{1/3}) \}$$
- where:
- K = Corrected fuel thermal conductivity
Ks = Fuel thermal conductivity, 100% TD
Kna = Sodium thermal conductivity
Pv = Gas filled porosity
Pna = Sodium filled porosity

Assuming that 25% of the porosity becomes logged with sodium and 75% remains voided after irradiation, for a typical fuel temperature of 1000 K,

$K_{na} = 58.31 \text{ W/m-K}$,
 $K_s = 20.0 \text{ W/m-K}$ for U-15Pu-10Zr, or
 $K_s = 34.44 \text{ W/m-K}$ for U-10Zr.

Then:

for 25% porous U-15Pu-10Zr $K/K_s = 0.728$
 and for 15% porous U-10Zr $K/K_s = 0.785$.

Metal Fuel Thermal Conductivity, W/m-K.

Temperature deg K	U-15Pu-10Zr		U-10Zr		Sodium
	100% TD	75% TD	100% TD	75% TD	
293	9.8	7.13	16.4	12.87	92.63
373	11.4	8.30	18.0	14.13	88.18
473	13.4	9.76	20.1	15.78	82.84
573	15.5	11.28	22.4	17.58	77.74
673	17.7	12.89	24.9	19.55	72.87
773	20.1	14.63	27.6	21.67	68.20
873	22.6	16.45	30.5	23.94	63.73
973	25.3	18.42	33.6	26.38	59.44
1073	27.8	20.24	36.8	28.89	55.31
1173	30.6	22.28	40.3	31.64	51.33

- #606-701 CPFTAB (L, IFUEL) Fuel specific heat, J/Kg-K. Fuel specific heat properties taken from Metallic Fuels Handbook, [B4], and sodium properties taken from Ref. B1.
- #766-785 CPFTAB (L) Temperatures for fuel specific heat tables, deg K.

Metal Fuel Specific Heat, J/Kg-K.

Temperature deg K	U-15Pu-10Zr		U-10Zr		Sodium
	100% TD	75% TD	100% TD	75% TD	
298	133.04	133.04	150.70	150.70	1282.41
400	146.20	146.20	153.14	153.14	1282.41
500	158.87	158.87	169.72	169.72	1282.41
600	172.02	172.02	191.18	191.18	1282.41
700	185.18	185.18	216.05	216.05	1272.42
800	197.85	197.85	241.90	241.90	1265.75

873	207.60	207.60	261.29	261.29	1263.09
890	191.36	191.36	265.80	265.80	1262.47
923	159.84	159.84	239.76	239.76	1262.40
1000	169.59	169.59	178.99	178.99	1263.22
1100	182.26	182.26	178.99	178.99	1267.95
1200	194.93	194.93	178.99	178.99	1277.95
1300	208.08	208.08	178.99	178.99	1291.59
1379	217.83	217.83	178.99	178.99	1307.07
1400	217.68	217.68	178.99	178.99	1311.86
1487	217.08	217.08	178.99	178.99	1335.14
1588	216.37	216.37	203.62	203.62	1370.16
1600	216.37	216.37	206.54	206.54	1374.99
1657	216.37	216.37	220.44	220.44	1400.13
1700	216.37	216.37	220.44	220.44	1421.81

#786-791	TFSOL	Fuel solidus temperature, deg K. 1379.0 for U-15Pu-10Zr 1487.0 for U-10Zr 370.0 for Sodium
#794-799	TFLIQ	Fuel liquidus temperature, deg K. 1588.0 for U-15Pu-10Zr 1657.0 for U-10Zr 380.0 for Sodium
#802-807	UFMELT (IFUEL)	Fuel Heat of fusion, J/Kg. 9.16000D+04 for U-15Pu-10Zr 9.02000D+04 for U-10Zr 1.13000D+05 for Sodium
#810	TESOL (ICLAD)	Cladding solidus temperature, deg K. 1690.15 for HT-9
#813	TELIQ (ICLAD)	Cladding liquidus temperature, deg K. 1710.15 for HT-9
#816	UEMELT (ICLAD)	Cladding heat of fusion, J/Kg. 2.68300D+05 for HT-9
#819-830	CPCTAB	Cladding specific heat, J/Kg-K.
#879-890	CPCTAM	Temperatures for cladding specific heat tables, deg K.

HT-9	
Temperature deg K	Specific Heat J/Kg-K
373.0	482.50
473.0	521.48
573.0	563.37
673.0	615.38
773.0	684.68
873.0	784.96
923.0	866.08

973.0	1021.50
998.0	1021.00
1023.0	868.34
1073.0	747.97
1088.6	762.14

#990-1002 CROETB Specific heat * density for HT-9 cladding, J/m**3-K.
(L)

#1050-1061 CROETM Temperatures, deg K.
(L)

Temperature deg K	HT-9 Specific Heat * Density J/m**3-K
373.0	3.73566D+6
473.0	4.02379D+6
573.0	4.33085D+6
673.0	4.71178D+6
773.0	5.22021D+6
873.0	5.95863D+6
923.0	6.55969D+6
973.0	7.71948D+6
998.0	6.54728D+6
1023.0	5.62698D+6
1073.0	5.72962D+6
1088.6	5.72962D+6

#1086 RGASSI Ideal gas constant, J/gram-mole.
= 8.31434

B. Block 63, PMATCH

This input block contains thermophysical properties for the subassemblies represented by each channel that are not already input in Block 13, PMATCM. Only non-zero values are reviewed, unless a value of zero has a special meaning.

Location, Input Variable

#2 AHBPAR Same as oxide, but the value is not used in the calculation. Its only function is to provide a non-zero value.

#3 BHBPAR Same comment as for AHBPAR.

#4 CHBPAR Same comment as for AHBPAR.

#5	HBMAX	The maximum bond conductance when a gap exists, and the minimum when a gap does not exist is set to $1.00 \times 10+6$ for metallic fuel; HBMAX = $1.0 \times 10+6$.
#6	HBMIN	The minimum value of bond conductance at any time is set to $1.00 \times 10+6$ for irradiated metallic fuel.
#7	HBPAR	Set to 0.0286, but not used in the calculation.
#11-15	XKSTIZ	The thermal conductivity of the inner structure node by axial zone is set to 27.2 W/mK based on the properties of HT-9 at the operating conditions.
#18-22	XKSTOZ	The thermal conductivity of the outer structure node by axial zone is set to 27.2 W/mK based on the properties of HT-9 at the operating conditions.
#25	DEL	The product of Stefan-Boltzmann's constant and the emissivity of the cladding is $4.818 \times 10-8$ based on Block 13, Loc. 1107 and 1109; DEL = $4.818 \times 10-8$.
#27	POGAS	The initial gas pressure in the fuel pin at the reference temperature TR (Block 13, Loc. 419) is assumed to be atmospheric pressure; POGAS = $1.01325 \times 10+5$.
#28-32	XKRF	The thermal conductivity of the reflector by axial zones is set to 27.2 W/mK based on the properties of HT-9 at the operating conditions.
#35	DENSS	The density of the cladding at the reference temperature TR is 7760 kg/m ³ based on the properties of HT-9.
#36	COOLDN	The coolant density used in the calculation of the coolant void reactivity worth is set to 845 kg/m ³ based on comparison with the void worth listed in Table II.A.iii.1.
#37-41	RHOCSI	The product of density and heat capacity for the inner structure is $5.9856 \times 10+6$ based on the properties of HT-9 at the operating temperature.
#44-48	RHOCSO	The product of density and heat capacity for the outer structure is $5.9856 \times 10+6$ based on the properties of HT-9 at the operating temperature.
#51-55	RHOCCR	The product of density and heat capacity for the reflector is $5.9856 \times 10+6$ based on the properties of HT-9 at the operating temperature.
#58	RHO CG	The product of density and heat capacity for gas in the fission gas plenum is $5.57 \times 10+5$ based on previous work.
#59	RG	The thermal resistance of the gas in the fission gas plenum is set to 0.03 from previous work.

#73	FUELEX	The axial expansion coefficient of the fuel is set to 2.01×10^{-5} per K based on the available data for this metallic fuel composition at operating temperature.
#74	CLADEX	The axial expansion coefficient of the cladding is set to 1.40×10^{-5} per K based on the available data for HT-9 at operating temperature.
#75	YFUEL	The Young's modulus of the metal fuel is estimated at 2.80×10^9 based on available data at operating temperature.
#76	YCLAD	The Young's modulus of the cladding is estimated at 1.52056×10^{11} based on available data.
#77	FULREX	The fuel linear expansion coefficient is the same as FUELEX, Loc. 73; $FULREX = 2.01 \times 10^{-5}$.
#78	CLDREX	The cladding linear expansion coefficient is the same as CLADEX, Loc. 74; $CLDREX = 1.40 \times 10^{-5}$.
#79	EXPCFF	The effective multiplier for axial fuel expansion is set to 1.0 for nominal fuel expansion.

C. Block 65, FUELIN

The only input data in this block is one entry, Loc. 2, which must be input to the code. The suggested value of 0.2 is input. This value is only used for avoiding difficulties in job execution, but the calculations do not use this value for anything.

VI. Coolant Flow

The thermal-hydraulic analysis for the interchangeable metal and oxide core designs was performed using the multi-assembly, steady-state thermal-hydraulic code SUPERENERGY-2 [B3]. Hot channel subfactor data was taken from current work with appropriate changes made to account for the sodium bond used in the metal fuel. A separate thermal-hydraulic analysis was performed for each design, and no driver or blanket shuffling was employed [B3].

Nozzle orificing was used with the number of discriminator zones fixed at 10. Orificing zones were defined by grouping assemblies with respect to maximum value of power times skew over lifetime. Assemblies with similar values were placed in the same orifice zone, and the mixing of driver and blanket assemblies within a zone was not permitted. The core fuel-cycle calculations were performed using the equilibrium-cycle approach with the assembly powers adjusted by stage factors to obtain the actual power at a specific point in assembly life. Explicit gamma transport and gamma heating rates are included in the spatial power profiles used to predict temperatures. Thus the heating rate in reflector and control assemblies where gamma heating is an important contribution to assembly power is properly predicted.

A fixed percentage of the total available flow is allocated for other than fuel and blanket assemblies. Reflector assemblies received 1.4% of the flow, control assemblies 0.5%, and 1.4% was allocated for cold bypass flow. The remainder of the total flow is allocated among driver and blanket assemblies so as to minimize the peak cladding midwall temperature for all fueled orifice zones over all life. This orificing strategy yielded a peak two-sigma cladding midwall temperature of 946 K (1244 deg F) for the metallic core.

A. Block 14, PRIMIN

This input block contains data on the coolant flow and temperatures in the primary loop. Most of the data is used by PRIMAR-4, while the rest involves the boiling model. Only nonzero values are discussed.

Location, Input Variable

#1	PX	The coolant exit pressure at ZPLENU, Loc. 88, is calculated based on atmospheric pressure at the surface of the outlet plenum, with the elevations as shown in Fig. D-2. The pressure and elevation ZPLENU must be consistent with the PRIMAR-4 pressure and elevation in Block 18. For the current calculation, the value is 1.52599×10^5 Pa; $PX = 1.01325 \times 10^5 + (830 \text{ kg/m}^3)(9.807 \text{ m/sec}^2)(6.2992 \text{ m})$ $= 1.52599 \times 10^5 \text{ at the subassembly exit.}$
#45	TOTAB	Same as for oxide; TOTAB = 630.15.
#65	TOTME	Same as for oxide; TOTME = 0.0.
#85	TNTRY	Same as for oxide; TNTRY = 783.0.
#87	ZPLENL	The reference elevation for the core inlet plenum is set to the elevation of the bottom of the subassembly, ZPLENL = -1.49860 m.
#88	ZPLENU	The reference elevation for the core outlet plenum is set to the elevation of the top of the subassembly, ZPLENU = 3.20041 m.
#90	DZBGCL	Same as for oxide; DZBGCL = 0.10.
#91	DZBCGU	Same as for oxide; DZBCGU = 0.25.

B. Block 64, COOLIN

This input block contains all of the coolant flow information required by each channel. Most of the data is based on previous work, while the rest is from the current core orificing [B3].

Location, Input Variable

#1	AFR	The liquid slug friction factor coefficient is set to the current recommended value, 0.4089.												
#2	BFR	The liquid slug friction factor coefficient is set to the current recommended value, -0.25.												
#3-5	C1,C2,C3	The coefficients in the convective heat transfer correlation (Lyon-Martinelli) are set to the current recommended values: C1 = 0.0144 C2 = 0.86 C3 = 4.480												
#7	RELAM	The Reynolds number for the transition from the laminar friction factor correlation to the turbulent friction factor correlation is set to the the current recommended value of 1509.												
#8	AFLAM	The coefficient in the laminar friction factor correlation is set to the current recommended value of 99.												
#47	WO	The steady-state coolant flow rate per pin in the channel is obtained from the process of grouping the channels, and is the average of the flow rates of the assemblies in each channel, weighted by the number of assemblies in each orifice zone; <table data-bbox="588 1067 1136 1191"> <thead> <tr> <th>Channel</th><th>WO</th><th></th></tr> </thead> <tbody> <tr> <td>1</td><td>0.128237</td><td>in kg/sec/pin</td></tr> <tr> <td>2</td><td>0.117934</td><td>"</td></tr> <tr> <td>3</td><td>0.134990</td><td>"</td></tr> </tbody> </table>	Channel	WO		1	0.128237	in kg/sec/pin	2	0.117934	"	3	0.134990	"
Channel	WO													
1	0.128237	in kg/sec/pin												
2	0.117934	"												
3	0.134990	"												
#67-68	THETA1,2	The coefficients in the numerical scheme for determining the degree of implicitness. Both are set to 0.5.												
#69	DTLMAX	The maximum temperature change of the liquid coolant per time step is set to the suggested value of 15.												
#70	DTVMAX	The maximum temperature change of the coolant vapor per time step is set to the suggested value of 50.												
#71	DZIMAX	The maximum motion of the liquid/vapor interface per time step is set to the suggested value of 0.10.												
#72	HCOND	The condensation heat transfer conductance of the coolant is set to 63000 based on previous work.												
#73	SLMIN	The minimum initial length of a liquid slug is set to the suggested value of 0.02.												
#75	WFMIN	The minimum thickness of a coolant film on the cladding is set to the default value of 1.00×10^{-7} .												

#76	WFMIN	The same as WFMIN, 1.00×10^{-7} .
#77	WFS00	The same as WFO, Loc. 84; 1.427×10^{-4} .
#84	WFO	The initial film thickness is set to 0.15 of the total liquid volume fraction = 1.427×10^{-4} .
#165	DTS	The degrees of superheat required before any bubble formation is set to 10 K.

REFERENCES

- B-1 G. Birgersson et al., "The SAS4A LMFBR Accident Analysis Code System," ANL/RAS 83-38, Revision 2 (February 1988).
- B-2 E. K. Fujita and D. C. Wade, "The Neutronic and Fuel Cycle Performance of Interchangeable 3500 MWth Metal and Oxide Fueled LMRs," ANL-FRA-163 (March 1989).
- B-3 R. B. Vilim and R. N. Hill, "The Steady-State Thermal-Hydraulic Performance of Interchangeable 3500 MWth Metal and Oxide Fueled LMRs," ANL-FRA-164 (March 1989).
- B-4 G. L. Hofman et al., "Metallic Fuels Handbook," ANL-IFR-29 (November 1985).
- B-5 Nuclear Systems Materials Handbook, TID-26666, Hanford Engineering Development Laboratory.

APPENDIX C. RADIAL CORE EXPANSION REACTIVITY FEEDBACK MODEL

The study is using the detailed radial core expansion reactivity feedback model. This model requires additional input data which is not included in the neutronics information used in generating the data in Input Blocks 12 and 62; see Appendices A and B. This appendix covers the generation of all the required input data for both fuel types. The input variables and the data for each fuel type are described in detail.

I. Metallic Fuel Design

Some of the input data is used in Block 1, where all of the channel independent integer data for the core models is stored. The remainder of the input data is in Block 12. A schematic of the core restraint system model is shown in Fig. C-1. The calculation of the data for the metallic fuel design is as follows:

A. Block 1, INPCOM

Location, Input Variable

#36	IRADEX	This integer chooses the radial core expansion reactivity feedback model. Since the detailed model is to be used, and the grid plate temperature is calculated based on the inlet plenum wall temperature, IRADEX = -4.
#51	NSUBTC	The total number of subassemblies in the active core region, i.e., within and including the outermost row of drivers. There are 396 drivers, 163 internal blankets, and 36 control rods in this region, so NSUBTC = 595.
#52	MTGRD	The structural material for the support grid is assumed to be 316 stainless steel, so MTGRD = 1.
#53	MTACLP	The above-core load pad material is assumed to be the same as the hexcan duct material, HT-9, so MTACLP = 2.
#54	MTTLP	The top load pad material is assumed to be the same as the hexcan duct material, HT-9, so MTTLP = 2.
#59	NSUBTR	The total number of subassemblies in the reactor is the total number in the active core region, 595, plus the radial blankets 90, and the radial shields, 306; NSUBTR = 991.
#60	NRRNGS	There is only one restraint ring in the core design, at the top load pad elevation, so NRRNGS = 1.
#61	MTRRAC	Not used since there is only one restraint ring.
#62	MTRRT	The restraint ring is also assumed to be 316 stainless steel, so MTRRT = 1.

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Fig. C-1. Core Restraint System Model for the 3500 MWth Metallic and Oxide Fuel Core Designs.

- #63 MTRFAC All subassembly load pads are assumed to be the same as the duct material, HT-9, so MTRFAC = 2.
- #64 MTRFT As in Loc. 63, MTRFT = 2.
- #65 IROPT The assumption for low normalized power-to-flow ratios is that there is sufficient bending of the assemblies exterior to the active core region to continue to hold the assemblies together, even though the top of the average driver assembly in the outer row does not push against the restraint ring (through the top load pads of the radial blankets and shields). This is likely when B_4C shields are used due to the reverse radial temperature gradient, causing inward bending. Activating this option provides the most conservative estimate of the negative reactivity feedback during almost all accident scenarios. IROPT = 1.

B. Block 12, POWINA

Location, Input Variable

- #408 SLLMAX The grid plate design assumes that the subassemblies are free to tilt at the grid plate, so an arbitrarily large value is chosen; SLLMAX = 1.00. This would normally be calculated from design information on the grid plate to subassembly connection, but such data was not available.
- #409 PITCHG The subassembly pitch at the grid plate is given in Table I as 6.131 in. = 0.155727 m at power. This implies an inlet temperature of 630 K. The reference temperature is 300 K, and the mean thermal expansion coefficient for the grid plate between these two temperatures is 1.7525×10^{-5} per K. The grid plate pitch at the reference temperature is then 0.154827 m. This will result in the correct pitch at nominal full power conditions, so PITCHG = 0.154827.
- #410 PITCHA The flat-to-flat dimension of the subassembly load pads is not given in the design information. It is assumed that the design would provide for the pitch at the load pads to be the same as the pitch at the grid plate at normal full power conditions. This means the load pad pitch would be 6.131 in. = 0.155727 m at an average temperature of the load pads of 745 K. This is obtained by assuming that the load pad experiences only 75% of the coolant temperature rise through the core, i.e. $630 \text{ K} + 0.75 (783 - 630) \text{ K} = 745 \text{ K}$. The mean thermal expansion coefficient for HT-9 between the reference temperature of 300 K and the operating temperature of 745 K is 1.215×10^{-5} per K, so the load pad dimension at the reference temperature is 0.154887 m; PITCHA = 0.154887.
- #411 PITCHT The flat-to-flat dimension of the top load pad is the same as for the above-core load pad, Loc. 410, so PITCHT = 0.154887.

- #412 RDEXCF The radial expansion coefficient for the metallic fuel core design is given in Table III as $-76.5 \text{ } \$/\text{m}$, so $\text{RDEXCF} = -76.5$.
- #413 TLPRRC The clearance between the top load pads and the restraint ring is set based on the amount of thermal bending of the subassembly at a normalized power-to-flow ratio of 0.7. This is the power-to-flow ratio above which the core geometry should be well-defined to ensure controllability at nominal power throughout all of the operational cycles. Given all of the other values for the input variables for this model, a value of $1.20 \times 10^{-3} \text{ m}$ resulted in the transition to a well-defined geometry occurring at this normalized power-to-flow ratio, so $\text{TLPRRC} = 1.20 \times 10^{-3}$.
- #414 BNDMM1 The applied bending moment at the top of the core is based on the estimated flat-to-flat temperature difference in the radial direction at that elevation. This temperature difference is typically around 10 K, and this was used to estimate the thermal bending of the subassembly. Using the equation that the moment is equal to $\alpha \Delta T/D$, this yields about 9.6×10^{-4} , which was rounded-off to 1.00×10^{-3} . As more information becomes available, this estimate can be improved; $\text{BNDMM1} = 0.001$.
- #415 BNDMM2 The applied bending moment for the upper region is assumed to be the same as the one at the top of the core. There is considerable doubt as to the behavior of the flat-to-flat temperature difference in this region of the subassembly. However, the results from the model have proven to be not particularly sensitive to the value as long as the core geometry is well-defined above a power-to-flow ratio of 0.7. If it is believed that a lower value is appropriate, the top load pad and restraint ring clearance must be recalculated to reflect the change in value. Therefore, $\text{BNDMM2} = 0.001$.
- #419 DFLTCS All subassemblies are assumed to be straight at zero power, so $\text{DFLTCS} = 0.0$.
- #420 DFLTSS As for Loc. 419, $\text{DFLTSS} = 0.0$.
- #421 ACLPRC Not used as there is only one restraint ring in the design.
- #422 FCDTR1 Same as Loc. 421.
- #423 FCDTR2 The top restraint ring is assumed to experience only 10% of the coolant temperature rise through the core. While there is not any design information to evaluate the accuracy of this number, it should be stressed that the particular value used is largely unimportant for any transient results due to the long time constant associated with the ring thermal response. However, it is extremely important to have the value be consistent with the other input variables so the core loading behavior with power-to-flow ratio is correct.

If this input value is changed, others also need to be changed. For this case, $FCDTR2 = 0.10$.

- #424 FCDTRF The reflector region of the model is identified as all of the subassemblies exterior to the outermost row of driver assemblies. This includes radial blankets and radial shields in the present case. An estimate of 35% is made for the average assembly in the outer region, but the results are not particularly sensitive to this number and large errors can be tolerated. This input value must be in a consistent set with the other values in the model in order to ensure that the core configuration is well-defined above a power-to-flow ratio of 0.7; $FCDTRF = 0.35$.
- #425 DRCOLL Not used for the present case.
- #426 CRSAC The additional clearance remaining in the interior of the core is due to the non-ideal packing of the subassembly load pads as the subassemblies are pushed together during the ascent to power. As with some other input variables in this model, the particular value is not important, only its relation to several of the other input variables. The value used in this study is based on the measured value in a full-scale simulation of the FFTF core load pads, and factored based on the relative number of assemblies in the two cores: $6.35 \times 10^{-4} \sqrt{(595/91)} = 1.624 \times 10^{-3} \text{ m}$, since the core size is proportional to the square root of the number of assemblies; $CRSAC = 1.624 \times 10^{-3}$.
- #427 RR1TC Not used as there is only one restraint ring.
- #428 RR2TC Estimated at 500 seconds, so $RR2TC = 500$. This value does not have any appreciable impact on the near term results for transients, but can be involved in the details of the long-term response.

II. Oxide Fuel Design

Much of the input data is the same for the oxide fuel core. However, due to the different location of the core within the subassembly, as shown in Fig. C-1 some of the input is changed. The values which are changed for the oxide design are indicated as follows:

A. Block 1, INPCOM

Location, Input Variable

- #36 IRADEX Same, $IRAD\bar{E}X = -4$.
- #51 NSUBTC Same total, but 402 drivers, 157 internal blankets, and 36 control subassemblies; $NSUBTC = 595$.
- #52 MTGRD Same, $MTGRD = 1$.

#53	MTACLP	Same, MTACLP = 2.
#54	MTTLP	Same, MTTLP = 2.
#59	NSUBTR	Same, NSUBTR = 991.
#60	NRRNGS	Same, NRRNGS = 1.
#61	MTRRAC	Same, MTRRAC = 0.
#62	MTRRT	Same, MTRRT = 1.
#63	MTRFAC	Same, MTRFAC = 2.
#64	MTRFT	Same, MTRFT = 2.
#65	IROPT	Same, IROPT = 1.

B. Block 12, POWINA

Location, Input Variable

#408	SLLMAX	Same, SLLMAX = 1.00.
#409	PITCHG	Same, PITCHG = 0.154827.
#410	PITCHA	Same, PITCHA = 0.154887.
#411	PITCHT	Same, PITCHT = 0.154887.
#412	RDEXCF	The radial expansion coefficient for the oxide fuel core design is given in Table III as - 64.0 $\$/m$, so RDEXCF = - 64.0.
#413	TLPRRC	Using the same criteria as with the metallic fuel design, the higher location of the core in the subassembly for the oxide core requires that the clearance at the top load pads be smaller. The resulting value is - 7.0×10^{-5} m. This value is at the reference temperature of 300 K, and the difference in thermal expansion coefficients between the core assemblies and the restraint ring increases this value so that a positive clearance would be present at operating conditions in the absence of any thermal bending of the subassembly. The clearance at refueling temperatures may not be sufficient, however. For the present study, this is the value required for the reactivity behavior shown in Fig. 3, so that TLPRRC = -7.0×10^{-5} .
#414	BNDMM1	Same, BNDMM1 = 1.00×10^{-3} .
#415	BNDMM2	Same, BNDMM2 = 1.00×10^{-3} .
#419	DFLTCS	Same, DFLTCS = 0.0.

#420	DFLTSS	Same, DFLTSS = 0.0.
#421	ACLPRC	Same, ACLPRC = 0.0.
#422	FCDTR1	Same, FCDTR1 = 0.0.
#423	FCDTR2	Same, FCDTR2 = 0.10.
#424	FCDTRF	Same, FCDTRF = 0.35.
#425	DRCOLL	Same, DRCOLL = 0.0.
#426	CRSAC	Same, CRSAC = 1.624×10^{-3} .
#427	RR1TC	Same, RR1TC = 0.0.
#428	RR2TC	Same, RR2TC = 500.0.

These input variables are repeated within the description of the entire input blocks for Block 1 and Block 12 in Appendices A and B.

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APPENDIX D. PRIMARY AND INTERMEDIATE SYSTEMS MODELLING

This appendix contains the development of the model of the LSPB reactor plant using PRIMAR-4 in SAS4A [D1]. All of the input data required for modelling the remainder of the reactor plant is described in this appendix, including the primary and intermediate loops and both decay heat removal systems. The input data is also listed in Appendices F and G for the oxide and metallic fuel cores, respectively. The modelling is identical for each fuel type, with the exception that the elevations are different to reflect the different location of the core within the assembly, and all elevations are referenced to the bottom of the core axial mesh. This is discussed in Appendices A and B for the two fuel types. The input data is contained in Block 3, INPMR4, for all of the integer data, and in Block 18, PMR4IN, for the floating point data.

The model of the LSPB plant that was developed for this study is shown in Fig. D-1, which is the same as Fig. 5 in the main report. The corresponding tables, Tables IV and V, are reproduced here as Tables D-1 and D-2. The reasoning behind this model is given in the following discussion for the primary circuit, the intermediate loop, and the decay heat removal systems.

I. Primary Circuit

The primary circuit contains all of the sodium which flows through the core in the reactor vessel. The sodium travels from the core into the outlet plenum, then through the heat exchanger for the intermediate heat transport system (IHTS). The sodium flows out of the heat exchanger into the cold pool, where the inlet pipes to the pumps are located. The circuit is completed by the sodium flowing back through the pumps into the inlet plenum and up through the core assemblies.

As described in Section II.A, the detailed models in the SAS4A channels were used only for the driver and internal blanket subassemblies. In PRIMAR-4, all of the SAS4A channels are represented by liquid element 1, connecting compressible volumes CV1 and CV2, which simulate the inlet and outlet plenums. These are listed in Table D-1. Starting with this arrangement, the remainder of the core assemblies are represented using more liquid elements between CV1 and CV2. In this design there are 90 radial blanket assemblies, and they are each represented by three liquid elements (LE), LE2, LE3, and LE4. Since these elements are used to represent a radial blanket assembly, LE2 and LE4 are pipe elements with characteristics of the pin section of the assembly below and above the power producing section of the pin. The power producing region is represented by LE3, which is a PRIMAR-4 "bypass channel". This type of element was designed for simulating the power producing regions of the core which may exist outside of the SAS4A channels, depending on the modelling. Since all of the radial blanket assemblies are considered to be identical, the 90 assemblies are represented with 1 set of these 3 elements duplicated 90 times. This is described in Section V under CVLMLT, (Block 18, Loc. 82-161), the multiplicity factors for the liquid segments. The pipe and "bypass channel" elements in PRIMAR-4 use the same equation for calculating the liquid flow rate, so that the elements LE2, LE3, and LE4 can be combined into one liquid segment, LS2. The SAS4A channels are always LS1. However, the heat transfer calculations are different, so three temperature groups are required, T1, T2, and T3. These are listed in Table D2.

Fig. D-1. Schematic of the PRIMAR-4 Model of the LSPB Reactor Plant.

TABLE D-1. PRIMAR-4 Model of the LSPB Reactor System: Volumes

Volume	Compressible Volume No.
Core inlet plenum	1
Core outlet plenum/hot pool	2
Cold pool	3
Pump outlet plenum for pump group 1	4
Pump outlet plenum for pump group 2	5
Sodium volume between baffle and redan, hot side	6
Sodium volume between baffle and redan, cold side	7
Intermediate loop pump bowl	8
Pipe tee section	9
Intermediate loop sodium expansion tank	10
DRACS-I loop sodium tank	11
DRACS-II loop sodium tank	12

Table D-2. PRIMAR-4 Model of the LSPB Reactor System: Elements

Component	Liquid Flow Element	Liquid Flow Segment	Temperature Group
PRIMARY CIRCUIT			
Core	1	1	-
Radial Blankets:			
Lower section	2	2	1
Blanket section	3	2	2
Upper section	4	2	3
Radial Shields and Control Assemblies:			
Lower section	5	3	4
Blanket section	6	3	5
Upper section	7	3	6
Core Bypass Flow:			
Lower section	8	4	7
Blanket section	9	4	8
Upper section	10	4	9
IHX:			
Inlet section	11	5	10
Tube section	12	5	11
Outlet section	13	5	12
Pump Group 1:			
Pump inlet pipe	14	6	13
Pump	15	6	13
Pump outlet pipe	16	7	14
Pump outlet pipe	17	7	14
Pump outlet pipe	18	8	15
Pump outlet pipe	19	8	15
Pump Group 2:			
Pump inlet pipe	20	9	16
Pump	21	9	16
Pump outlet pipe	22	10	17
Pump outlet pipe	23	10	17
Pump outlet pipe	24	11	18
Pump outlet pipe	25	11	18
Outlet plenum to redan/ baffle volume flow path	26	12	19
Cold pool to redan/baffle volume flow path	27	13	20

Table D-2. PRIMAR-4 Model of the LSPB Reactor System: Elements (Cont'd)

Component	Liquid Flow Element	Liquid Flow Segment	Temperature Group
DRACS-I IHX:			
Inlet section	28	14	21
Shell section	29	14	22
Outlet section	30	14	23
DRACS-II IHX:			
Inlet section	31	15	24
Tube section	32	15	25
Outlet section	33	15	26
INTERMEDIATE HEAT TRANSPORT SYSTEM			
Pump	34	16	27
Horizontal pipe	35	16	27
Vertical downcomer	36	16	27
IHX shell side	37	16	11
Vertical outlet pipe	38	16	28
Horizontal pipe	39	16	28
Steam generator	40	16	29
Vertical pipe	41	16	30
Horizontal pipe	42	17	31
Sodium expansion line	43	18	32
DRACS-I			
Pipe	44	19	33
Air heat exchanger	45	19	34
Pipe	46	19	35
Vertical pipe	47	19	35
IHX tube side	48	19	22
Vertical pipe	49	19	36
Pipe	50	19	36
DRACS-II			
Pipe	51	20	37
Air heat exchanger	52	20	38
Pipe	53	20	39
Vertical pipe	54	20	39
IHX shell side	55	20	25
Vertical pipe	56	20	40
Pipe	57	20	40

The shield and control assemblies are not represented by the core channels either. The flow through these assemblies is simulated by one set of three elements, in the same manner as for the radial blankets. However, in this case, only one set is used which covers the total flow area of these assemblies. The calculation for the characteristics of these three elements, LE5, LE6, and LE7, is covered in detail in Section V. A "bypass channel" element was included as LE6, but does not produce power in the present version of the model. It was mainly included to allow flexibility for modelling some minor effects, such as power produced in the control rods or in the radial shields, if desired. As with the radial shield assemblies, the three elements are in the same liquid segment, LS3, but use three different temperature groups, T4, T5, and T6.

Another set of three elements was used to account for the remaining flow through the core region, typically identified as "core bypass flow", and is associated with leakage at the grid plate/subassembly connection. This sodium flows up to the outlet plenum in the gaps between the subassemblies. Since this region does not really have a defined geometry, small values for all of the input variables connected with LE7, LE8, and LE9 were assumed, just large enough to ensure stability of the heat transfer calculation in PRIMAR-4 consistent with the remainder of the model. These elements are in liquid segment LS4 and in temperature groups T7, T8, and T9.

The outlet plenum is modelled by one volume, CV2, which contains sodium and cover gas. Most of the sodium flow leaves CV2 through the heat exchangers to the intermediate heat transport system. In the LSPB design, there are 4 identical IHTS loops, each with its own steam generator. The current PRIMAR-4 model of LSPB was developed with one IHX and its associated intermediate loop out to the steam generator. This loop is then duplicated for a total of 4 IHTS loops. The elements LE11, LE12, and LE13 model the IHX, where LE12 covers the heat transfer section between the primary sodium and the intermediate loop. LE11 and LE13 are piping elements with characteristics simulating the inlet and outlet sections of the IHX. All three elements are in the same liquid segment, LS5, but three temperature groups are required, T10, T11, and T12. The multiplicity factor is 4 to account for all 4 IHX loops. The sodium leaving LE13 flows into the cold pool, CV3.

The sodium is taken from the cold pool by the primary circuit pumps. In LSPB, there are also 4 of these, each with one inlet pipe section and two outlet pipes feeding into the core inlet plenum, CV1. For the PRIMAR-4 model, the pumps have been modeled as two groups. The first group contains elements LE14 and LE15 going from CV3 to CV4, where CV4 represents the pump outlet plenum. Each pump has two pipes from the pump outlet plenum to the core inlet plenum. These are LE16, LE17, LE18, and LE19 in the model, where LE16 and LE18 are associated with the vertical section of each pipe coming down from the pump, and LE17 and LE19 with the horizontal section going into the core inlet plenum. PRIMAR-4 models pipes as straight sections between the two elevations associated with each element, so that if a pipe has a vertical and a horizontal section, each section must have its own liquid element. The elements LE14 and LE15 are in liquid segment LS6, LE16 and LE17 are in liquid segment LS7, and LE18 and LE19 are in LS8. These elements are also in temperature groups T13, T14 and T15, respectively, as listed in Table D-2.

This grouping of elements and compressible volume, with their associated liquid segments and temperature groups, is repeated for LE20, LE21, CV5, LE22, LE23, LE24, and LE25. In the present configuration, each grouping represents 2 pump circuits. This is reflected in the multiplicity factors for the segments representing these elements. The multiplicity is 2 at the beginning of the segments LS6 and LS9, and 2 at the end of the segments LS7, LS8, LS10, and LS11. This can be easily changed to having one grouping representing 3 pump loops, with the other grouping isolating one of the pump loops. This would be useful if the effect of the seizure of one pump were to be analyzed, or if the effect of the breaking of 1 out of 8 of the pipes from the pumps to the core inlet plenum were to be investigated. The latter is planned for the next stage of the study.

The modelling of the primary coolant pumps is handled by the "homologous" pump model in PRIMAR-4. This model allows pump operation in all four modes, normal pump mode, energy dissipation mode, turbine mode, and reverse pump mode. This pump model can be applied to centrifugal pumps of the type in LSPB. Basically the operating characteristics of the pump can be defined in terms of a set of parameters used in conjunction with nondimensional or normalized variables. The parameters used are typical values for this type of centrifugal pump, as derived by comparison with data on a prototype pump for the Clinch River Breeder Reactor (CRBR). All of this information is covered in Section V.

The other components that interact with the primary circuit are the two DRACS loops. The first, DRACS-I, has a heat exchanger totally within the outlet plenum. The heat exchanger in the outlet plenum is modelled as an IHX at present, with LE28, LE29, and LE30 representing the sodium flow on the shell side of the heat exchanger. As with the IHTS IHX, LE28 and LE30 are pipe elements for the inlet and outlet sections of the heat exchanger, and LE29 is the heat transfer section. The DRACS loop is treated as an additional intermediate loop. Having the sodium enter from and exit to CV2 required a special treatment in the input data, as indicated in Section IV and V, as well as a code modification, listed in Appendix H. Both loops are natural circulation loops, so no pump modelling is required. The DRACS characteristics at full power are used throughout each transient. The modelling for DRACS-II is similar, using LE31, LE32, and LE33, except that it connects CV2 and CV3. The same special treatment used for DRACS-I was also used for DRACS-II for consistency, although it wasn't necessary in this case. The only additional feature for DRACS-II is the use of a valve element at the exit of the IHX. This allows the DRACS-II loop to be shut off during normal operation of the reactor, and activated when the sodium level in the reactor is calculated to rise above the wall of the well surrounding the DRACS-II IHX.

Another important feature of the LSPB design is the horizontal boundary between the hot pool and the cold pool, called the redan. On both sides of this boundary, baffle plates are placed to prevent mixing between the bulk of the sodium in the pools and the sodium between the baffles and the redan. These baffles do allow flow from the baffle region out into the pools, but this is greatly restricted by the close fit of the baffles to the vessel components. In the model, the volumes are represented by CV6 and CV7, with each connected to the larger pool by a short pipe section. This allows sodium to enter or leave the baffle region in response to temperature changes in the

sodium. These volumes are particularly important in long-term transients as they provide a considerable heat sink to moderate the temperature rise of the hot and cold pools. These heat transfer paths are also included, described in Appendix D under the term "component-to-component heat transfer". While there are many heat transfer paths, only the paths from CV2 to CV6, from CV6 to CV7, and from CV7 to CV3 are included at present.

II. Intermediate Loop

The intermediate heat transport system is identified as the intermediate loop in the model. It contains the intermediate loop pump, the steam generator, and all necessary piping. Starting with the centrifugal pump in the cold return section of the loop, the pump, LE34, is also modelled with the "homologous" pump model in PRIMAR-4. The detailed characteristics are listed in Section V. The sodium flows through a long horizontal pipe from the steam generator building into the containment building, LE35. The sodium then flows vertically down into the IHX, LE36. The intermediate loop sodium is on the shell side of the heat exchanger, LE37, and flows vertically upward out of the IHX up to an elevation slightly lower than the cold return section. Another long horizontal section takes the sodium back to the steam generator, LE40.

The steam generator uses the simple model available in PRIMAR-4, where the normalized temperature drop across the steam generator is specified as a function of time. This is easily changed to the other option, which specifies the steam generator outlet temperature as a function of time, if desired for a particular accident. The sodium flows out of the steam generator and returns to the pump. In this section of the intermediate loop, there is a pipe connection to the sodium expansion tank, CV10. The pipe tee fitting is represented by CV9. The pump also has a pump bowl with a cover gas connected to the cover gas in the sodium expansion tank. This is modelled by CV8 and the gas segment between CV8 and CV10.

The intermediate loop from the pump, through the IHX and the steam generator and up to the pipe tee CV9 is represented by one liquid segment, LS16. The connection between the pipe tee and the pump bowl is LS17, with LS18 connecting to the sodium overflow tank. Since the IHX heat transfer section is already included in temperature group T11, the segment LS16 is divided into 4 new groups as listed in Table D-2.

III. DRACS Loops

The LSPB design incorporates two independent and diverse decay heat removal systems, DRACS-I and DRACS-II. The heat exchangers in the reactor vessel for these systems have been included in the modelling of the primary circuit. For the modelling in PRIMAR-4, both DRACS loops are modelled with the same set of liquid elements and volumes, but with slightly different characteristics where required. Each decay heat loop contains 7 elements, all in 1 liquid segment. Each loop also has 4 new temperature groups, with the heat transfer section of the heat exchanger already included in the temperature groups established for the primary circuit.

Each DRACS loop contains a sodium-to-air heat exchanger. The nominal capacity for each loop was used to initialize the flow rate and temperature distribution within each loop. As such, each loop begins the transient

calculations operating at its full rated capacity for nominal power reactor conditions. Since this is only 28 MW, in comparison to the total reactor power of 3500 MW, this has a negligible effect on the course of the transients until the steam generator is lost and power is reduced to decay heat level.

IV. Block 3, INPMR4

The input data for INPMR4, input block 3, are summarized in this section. It supports the discussion in Section II.B on the modelling of the reactor system using PRIMAR-4. First a summary of each input location is given, with the appropriate values for each, and then a listing of the input block is given.

The listing of the input data block is included in Appendix F for the oxide fuel core and Appendix G for this metallic fuel core. The input data in Block 3, INPMR4, is identical for both metallic and oxide versions.

The elements shown in Fig. D-1 represent the various components as listed in Tables D-1 and D-2. For those elements which represent duplicate components, this is also indicated in the tables. The arrangement in Fig. D-1 given by the input data in Block 3 as follows.

Location, Input Variable

- Loc. 1 The number of compressible volumes in the primary loop is 7, so NCVP = 7.

- Loc. 2 The number of compressible volumes in the secondary, or intermediate, loops is 5, so NCVS = 5. The secondary loops include the intermediate heat transport loop and both DRACS loops, where there are 3 volumes in the IHTS and 1 volume each in the DRACS loops. The fact that the intermediate loop in the model represents 4 identical loops in the plant does not affect this input value.

- Loc. 3 Not yet operational.

- Loc. 4 The number of liquid segments in the primary loop is 15, as indicated in Table D-2. Thus NSEGLP = 15.

- Loc. 5 The number of liquid segments in the secondary loop is 5, as indicated in Table D-2. Each DRACS loop comprises only 1 liquid segment which starts and ends at the same compressible volume.

- Loc. 6 Not yet operational

- Loc. 7 No gas segments are used in the primary loop.

- Loc. 8 The number of gas segments in the secondary loop is 1, between CV8 and CV10, as shown in Fig. D-1. This is the only gas segment in the model. Then NSEGGs = 1.

- Loc. 9 Not yet operational

Loc. 10 The total number of liquid flow elements is 57, as listed in Table D-2. and shown on Fig. D-1, so NELEMT = 57.

Loc. 11 These variable indicate the compressible volume type
-48 for each compressible volume in the model. The selection for each volume should be clear, as follows:

Compressible Volume	Loc.	Type	Description
Primary loop			
1	11	1	Inlet plenum
2	12	7	Outlet plenum with cover gas
3	13	8	Pool with cover gas
4	14	4	Liquid volume, no cover gas
5	15	4	Liquid volume, no cover gas
6	16	4	Liquid volume, no cover gas
7	17	4	Liquid volume, no cover gas
IHTS			
8	18	9	Pump bowl with cover gas
9	19	4	Liquid volume, no cover gas
10	20	8	Pool with cover gas
DRACS-I			
11	21	8	Pool with cover gas
DRACS-II			
12	22	8	Pool with cover gas

Loc. 49 The liquid flow element types are given as follows, based
-188 on the modelling given in Table D-2:

Flow Element	Loc.	Type	Description
Primary Loop			
1	49	1	Core subassemblies
2	50	3	Pipe
3	51	2	Bypass channel
4	52	3	Pipe
5	53	3	Pipe
6	54	2	Bypass channel
7	55	3	Pipe
8	56	3	Pipe
9	57	2	Bypass channel
10	58	3	Pipe
11	59	3	Pipe
12	60	7	IHX, tube side
13	61	3	Pipe
14	62	3	Pipe
15	63	5	Pump impeller
16	64	3	Pipe
17	65	3	Pipe
18	66	3	Pipe

19	67	3	Pipe
20	68	3	Pipe
21	69	5	Pump impeller
22	70	3	Pipe
23	71	3	Pipe
24	72	3	Pipe
25	73	3	Pipe
26	74	3	Pipe
27	75	3	Pipe
28	76	3	Pipe
29	77	6	IHX, shell side
30	78	3	Pipe
31	79	3	Pipe
32	80	7	IHX, tube side
33	81	11	Valve
Intermediate loop			
34	82	5	Pump impeller
35	83	3	Pipe
36	84	3	Pipe
37	85	6	IHX, shell side
38	86	3	Pipe
39	87	3	Pipe
40	88	8	Steam generator, sodium side
41	89	3	Pipe
42	90	3	Pipe
43	91	3	Pipe
DRACS-I			
44	92	3	Pipe
45	93	12	Air heat exchanger, sodium side
46	94	3	Pipe
47	95	3	Pipe
48	96	7	IHX, tube side
49	97	3	Pipe
50	98	3	Pipe
DRACS-II			
51	99	3	Pipe
52	100	12	Air heat exchanger, sodium side
53	101	3	Pipe
54	102	3	Pipe
55	103	6	IHX, shell side
56	104	3	Pipe
57	105	3	Pipe

Loc. 189 The compressible volumes at each end of the liquid segment
 -258 can be found by referring to Fig. D-1 and Table D-2. The
 results are as follows:

Segment No.	Loc.	Starts with CV No.	Loc.	Ends with CV No.
Primary loop				
1	189	1	190	2
2	191	1	192	2
3	193	1	194	2
4	195	1	196	2
5	197	2	198	3
6	199	3	200	4
7	201	4	202	1
8	203	4	204	1
9	205	3	206	5
10	207	5	208	1
11	209	5	210	1
12	211	2	212	6
13	213	3	214	7
14	215	2	216	2
15	217	2	218	3
Intermediate loop				
16	219	8	220	9
17	221	9	222	8
18	223	9	224	10
DRACS-I				
19	225	11	226	11
DRACS-II				
20	227	12	228	12

Loc. 269 The compressible volumes at the ends of the gas segments
 -324 are found in the same manner as for liquid segments,
 Loc. 189-268. Since there is only one gas segment, in the
 intermediate loop, the input data are as follows:

Segment No.	Loc.	Starts with CV No.	Loc.	Ends with CV No.
1	269	8	270	10

Loc. 325 The number of elements in each liquid segment can be
 -364 found by examining Fig. D-1 and Table D-2. The combining
 of elements into one segment implies that the fluid flow
 calculations for each element in the segment is the same,
 with the result as follows:

Segment No.	Loc.	Number of Elements
-------------	------	--------------------

Primary loop

1	325	1
2	326	3
3	327	3
4	328	3
5	329	3
6	330	2
7	331	2
8	332	2
9	333	2
10	334	2
11	335	2
12	336	1
13	337	1
14	338	3
15	339	3

Intermediate loop

16	340	8
17	341	1
18	342	1

DRACS-I

19	343	7
----	-----	---

DRACS-II

20	344	7
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Loc. 365 The number of the first element in each liquid segment
-404 is also given in Table D-2 and Fig. D-1.

Segment No.	Loc.	Number of Elements
-------------	------	--------------------

Primary loop

1	365	1
2	366	2
3	367	5
4	368	8
5	369	11
6	370	14
7	371	16
8	372	18
9	373	20
10	374	22
11	375	24
12	376	26
13	377	27
14	378	28
15	379	31

Intermediate loop

16	380	34
17	381	42
18	382	43

DRACS-I

19	383	44
----	-----	----

DRACS-II

20	384	51
----	-----	----

Loc. 405 The number of sodium pumps in the model is 2 in the primary loop, (with each representing 2 identical pumps), and 1 in the intermediate loop, (with 4 identical intermediate loops in the model), for a total of 3, so NPUMP = 3.

Loc. 406 The element number corresponding to each pump in the model
-417 is:

Pump No.	Loc.	Element No.
----------	------	-------------

Primary loop

1	406	15
2	407	21

Intermediate loop

3	408	34
---	-----	----

Loc. 418 The type of pump being represented by the model is a
-429 centrifugal pump, which uses the homologous pump model in PRIMAR-4:

Pump No.	Loc.	Pump Model
----------	------	------------

Primary loop

1	418	2
2	419	2

Intermediate loop

3	420	2
---	-----	---

Loc. 430 The homologous pump model contains a provision for locking
-469 the pump rotor at a prescribed rotor speed, so that none of the options in ILRPMP need to be used; they are all 0.

Loc. 470 The number of IHXs is 1 in the primary loop for the exchange of heat with the intermediate loop (representing 4 identical intermediate loops), and 1 for each of the decay heat removal systems, for a total of 3, so that NIHX = 3.

Loc. 471 Not yet operational

Loc. 472 There are not any check valves in the model.

Loc. 473 The element number of the IHXs is given in Table D-2. as
-476 follows:

IHX No.	Loc.	Element No.
Primary loop		
1	473	12
2	474	29
3	475	32

Loc. 477 Not yet operational
-480

Loc. 481 The element numbers of the IHXs in the intermediate loops
-484 is given by the intermediate heat transport system, and
also by considering the DRACS loops to be intermediate
loops:

IHX No.	Loc.	Element No.
Intermediate loop		
1	481	37
DRACS-I		
2	482	48
DRACS-II		
3	483	55

Loc. 485 Not yet operational
488

Loc. 489 All of the calculations for the heat exchangers is done
-492 with the detailed model, so this input is 0.

Loc. 493 Not yet operational
-496

Loc. 497 This input activates different corrections to the mass
flow difference between the SAS channels and PRIMAR-4 as
a result of the use of extrapolated values for mass flow
rate between these two models. The current recommended
value is 1 for this input, which applies a correction to
the mass of the outlet plenum, so IPRADJ = 1.

Loc. 498 Set to 0 since debugs are not required.
-508

Loc. 509 Not yet operational

Loc. 510 There are not any pipe breaks for the present analyses.
-511

- Loc. 512 The number of temperature groups is obtained by combining elements in the same segment which have the same heat transfer calculation. As listed in Table D-2, there are 40 temperature groups, so $NTGPT = 40$. Note that both sides of each IHX are represented by the same temperature group, not by individual ones.
- Loc. 513 The number of nodes in each temperature group is a user choice,
 -612 subject to the restriction given in the input description. As would be expected, more nodes have been placed in elements with potentially large temperature gradients in the flow direction, as follows:

Temperature Group No.	Loc.	Number of Nodes
Primary loop		
1	513	4
2	514	7
3	515	4
4	516	4
5	517	7
6	518	4
7	519	4
8	520	7
9	521	4
10	522	4
11	523	41
12	524	4
13	525	4
14	526	4
15	527	4
16	528	4
17	529	4
18	530	4
19	531	4
20	532	4
21	533	4
22	534	41
23	535	4
24	536	4
25	537	41
26	538	4
27	539	20
28	540	20
29	541	41
30	542	10
31	543	4
32	544	4
33	545	4
34	546	2
35	547	10
36	548	10
37	549	4
38	550	2
39	551	10
40	552	10

Loc. 613 The first element in each temperature group is found
 -712 from the list of elements in Table D-2, as follows:

Temperature group No.	Loc.	First element
Primary loop		
1	613	2
2	614	3
3	615	4
4	616	5
5	617	6
6	618	7
7	619	8
8	620	9
9	621	10
10	622	11
11	623	12
12	624	13
13	625	14
14	626	16
15	627	18
16	628	20
17	629	22
18	630	24
19	631	26
20	632	27
21	633	28
22	634	29
23	635	30
24	636	31
25	637	32
26	638	33
27	639	34
28	640	38
29	641	40
30	642	41
31	643	42
32	644	43
33	645	44
34	646	45
35	647	46
36	648	49
37	649	51
38	650	52
39	651	53
40	652	56

Loc. 713 The last element in the temperature group is also found
 -812 from Table D-2:

Temperature group No.	Loc.	Last element
--------------------------	------	--------------

Primary loop

1	713	2
2	714	3
3	715	4
4	716	5
5	717	6
6	718	7
7	719	8
8	720	9
9	721	10
10	722	11
11	723	12
12	724	13
13	725	15
14	726	17
15	727	19
16	728	21
17	729	23
18	730	25
19	731	26
20	732	27
21	733	28
22	734	29
23	735	30
24	736	31
25	737	32
26	738	33
27	739	36
28	740	39
29	741	40
30	742	41
31	743	42
32	744	43
33	745	44
34	746	45
35	747	47
36	748	50
37	749	51
38	750	52
39	751	54
40	752	57

Loc. 813 The number of bypass channel elements is 3, as listed in Table D-2, being element numbers 3, 6, and 9. Therefore NBYP = 3.

Loc. 814 The number of nodes in the lower part of walls A and B in
-821 the bypass channel is set to 7, which gives the maximum number of nodes in each bypass channel:

Bypass Channel	Element No.	Loc.	Number of Nodes
1	3	814	7
2	6	815	7
3	9	816	7

- Loc. 822 The decay curve to be used for each bypass channel has
-829 set to 1; this is not important since only the first
 bypass channel is generating any power in this simulation.
- Loc. 830 The element numbers for the bypass channels are as listed
-837 in the description for Loc. 814-821.
- Loc. 838 Not used by the code.
- Loc. 839 The number of steam generators is 1, element 40 in the
 intermediate loop, so NSGN = 1.
- Loc. 840 The element number for the steam generator is 40, as
-851 shown in Fig. D-1 and Table D-2.
- Loc. 852 The table look-up option is used for all steam generator
-863 modelling; the choice of either normalized temperature
 drop or steam generator outlet temperature is made based
 on the accident scenario and the characteristics of the
 steam generator. At present, it is set to 1.
- Loc. 864 The choice of steam generator type needs to be made
-875 regardless of the modelling used for the steam generator.
 In this case, it is appropriate to use a once-through
 steam generator, type 3.
- Loc. 876 Locations not needed with the present model.
-972
- Loc. 973 There is one valve in the model, element 33, to control
 the flow through DRACS-II in response to sodium level
 changes in the reactor; NVALVE = 1.
- Loc. 974 The element number for the valve is 33.
-981
- Loc. 982 The DTMPB table for the valve loss coefficient is set
-989 to table 2, since the steam generator table is table 1.
- Loc. 990 Not used.
- Loc. 991 There are 2 air dump heat exchangers in the model, one for
 each DRACS loop, so NDHX = 2.
- Loc. 992 The element numbers for the air dump heat exchangers are
-995 45 and 52, as listed in Table D-2.
- Loc. 996 Both air dump heat exchangers are natural circulation, so
-999 this integer is set to 0 in both cases.

Loc. 1000 Not used for this model.
-1007

Loc. 1008 The model contains a contribution to the control rod driveline expansion reactivity feedback from the reactor vessel; since the control rods are suspended from the top of the reactor vessel, an expansion of the reactor vessel would move the control rods out of the core. To include this possibility, NEXPFB = 4, which would allow contributions from the walls of CV2, CV6, CV7, and CV3.

Loc. 1009 The compressible volumes contributing to the expansion of
-1018 the reactor vessel, moving the core with respect to the top of the reactor vessel; compressible volumes are designated by negative integers. Based on Fig. D-1, the walls of CV2, CV6, CV7, and CV3 would be involved:

Loc.	Value
1009	-2
1010	-6
1011	-7
1012	-3

Loc. 1019 Not used by the model.

Loc. 1020 No bypass channels are involved in component-to-component
-1051 heat transfer in the present model.

Loc. 1052 Component-to-component heat transfer was not included in
-1081 the first version of the model, but has been added as the model is being refined. At present, the heat transfer between CV2 and CV6, between CV7 and CV3, and between CV6 and CV7 has been added, as this provides a significant heat sink to limit the temperature rise of the outlet plenum. Including these heat transfer paths also means that the code must execute a steady-state "null transient" in order to initialize the temperatures. In these locations, the wall of a particular element or volume is listed. The corresponding coolant in another element or volume is listed in Loc. 1082-1111. Given the heat transfer paths described above, the input data in these locations is as follows:

Heat Transfer Path No.	Wall Temperature	Loc.	Value
1	CV2	1052	-2
2	CV3	1053	-3
3	CV6	1054	-6

Loc. 1082 These locations contain the corresponding elements for the
-1111 coolant transferring heat with the wall in Loc. 1052-1081.

Heat Transfer Path No.	Coolant Temperature	Loc.	Value
1	CV6	1082	-6
2	CV7	1083	-7
3	CV7	1084	-7

Loc. 1112 Not used for the present model.
-1152

Loc. 1153 There are two compressible volumes which share the same cover gas, the outlet plenum and the cold pool in the reactor vessel, or CV2 and CV3. Thus, NCCV = 2.

Loc. 1154 The number of the first compressible volume which shares the cover gas space is CV2, so ICCVFS = 2.

Loc. 1155 The option to specify the steady-state temperature drop
-1158 across an IHX is used for DRACS-I. This was required since the code will not initialize correctly for an IHX which has the inlet and outlet in the same compressible volume. For consistency, this option was also applied to DRACS-II. The results are:

IHX No.	Loc.	Value
1	1155	0
2	1156	1
3	1157	1

Loc. 1159 Not used in the current model.
-1170

Loc. 1171 Even though there is a flow segment which provides a flow path in parallel to the main IHX flow path, the present model treats all heat exchangers in the reactor vessel as IHXs, and there is not a bypass of any IHX: NIHXBY = 0.

Loc. 1172 Not used in the current model.
-1287

Loc. 1288 The steady-state "null transient" is required to initialize the system temperatures when component-to-component heat transfer is included. For the present model, a null transient of 4000 sec is sufficient to stabilize all of the temperatures for the accident calculations, and a step size of 20 sec is a typical value, so 200 time steps are needed; ISSCPC = 200.

Loc. 1289 For analysis purposes, a printout every 20 steps during the null transient gives adequate information, so ISST15 = 20.

V. Block 18, PMR4IN

Location, Input Variable

#2-21 FLOSSL (ISGL) Initial flow rate in liquid flow segments, Kg/sec. The input for these locations is the initial flow through the 20 (ISGL = 20) liquid segments. The flow through the core, radial blanket, radial shield, control rods, and bypass for the oxide fuel core and the metallic fuel core are obtained from Ref. D2. The pump flow and the primary and secondary flows through the DRACS are the same for both the oxide and metallic cores. The flow through each primary pump is obtained from Ref. D2. The flow rate in the intermediate loops and the DRACS loops are from Ref. D3. The flow rates are listed below.

	Flow Rate (kg/sec)	
	Oxide Core	Metallic Core
Core	16449.7	16391.6
Radial Blanket	932.8	980.7
Radial Shield and Control Rod	335.0	345.4
Bypass	251.6	251.6
Primary Pump	4492.3	4492.3
Secondary Pump	4460.4	4460.4
DRACS-I : Primary	63.0	63.0
Secondary	65.8	65.8
DRACS-II: Primary	65.8	65.8
Secondary	63.0	63.0

FLOSSL(1) Flow through the core.
 = 16449.7 Kg/sec for the oxide core.
 = 16391.6 Kg/sec for the metallic core.

There are 90 radial blanket assemblies in either the oxide or metallic core. In this modelling only the flow through one assembly is used.

FLOSSL(2) Total radial blanket flow/90 for the flow through a single blanket assembly.
 = 932.8/90
 = 10.3644 Kg/sec for the oxide core.
 = 980.7/90
 = 10.8961 Kg/sec for the metallic core.

FLOSSL(3) Flow through radial shield and control rods.
 = 335.0 Kg/sec for the oxide core.
 = 345.4 Kg/sec for the metallic core.

- FLOSSL(4) Flow through the bypass channel.
 = 251.56 Kg/sec for both oxide and metallic fuel designs.
 In the following locations the input for the oxide and metallic core are the same. The total primary flow is the same for both cores at 17969.1 Kg/sec.
- FLOSSL(5) Flow from the reactor outlet plenum through the IHX.
 There are 4 IHX's in the primary loop. The flow through DRACS-II is in parallel with the IHXs, and must be subtracted from the total flow through the IHXs. The resulting flow through each IHX is :
 $(17969.1 - 65.8) / 4$ Kg/sec.
 = 4475.85 Kg/sec
- FLOSSL(6) Flow through each primary pump inlet pipe.
 = 4492.3 Kg/sec
- FLOSSL(7) Flow through the first discharge pipe of a primary pump.
 Since there are two discharge pipes in each primary pump, the flow through each pipe is 1/2 of the flow in each pump.
 $= 4492.3 / 2$ Kg/sec
 = 2246.2 Kg/sec
- FLOSSL(8) Flow through the second discharge pipe of a primary pump.
 The flow through the second discharge pipe is assumed to be the same as through the first.
 = 2246.2 Kg/sec
- FLOSSL(9) Flow through each primary pump. The flow rate is assumed equal in each of the 4 primary pumps.
 = 4492.3 Kg/sec
- FLOSSL(10) Flow through the first discharge pipe of the second primary pump group. The flow rate is assumed to be equal in each discharge pipe.
 $= 4492.3 / 2$ Kg/sec
 = 2246.2 Kg/sec
- FLOSSL(11) Flow through the second discharge pipe of the second primary pump group. Assumed to be the same as the flow through the first discharge pipe.
 = 2246.2 Kg/sec
- FLOSSL(12) Flow between CV2 and CV6. There is nominally no flow between CV2 and CV6 at steady-state.
 = 0.0 Kg/sec
- FLOSSL(13) Flow between CV3 and CV7. There is nominally no flow between CV3 and CV7 at steady-state.
 = 0.0 Kg/sec
- FLOSSL(14) The primary-side flow in DRACS-I.
 = 63.0 Kg/sec

FLOSSL(15) The primary-side flow in DRACS-II.
= 65.8 Kg/sec

FLOSSL(16) Flow rate through the intermediate loop, including the pump and the steam generator.
= 4460.4 Kg/sec

FLOSSL(17) Flow through the pipe connecting the pipe tee to the pump bowl. This is the same as the flow through the remainder of the intermediate loop.
= 4460.4 Kg/sec

FLOSSL(18) Flow entering the intermediate sodium expansion tank. There is nominally no flow into this tank at steady-state.
= 0.0 Kg/sec

FLOSSL(19) The secondary-side flow in DRACS-I.
= 65.8 Kg/sec

FLOSSL(20) The secondary-side flow in DRACS-II.
= 63.0 Kg/sec

#42-81 ZINL (ISGL) Elevation of the inlet to the liquid segments. The input in these locations gives the elevation of the inlet to the 20 liquid segments. To be consistent with SAS channel models, the zero level of the oxide core is the end of the bottom axial blanket. The zero level of the metallic core is 4 inches below the bottom of the fuel region. The assembly elevations of the oxide and metallic cores are shown in Figs. A-2 and B-2, respectively. The elevation of the remainder of the reactor system is estimated from Figs. D-2, D-3, and D-4 from Ref. D3.

ZINL(1) Elevation of the inlet of the core element.

For the oxide core:

$$\begin{aligned} &= -(40.0 + 1.0 + 2.5 + 6.0 + 4.0 + 13.5) \\ &= -67.0 \text{ in} \\ &= -1.7018 \text{ m} \end{aligned}$$

For the metallic core:

$$\begin{aligned} &= -(-4.0 + 1.0 + 2.5 + 24.0 + 18.0 + 4.0 + 13.5) \\ &= -59.0 \text{ in} \\ &= -1.4986 \text{ m} \end{aligned}$$

The difference between the zero elevation of the oxide and metallic core is $1.7018\text{m} - 1.4986\text{m} = 0.2032\text{m}$.

ZINL(2) Elevation of the inlet of the radial blanket assembly.
= -1.7018 m for the oxide core
= -1.4986 m for the metallic core

ZINL(3) Elevation of the inlet of the radial shield and control rods.

Fig. D-2. Elevations of the Reactor Internals in LSPB.

Fig. D-3. Cross-Section Elevations of the Primary Heat Transfer System in LSPB.

Fig. D-4. Intermediate Loop Heat Transfer System Profile in LSPB.

- = -1.7018 m for the oxide core
 = -1.4986 m for the metallic core
- ZINL(4) Elevation of the inlet of the segment representing the core bypass flow.
 = -1.7018 m for the oxide core
 = -1.4986 m for the metallic core
- ZINL(5) Elevation of the inlet of the IHX
 According to Fig. D-2, from Ref. D3, the zero reference level in PRIMAR-4 corresponds to the following height in Fig. D-2; $(64'4'' - 185'' + 67'') = 654''$ for the oxide core.

 = $81'3'' - 654'' = 321''$
 = 8.1534 m for the oxide core

 = $8.1534 + 0.2032$ m
 = 8.3566 m for the metallic core
- ZINL(6) Elevation of the inlet of the primary pump in the first primary pump group. From Fig. D-3 by measurement:
 = $44'6'' - 654''$
 = -120"
 = -3.0480 m for the oxide core

 = $-3.0480 + 0.2032$ m
 = -2.8448 m for the metallic core
- ZINL(7) Elevation of the inlet of first discharge pipe of the first primary pump group. From Fig. D-3 by measurement:
 = $54'10'' - 654'' = 4''$
 = 0.1016 m for the oxide core

 = $0.1016 + 0.2032$ m
 = 0.3048 m for the metallic core
- ZINL(8) Elevation of inlet of the second discharge pipe of the first primary pump group. This elevation is the same as for the first discharge pipe.
 = 0.1016 m for the oxide core
 = 0.3048 m for the metallic core
- ZINL(9) Elevation of the inlet of primary pump in the second primary pump group. This elevation is the same as that of the first primary pump group.
 = -3.0480 m for the oxide core
 = -2.8448 m for the metallic core
- ZINL(10) Elevation of the inlet of the first discharge pipe in the second primary pump group. This elevation is the same as that of the discharge pipe of the first primary pump group.
 = 0.1016 m for the oxide core
 = 0.3048 m for the metallic core

- ZINL(11) Elevation of the inlet of the second discharge pipe in the second primary pump group. This elevation is the same as that of the first discharge pipe of the second primary pump group.
 $= 0.1016 \text{ m}$ for the oxide core
 $= 0.3048 \text{ m}$ for the metallic core
- ZINL(12) Elevation of the inlet of the pipe connecting the outlet plenum to the upper volume between the baffle and the redan. Estimated from Fig. D-2, given the nominal element length in Loc. 187:
 $= 185'' - 67'' = 118''$
 $= 2.9972 \text{ m}$ for the oxide core
 $= 2.9972 + 0.2032 \text{ m}$
 $= 3.2004 \text{ m}$ for the metallic core
- ZINL(13) Elevation of the inlet of the pipe connecting the cold pool to the lower volume between the baffle and the redan. Estimated from Fig. D-2, with the lower baffle 5' below the upper baffle.
 $= 118'' - 5' = 56''$
 $= 1.4732 \text{ m}$ for the oxide core
 $= 1.4732 + 0.2032 \text{ m}$
 $= 1.6764 \text{ m}$ for the metallic core
- ZINL(14) Elevation of the inlet of the DRACS-I heat exchanger. Assume the inlet of DRACS-I for the oxide core is at an elevation of 78' 9", from Fig. D-5, then:
 $= 78' 9'' - 654'' = 291''$
 $= 7.3914 \text{ m}$ for the oxide core
 $= 7.3914 + 0.2032 \text{ m}$
 $= 7.5946 \text{ m}$ for the metallic core
- ZINL(15) Elevation of the inlet of the DRACS-II heat exchanger. Assume the inlet of DRACS-II is at the top of the hot pool at 85', from Fig. D-5.
 $= 85' - 654'' = 365''$
 $= 9.2964 \text{ m}$ for the oxide core
 $= 9.2964 + 0.2032 \text{ m}$
 $= 9.4996 \text{ m}$ for the metallic core
- ZINL(16) Elevation of the cold return pipe in the intermediate loop. Fig. D-4 indicates this elevation at 117' at the top of the IHX, and it is assumed to be the same back to the pump outlet.
 $= 117' - 654'' = 750''$
 $= 19.0500 \text{ m}$ for the oxide core
 $= 19.0500 + 0.2032 \text{ m}$
 $= 19.2532 \text{ m}$ for the metallic core

Fig. D-5. Schematic of the Decay Heat Removal Systems in LSPB.

ZINL(17) Elevation of the pipe between the pipe tee and the pump bowl. Assume this elevation is the same as the cold return pipe.
 = 19.0500 m for the oxide core
 = 19.2532 m for the metallic core

ZINL(18) Elevation of the inlet of the pipe connecting the pipe tee with the sodium expansion tank, CV10. Assume this elevation to be the same as that of the cold return pipe.
 = 19.0500 m for the oxide core
 = 19.2532 m for the metallic core

ZINL(19) Elevation of the inlet of the air dump heat exchanger for DRACS-I. This value is taken from Fig. D-5.
 = $88.75' + 74.25' - 654" = 1302"$
 = 33.0708 m for the oxide core
 = $33.0708 + 0.2032$ m
 = 33.2740 m for the metallic core

ZINL(20) Elevation of the inlet of the air dump heat exchanger for DRACS-II. This value is assumed to be the same as for DRACS-I.
 = $99.25' - 69.25' - 654" = 1302"$
 = 33.0708 m for the oxide core
 = 33.0708 m + 0.2032 m
 = 33.2740 m for the metallic core

#82-161 CVLMLT (M,ISGL) Multiplicity factors at liquid segment ends. M=1 at inlet, M=2 at outlet. The multiplicity factors are the same for for both the oxide and metallic cores because both cores have the same PRIMAR-4 model of the reactor plant.

There is one core channel.
 CVLMLT(1,1) = 1, CVLMLT(2,1) = 1

There are 90 radial blanket subassemblies in the core. These are represented by one segment, with the flow rate duplicated for the total of 90 identical subassemblies.
 CLVMLT(1,2) = 90, CVLMLT(2,2) = 90

There is one segment representing the radial shields and control rods.
 CVLMLT(1,3) = 1, CVLMLT(2,3) = 1

There is one segment representing the core bypass flow.
 CVLMLT(1,4) = 1, CVLMLT(2,4) = 1

There are 4 IHXs in the reactor. These are represented by one segment, with the flow rate duplicated for the total through all IHXs. All IHXs are assumed to be identical.
 CVLMLT(1,5) = 4, CVLMLT(2,5) = 4

There are 4 pump loops in the reactor. For flexibility, these 4 loops are represented by 2 pump loops in the model, with each loop representing 2 pump loops in the reactor. Within each pump loop, there is one connection to the pump volume, CV4.

$CVLMLT(1,6) = 2$, $CVLMLT(2,6) = 1$

For each pump loop in the model, there is 1 connection for each outlet pipe to CV4. The exit of the pump outlet pipe represents 2 pipes in the reactor.

$CVLMLT(1,7) = 1$, $CVLMLT(2,7) = 2$

For each pump loop in the model, there is 1 connection for each outlet pipe to CV4. The exit of the pump outlet pipe represents 2 pipes in the reactor.

$CVLMLT(1,8) = 1$, $CVLMLT(2,8) = 2$

Same as for segment 6:

$CVLMLT(1,9) = 2$, $CVLMLT(2,9) = 1$

Same as for segment 7:

$CVLMLT(1,10) = 1$, $CVLMLT(2,10) = 2$

Same as for segment 8:

$CVLMLT(1,11) = 1$, $CVLMLT(2,11) = 2$

There is one pipe element representing the flow between the outlet plenum and the region between the upper baffle plate and the redan.

$CVLMLT(1,12) = 1$, $CVLMLT(2,12) = 1$

There is one pipe element representing the flow between the cold pool and the region between the lower baffle plate and the redan.

$CVLMLT(1,13) = 1$, $CVLMLT(2,13) = 1$

There is one DRACS-I in the reactor.

$CVLMLT(1,14) = 1$, $CVLMLT(2,14) = 1$

There is one DRACS-II in the reactor.

$CVLMLT(1,15) = 1$, $CVLMLT(2,15) = 1$

Within each intermediate loop, there is one pump and one steam generator, with single hot and cold pipes connected to the IHX.

$CVLMLT(1,16) = 1$, $CVLMLT(2,16) = 1$

Within each intermediate loop, there is one pipe between the pipe tee and the pump volume.

$CVLMLT(1,17) = 1$, $CVLMLT(2,17) = 1$

There is one line to a sodium expansion tank in each intermediate loop.

$CVLMLT(1,18) = 1$, $CVLMLT(2,18) = 1$

There is one DRACS-I in the reactor, with the entire loop represented by one segment.

CVLMLT(1,19) = 1, CVLMLT(2,19) = 1

There is one DRACS-II in the reactor, with the entire loop represented by one segment.

CVLMLT(1,20) = 1, CVLMLT(2,20) = 1

#162-301 ZOUTEL
(IELL)

The height at the outlet of each liquid element.
The reference elevation is the same as described for the segment inlet, Loc. 42-81.

ZOUTEL(1) Elevation of the top of the core subassemblies.
= 185" - 67" = 118"
= 2.9972 m for the oxide core

= 185" - 59" = 126"
= 3.2004 m for the metallic core

ZOUTEL(2) Elevation of the top of the pipe representing the part of the radial blanket assembly below the active fuel region. For the oxide core this is the interface between the lower fission gas plenum and the lower axial blanket.
= 0.0000 m

For the metallic core this is set to the bottom of the axial mesh in the core channels.
= 0.0000 m

ZOUTEL(3) Elevation of the top of the active fuel region in the radial blanket assemblies. For the oxide core;
= 68"
= 1.7272 m

For the metallic core this is the top of the axial mesh in the core channels.
= 92"
= 2.3368 m

ZOUTEL(4) Elevation of the top of the radial blanket assemblies. The assemblies are the same length for both the oxide and metallic cores, but the reference elevation is different.
= 185" - 67" = 118"
= 2.9972 m for the oxide core

= 185" - 59" = 126"
= 3.2004 m for the metallic core

ZOUTEL(5) Same as for element 2.
For both oxide and metallic cores, this elevation is 0.0.

- ZOUTEL(6) Same as for element 3.
 = 68"
 = 1.7272 m for the oxide core

 = 92"
 = 2.3368 m for the metallic core
- ZOUTEL(7) Same as for element 4.
 = 2.9972 m for the oxide core
 = 3.2004 m for the metallic core
- ZOUTEL(8) Same as for element 2.
 For both oxide and metallic cores, this is elevation 0.0.
- ZOUTEL(9) Same as for element 3.
 = 1.7272 m for the oxide core
 = 2.3368 m for the metallic core
- ZOUTEL(10) Same as for element 4.
 = 2.9972 m for the oxide core
 = 3.2004 m for the metallic core
- ZOUTEL(11) The elevation of the top of the heat transfer region in the IHX. From Fig. D-2, this is estimated as:
 = 80' 11" - 654" = 317"
 = 8.05182 m for the oxide core

 = 80' 11" - 646" = 325"
 = 8.25500 m for the metallic core
- ZOUTEL(12) The elevation of the bottom of the heat transfer region in the IHX. Using the tube length in Fig. D-3, the elevation is estimated from Fig. D-2 as:
 = 80' 11" - 26' - 654" = 5"
 = 0.12700 m for the oxide core

 = 80' 11" - 26' - 646" = 13"
 = 0.33020 m for the metallic core
- ZOUTEL(13) The elevation of the exit from the IHX into the cold pool is estimated from Fig. D-6 as:
 = 45' 6" - 654" = -108"
 = -2.7432 m for the oxide core

 = 45' 6" - 646" = -100"
 = -2.5400 m for the metallic core
- ZOUTEL(14) The elevation of the primary pump impeller is estimated from Fig. D-7.
 = 100' - 108" - 654" = 138"
 = 3.5052 m for the oxide core
 = 100' - 108" - 646" = 146"
 = 3.7084 m for the metallic core

Fig. D-6. Schematic of the Intermediate Heat Exchanger in LSPB.

Fig. D-7. Schematic of the Two-Stage Primary Pump in LSPB.

ZOUTEL(15) The elevation of the primary pump outlet is estimated from Fig. D-7.

$$= 77' 2" - 654" = 272"$$

$$= 6.9088 \text{ m for the oxide core}$$

$$= 77' 2" - 646" = 280"$$

$$= 7.1120 \text{ m for the metallic core}$$

ZOUTEL(16) The elevation of the exit of the vertical section of the pump discharge pipe is estimated from Fig. D-3 as:

$$= 43' - 654" = -138"$$

$$= -3.5052 \text{ m for the oxide core}$$

$$= 43' - 646" = -130"$$

$$= -3.3020 \text{ m for the metallic core}$$

ZOUTEL(17) The elevation of the exit of the horizontal section of the pump discharge pipe is the same as the inlet:

$$= 43' - 654" = -138"$$

$$= -3.5052 \text{ m for the oxide core}$$

$$= 43' - 646" = -130"$$

$$= -3.3020 \text{ m for the metallic core}$$

ZOUTEL(18) Same as element 16.

$$= -3.5052 \text{ m for the oxide core}$$

$$= -3.3020 \text{ m for the metallic core}$$

ZOUTEL(19) Same as element 17.

$$= -3.5052 \text{ m for the oxide core}$$

$$= -3.3020 \text{ m for the metallic core}$$

Since the two primary pump groups are identical in design and position in the reactor, the elevations are the same.

ZOUTEL(20) Same as element 14.

$$= 3.5052 \text{ m for the oxide core}$$

$$= 3.7084 \text{ m for the metallic core}$$

ZOUTEL(21) Same as element 15.

$$= 6.9088 \text{ m for the oxide core}$$

$$= 7.1120 \text{ m for the metallic core}$$

ZOUTEL(22) Same as element 16.

$$= -3.5052 \text{ m for the oxide core}$$

$$= -3.3020 \text{ m for the metallic core}$$

ZOUTEL(23) Same as element 17.

$$= -3.5052 \text{ m for the oxide core}$$

$$= -3.3020 \text{ m for the metallic core}$$

ZOUTEL(24) Same as element 18.

$$= -3.5052 \text{ m for the oxide core}$$

$$= -3.3020 \text{ m for the metallic core}$$

- ZOUTEL(25) Same as element 19.
 = -3.5052 m for the oxide core
 = -3.3020 m for the metallic core
- ZOUTEL(26) The elevation of the exit of the pipe connecting the hot pool to the region between the upper baffle plate and the redan is based on the length assumed in Loc. 327, and the inlet elevation of the segment;
 = $4.699 - 1.7018 - 0.1$
 = 2.8972 m for the oxide core

 = $4.699 - 1.7018 - 0.1 + 0.2032$
 = 3.1004 m for the metallic core
- ZOUTEL(27) The elevation of the exit of the pipe connecting the cold pool and the region between the lower baffle plate and the redan is based on the length assumed in Loc. 328, and the inlet elevation of the segment;
 = $4.699 - 1.7018 - 1.524 + 0.1$
 = 1.5732 m for the oxide core

 = $4.699 - 1.7018 - 1.524 + 0.1 + 0.2032$
 = 1.7764 m for the metallic core
- ZOUTEL(28) The exit elevation of the primary side inlet section for the DRACS-I heat exchanger is estimated from Fig. D-8.
 = $78.75' - 654" = 291"$
 = 7.3914 m for the oxide core

 = $78.75' - 646" = 299"$
 = 7.5946 m for the metallic core
- ZOUTEL(29) The elevation of the bottom of the heat transfer section of the DRACS-I heat exchanger is estimated from Fig. D-8 as:
 = $66' 7" - 654" = 145"$
 = 3.6830 m for the oxide core

 = $66' 7" - 646" = 153"$
 = 3.8862 m for the metallic core
- ZOUTEL(30) The exit elevation of the primary side outlet section for the DRACS-I heat exchanger is assumed to be the same as the bottom of the heat transfer section:
 = 3.6830 m for the oxide core
 = 3.8862 m for the metallic core
- ZOUTEL(31) Assumed to be the same as for element 28.
 = $78.75' - 654" = 291"$
 = 7.3914 m for the oxide core
 = $78.75' - 646" = 299"$
 = 7.5946 m for the metallic core

Fig. D-8. Schematic of the Decay Heat Removal Heat Exchanger in LSPB.

- ZOUTEL(32) Assumed to be the same as for element 29.
 = 3.6830 m for the oxide core
 = 3.8862 m for the metal core
- ZOUTEL(33) The exit elevation of the outlet section of the DRACS-II heat exchanger is based on the assumed length of 0.1 m.
 = 3.6830 - 0.1 m
 = 3.5830 m for the oxide core

 = 3.8862 - 0.1 m
 = 3.7862 m for the metallic core
- ZOUTEL(34) The elevation of the outlet from the intermediate loop pump is based on the assumption that the cold return pipe is horizontal. The elevation is the same as the segment inlet elevation.
 = 117' - 654" = 750"
 = 19.0500 m for the oxide core

 = 117' - 646" = 758"
 = 19.2532 m for the metallic core
- ZOUTEL(35) Same as element 34.
 = 19.0500 m for the oxide core
 = 19.2532 m for the metallic core
- ZOUTEL(36) The elevation of the bottom of the heat transfer section is the same as for element 12;
 = 0.1270 m for the oxide core
 = 0.3302 m for the metallic core
- ZOUTEL(37) The elevation of the top of the heat transfer section is the same as for element 11;
 = 8.05182 m for the oxide core
 = 8.25500 m for the metallic core
- ZOUTEL(38) The exit elevation of the vertical pipe from the IHX is the same as the elevation of the hot sodium pipe going to the steam generator. From Fig. D-4:
 = 106' - 654" = 618"
 = 15.6972 m for the oxide core

 = 106' - 646" = 626"
 = 15.9004 m for the metallic core
- ZOUTEL(39) The elevation of the horizontal pipe carrying the hot sodium from the IHX to the steam generator is given on Fig. D-2, and is the same as for element 38:
 = 15.6972 m for the oxide core
 = 15.9004 m for the metallic core

- ZOUTEL(40) The elevation of the outlet from the steam generator is taken from Fig. D-4 using the straight-tube steam generator as;
 $= 31' 10'' - 654'' = -272''$
 $= -6.9088 \text{ m}$ for the oxide core
 $= 31' 10'' - 646'' = -264''$
 $= -6.7056 \text{ m}$ for the metallic core
- ZOUTEL(41) The exit elevation of the vertical pipe connecting the steam generator with the pipe tee is assumed to be the same as for the horizontal cold return pipe.
 $= 19.0500 \text{ m}$ for the oxide core
 $= 19.2532 \text{ m}$ for the metallic core
- ZOUTEL(42) The exit elevation of the horizontal pipe connecting the pipe tee with the pump volume is assumed to be the same as for the horizontal return pipe.
 $= 19.0500 \text{ m}$ for the oxide core
 $= 19.2532 \text{ m}$ for the metallic core
- ZOUTEL(43) The exit elevation of the pipe from the pipe tee to the sodium expansion tank taken from Fig. D-4,
 $= 132' 10'' - 654'' = 940''$
 $= 23.8760 \text{ m}$ for the oxide core
 $= 132' 10'' - 646'' = 948''$
 $= 24.0792 \text{ m}$ for the metallic core
- ZOUTEL(44) The elevation of the exit of the horizontal pipe from the sodium expansion tank to the air dump heat exchanger in DRACS-I is the same as the height of the inlet to segment 19:
 $= 33.0708 \text{ m}$ for the oxide core
 $= 33.2740 \text{ m}$ for the metallic core
- ZOUTEL(45) The elevation of the outlet of the air dump heat exchanger in DRACS-I. From Fig. D-9, the height of the tube section of the air dump heat exchanger is estimated at 4', or 1.2192 m.
 $= \text{ZOUTEL(44)} - 1.2192 \text{ m}$
 $= 31.8516 \text{ m}$ for the oxide core
 $= 32.0548 \text{ m}$ for the metallic core
- ZOUTEL(46) The exit elevation of the horizontal pipe from the air dump heat exchanger is the same as for element 45.
 $= 31.8516 \text{ m}$ for the oxide core
 $= 32.0548 \text{ m}$ for the metallic core
- ZOUTEL(47) The elevation of the bottom of the heat transfer region in the DRACS-I heat exchanger. This is the same as for element 29.
 $= 3.6830 \text{ m}$ for the oxide core
 $= 3.8862 \text{ m}$ for the metallic core

Fig. D-9. Elevations for the Air Dump Heat Exchanger in LSPB.

ZOUTEL(48) The elevation of the top of the heat transfer region in the DRACS-I heat exchanger. This is the same as for element 28.
 = 7.3914 m for the oxide core
 = 7.5946 m for the metallic core

ZOUTEL(49) The exit elevation of the vertical pipe from the heat exchanger up to the sodium expansion tank. This is assumed to be at the same level as the pipe leaving the sodium expansion tank.
 = 33.0708 m for the oxide core
 = 33.2740 m for the metallic core

ZOUTEL(50) The exit elevation of the horizontal pipe into the sodium expansion tank is the same as for element 49:
 = 33.0708 m for the oxide core
 = 33.2740 m for the metallic core

The model assumes that the two decay heat removal system loops are the same, even though there are some differences between them. The differences in elevation are minor, and have no effect on the transients.

ZOUTEL(51) Same as element 44.
 = 33.0708 m for the oxide core
 = 33.2740 m for the metallic core

ZOUTEL(52) Same as element 45.
 = 31.8516 m for the oxide core
 = 32.0548 m for the metallic core

ZOUTEL(53) Same as element 46.
 = 31.8516 m for the oxide core
 = 32.0548 m for the metallic core

ZOUTEL(54) Same as element 47.
 = 3.6830 m for the oxide core
 = 3.8862 m for the metallic core

ZOUTEL(55) Same as element 48.
 = 7.3914 m for the oxide core
 = 7.5946 m for the metallic core

ZOUTEL(56) Same as element 49.
 = 33.0708 m for the oxide core
 = 33.2740 m for the metallic core

ZOUTEL(57) Same as element 50.
 = 33.0708 m for the oxide core
 = 33.2740 m for the metallic core

#302-441 XLENEL The length of each element. The lengths of the elements
 (IELL) are the same for the oxide and metallic cores except for liquid segments 2, 3, and 4.

- XLENEL(1) The distance between the subassembly inlet and outlet.
The core channels have a length of 185", so to be
consistent with the core modelling,
= 185"
= 4.669 for both oxide and metallic cores
- XLENEL(2) The length of the lowest region in bypass channel 1.
= ZOUTEL(2) - ZINL(2)
= 1.7018 m for the oxide core
= 1.4986 m for the metallic core
- XLENEL(3) The length of the central region of bypass channel 1.
= ZOUTEL(3) - ZINL(3)
= 1.7272 m for the oxide core
= 2.3368 m for the metallic core
- XLENEL(4) The length of the highest region in bypass channel 1.
= ZOUTEL(4) - ZOUTEL(3)
= 1.2700 m for the oxide core
= 0.8636 m for the metallic core
- XLENEL(5) Same as element 2.
= 1.7018 m for the oxide core
= 1.4986 m for the metallic core
- XLENEL(6) Same as element 3.
= 1.7272 m for the oxide core
= 2.3368 m for the metallic core
- XLENEL(7) Same as element 4.
= 1.2700 m for the oxide core
= 0.8636 m for the metallic core
- XLENEL(8) Same as element 2.
= 1.7018 m for the oxide core
= 1.4986 m for the metallic core
- XLENEL(9) Same as element 3.
= 1.7272 m for the oxide core
= 2.3368 m for the metallic core
- XLENEL(10) Same as element 4.
= 1.2700 m for the oxide core
= 0.8636 m for the metallic core
- XLENEL(11) The length of inlet section of the IHX is estimated from
Figs. D-2 and D-3 as 6' 2 1/4", or
= 1.8860 m
- XLENEL(12) The length of the heat transfer region, given by the
distance between the tube sheet faces in Fig. D-3, is;
= ZOUTEL(11) - ZOUTEL(12)
= 26'
= 7.9248 m

XLENEL(13) The length of the outlet section of the IHX is estimated from Figs. D-2 and D-3 as 9'5", or
 $= ZOUTEL(12) - ZOUTEL(13)$
 $= 2.8702 \text{ m}$

XLENEL(14) The length of the inlet nozzle to the primary pump is estimated from Figs. D-3 and D-7 as 21'6".
 $= ZOUTEL(14) - ZINL(6)$
 $= 6.5532 \text{ m}$

XLENEL(15) The length of the primary pump impeller region is estimated from Fig. D-7 as 12'2".
 $= ZOUTEL(15) - ZOUTEL(14)$
 $= 3.4036 \text{ m}$

XLENEL(16) The length of the vertical section of the pump discharge pipe in the first primary pump group is estimated from Fig. D-3.
 $= ZINL(7) - ZOUTEL(16)$
 $= 3.6068 \text{ m}$

XLENEL(17) The length of the horizontal section of the pump discharge pipe of the first primary pump group is estimated from Fig. D-3.
 $= 13' 3"$
 $= 4.0386 \text{ m}$

XLENEL(18) Same as element 16.
 $= 3.6068 \text{ m}$

XLENEL(19) Same as element 17.
 $= 4.0386 \text{ m}$

Since the two pump groups are identical, the length of an element in one pump group is equal to the corresponding element in the other.

XLENEL(20) Same as element 14.
 $= 6.5532 \text{ m}$

XLENEL(21) Same as element 15.
 $= 3.4036 \text{ m}$

XLENEL(22) Same as element 16.
 $= 3.6068 \text{ m}$

XLENEL(23) Same as element 17.
 $= 4.0386 \text{ m}$

XLENEL(24) Same as element 18.
 $= 3.6068 \text{ m}$

XLENEL(25) Same as element 19.
 $= 4.0386 \text{ ms}$

- XLENEL(26) Element 26 is a fictitious pipe simulating the connection between the outlet plenum and the region between the upper baffle plate and the redan. The length of the pipe is assumed to be 0.1 m, which is an arbitrary number.
= 0.1 m
- XLENEL(27) Element 27 is a fictitious pipe simulating the connection between the cold pool and the region between the lower baffle plate and the redan. The length of the pipe is assumed to be 0.1 m, which is an arbitrary number.
= 0.1 m
- XLENEL(28) The length of the primary side inlet section of the DRACS-I heat exchanger is estimated from Fig. D-8 by assuming that the length is 1/4 of the diameter:
= 8.50"
= 0.2159 m
- XLENEL(29) The length of the heat transfer region in DRACS-I is estimated from Fig. D-8 :
= ZOUTEL(28) - ZOUTEL(29)
= 7.3914 - 3.6830
= 3.7084 m
- XLENEL(30) The length of the primary side outlet section in DRACS-I is assumed to be the same as the inlet section.
= 0.2159 m
- XLENEL(31) The length of primary side inlet section in the DRACS-II heat exchanger is estimated at 6'3" from Figs D-2 and D-8.
= 1.9050 m
- XLENEL(32) The length of the heat transfer region in the DRACS-II heat exchanger is assumed to be the same as for DRACS-I, element 29.
= 3.7084 m
- XLENEL(33) The length of the exit section of the DRACS-II heat exchanger is arbitrarily assumed to be 0.1 m.
= 0.1000 m
- XLENEL(34) The length of the intermediate loop pump is estimated from Fig. D-4 as 36", which is rounded-off to 1.0 m.
= 127' 3" - 124' 3" = 36"
= 1.0000 m
- XLENEL(35) The length of the horizontal pipe between the intermediate loop pump and the vertical pipe connecting to the IHX is estimated as [D-4],
= 692.8"
= 17.5970 m

- XLENEL(36) The length of the vertical downcomer for the IHX is estimated from Fig. D-4.
 $= ZOUTEL(35) - ZOUTEL(36)$
 $= 19.0500 - 0.1270$
 $= 18.9230 \text{ m}$
- XLENEL(37) The length of the heat transfer region on the secondary side of the IHX is the same as for the primary side, element 12.
 $= 26'$
 $= 7.9248 \text{ m}$
- XLENEL(38) The length of the vertical pipe from the heat transfer section to the horizontal pipe which goes to the steam generator is estimated from Fig. D-3.
 $= ZOUTEL(38) - ZOUTEL(37)$
 $= 7.6454 \text{ m}$
- XLENEL(39) The length of the horizontal pipe from directly over the IHX to the steam generator is estimated as [D3],
 $= 22.7076 \text{ m}$
- XLENEL(40) The length of the heat transfer region in the steam generator is estimated from Fig. D-4 as,
 $= ZOUTEL(39) - ZOUTEL(40)$
 $= 22.6060 \text{ m}$
- XLENEL(41) The length of the vertical pipe from the steam generator to the pipe tee is estimated from Fig. D-4 as,
 $= ZOUTEL(41) - ZOUTEL(40)$
 $= 25.9588 \text{ m}$
- XLENEL(42) The distance from the pipe tee to the intermediate loop pump is estimated as [D3],
 $= 6.4771 \text{ m}$
- XLENEL(43) The length of the pipe from the pipe tee to the sodium expansion tank is estimated from Fig. D-4 as,
 $= ZOUTEL(43) - ZINL(18)$
 $= 4.8260 \text{ m}$
- XLENEL(44) The length of the inlet pipe into the air dump heat exchanger in DRACS-I is estimated from Fig. D-5,
 $= 3.6069 \text{ m}$
- XLENEL(45) The length of the tubes in the air dump heat exchanger in DRACS-I is estimated from Fig. D-9 as,
 $= 36.8'$
 $= 11.2167 \text{ m}$
- XLENEL(46) The length of the horizontal pipe from the outlet of the air dump heat exchanger in DRACS-I is estimated from Fig. D-5 as,
 $= 14.2'$
 $= 4.3282 \text{ m}$

- XLENEL(47) The length of the vertical pipe from the air dump heat exchanger outlet line to the bottom of the heat transfer region in the DRACS-I heat exchanger is estimated from Fig. D-5 as,
 $= ZOUTEL(46) - ZOUTEL(47)$
 $= 28.1686 \text{ m}$
- XLENEL(48) The heat transfer length of the secondary side of the DRACS-I heat exchanger is the same as element 29,
 $= ZOUTEL(48) - ZOUTEL(47)$
 $= 3.7084 \text{ m}$
- XLENEL(49) The length of the vertical pipe between the heat exchanger and the elevation of the sodium expansion tank is estimated from Fig. D-4 as,
 $= ZOUTEL(49) - ZOUTEL(48)$
 $= 25.6794 \text{ m}$
- XLENEL(50) The length of the horizontal pipe to the sodium expansion tank is arbitrarily assumed to be 1.0 m.
 $= 1.0000 \text{ m}$

The modelling of the DRACS-II loop is identical to the DRACS-I loop model.

- XLENEL(51) Same as element 44.
 $= 3.6069 \text{ m}$
- XLENEL(52) Same as element 45.
 $= 11.2167 \text{ m}$
- XLENEL(53) Same as element 46.
 $= 4.3282 \text{ m}$
- XLENEL(54) Same as element 47.
 $= 28.1686 \text{ m}$
- XLENEL(55) Same as element 48.
 $= 3.7084 \text{ m}$
- XLENEL(56) Same as element 49.
 $= 25.6794 \text{ m}$
- XLENEL(57) Same as element 50.
 $= 1.0000 \text{ m}$

#442-581 AREAEL (IELL) The cross-sectional flow area of the liquid elements. The cross sectional area of each element is the same for both the metallic and oxide fuel cores.

- AREAEL(1) The flow area for the core region is not used; a non-zero value is input.
 $= 1.0000$

- AREAEL(2) The flow area of the lower region of bypass channel 1 is the flow area of a radial blanket assembly.
= 0.00445836 m**2
- AREAEL(3) The flow area of the central region of bypass channel 1 is the same as for the lower region.
= 0.00445836 m**2
- AREAEL(4) The flow area of the upper region of bypass channel 1 is the same as for the lower region.
= 0.00445836 m**2
- AREAEL(5) The flow area of the lower region in bypass channel 2 is calculated using the area fractions for steel and sodium for the radial shields [D2]. The control rod assemblies are also calculated in a similar manner.
= 1.14510 m**2
- AREAEL(6) The flow area of the central region of bypass channel 2 is the same as for the lower region.
= 1.14510 m**2
- AREAEL(7) The flow area of the upper region of bypass channel 2 is the same as for the lower region.
= 1.14510 m**2
- AREAEL(8) The flow area of the lower region in bypass channel 3 is based on the core bypass flow rate as compared to the flow rate in the radial shields, so that the pressure drop would be equivalent.
= 0.860023 m**2
- AREAEL(9) The flow area of the central region of bypass channel 3 is the same as for the lower region.
= 0.860023 m**2
- AREAEL(10) The flow area of the upper region of bypass channel 3 is the same as for the lower region.
= 0.860023 m**2
- AREAEL(11) The flow area for the entrance of the primary side of the IHX. Estimating from Fig. D-6, the flow into the IHX goes up past a shroud surrounding the IHX [D3] providing a 9" annular region around the outside of the IHX.
= $\pi/4((142 + 18)**2 - 142**2)$
= 4711.5 sq in
= 2.7545 m**2
- AREAEL(12) The primary side sodium is on the tube side of the IHX. The flow area inside the tubes is estimated by:
Number of tubes = 5600
Tube inside diameter = 1.035"
= $5600 \times \pi/4 \times 1.035**2 = 4711.4$ sq in
= 3.0397 m**2

AREAEL(13) The outlet flow area on the primary side of the IHX.
This is estimated from Fig. D-6, assuming the wall
thickness of the nozzle is 0.75".

$$= \pi/4 \times (68.5 \times 0.0254)^2$$

$$= 2.3776 \text{ m}^2$$

AREAEL(14) The flow area for the inlet of a primary pump is taken
from Fig. D-7.

$$= \pi/4 \times (31.5 \times 0.0254)^2$$

$$= 0.50278 \text{ m}^2$$

AREAEL(15) The flow area through the impeller region of a primary
pump is estimated from Fig. D-7.

$$= \pi/4 \times \{(69 \times 0.0254)^2 - (56 \times 0.0254)^2\}$$

$$= 0.8234 \text{ m}^2$$

AREAEL(16) The flow area of each of the two discharge pipes from
a primary pump. The discharge pipe diameter is 24" [D3].

$$= \pi/4 \times (24 \times 0.0254)^2$$

$$= 0.29186 \text{ m}^2$$

AREAEL(17) Same as element 16.

$$= 0.29186 \text{ m}^2$$

AREAEL(18) Same as element 16.

$$= 0.29186 \text{ m}^2$$

AREAEL(19) Same as element 16.

$$= 0.29186 \text{ m}^2$$

Since the two primary pump groups are identical, the flow
areas of the corresponding components are the same.

AREAEL(20) Same as element 14.

$$= 0.50278 \text{ m}^2$$

AREAEL(21) Same as element 15.

$$= 0.8234 \text{ m}^2$$

AREAEL(22) Same as element 16.

$$= 0.29186 \text{ m}^2$$

AREAEL(23) Same as element 17.

$$= 0.29186 \text{ m}^2$$

AREAEL(24) Same as element 18.

$$= 0.29186 \text{ m}^2$$

AREAEL(25) Same as element 19.

$$= 0.29186 \text{ m}^2$$

AREAEL(26) The flow area between the hot pool and the region between
the upper baffle plate and the redan is arbitrarily
assumed to 0.1 m^2 .

$$= 0.1000 \text{ m}^2$$

AREAEL(27) The flow area between the cold pool and the region between the lower baffle plate and the redan is arbitrarily assumed to be 0.1 m^2 .
 $= 0.1000 \text{ m}^2$

AREAEL(28) The flow area of the inlet section of DRACS-I. The flow area consists of a 12" high opening with 80% free area around the circumference of the DRACS-I heat exchanger.
 $= 0.8 * (12 * 0.0254) * (\pi * 32 * 0.0254)$
 $= 0.62264 \text{ m}^2$

AREAEL(29) The flow area on the shell side of DRACS-I is taken from Fig. D-8.

$$\begin{aligned} \text{Shell inside cross-sectional area} \\ &= \pi/4 * \{(32 - 2 * 0.9) * 0.0254\}^2 \\ &= 0.46213 \text{ m}^2 \end{aligned}$$

$$\begin{aligned} \text{Total tube cross-sectional area} \\ &= 306 * \pi/4 * (0.75 * 0.0254)^2 \\ &= 0.087217 \text{ m}^2 \end{aligned}$$

$$\begin{aligned} \text{Downcomer cross-sectional area} \\ &= \pi/4 * (16 * 0.0254)^2 \\ &= 0.12972 \text{ m}^2 \end{aligned}$$

$$\begin{aligned} &= 0.46213 - 0.08217 - 0.12972 \\ &= 0.24520 \text{ m}^2 \end{aligned}$$

AREAEL(30) The flow area of the outlet section of the DRACS-I heat exchanger is assumed to be the same as the inlet flow area.
 $= 0.62264 \text{ m}^2$

AREAEL(31) Same as element 28.
 $= 0.62264 \text{ m}^2$

AREAEL(32) The flow area on the tube side of the DRACS-II heat exchanger. From Fig. D-8, the total tube area is:
 $= 306 * \pi/4 * \{(0.75 - 2 * 0.034) * 0.0254\}^2$
 $= 0.072119 \text{ m}^2$

AREAEL(33) The flow area of the outlet section of the DRACS-II heat exchanger is assumed to be the same as the inlet flow area.
 $= 0.62264 \text{ m}^2$

AREAEL(34) The impeller flow area of the intermediate pump is assumed to be the same as the primary pump, element 15.
 $= 0.8234 \text{ m}^2$

AREAEL(35) The flow area of the horizontal pipe connecting the pump with the IHX is based on an outside diameter of 36", with a wall thickness of 0.5".
 $= \pi/4 * \{(36 - 2 * 0.5) * 0.0254\}^2$
 $= 0.62072 \text{ m}^2$

AREAEL(36) The flow area of the vertical pipe and downcomer in the IHX. Using the same 36" pipe with a 0.5" wall thickness,
 $= 0.62072 \text{ m}^2$

AREAEL(37) The flow area of the shell side of the IHX is estimated from Ref. D2 and D3 as,

$$\begin{aligned} \text{Flow area across the shell} \\ &= \pi/4 \times (11 \times 12 \times 0.0254)^2 \\ &= 8.82887 \text{ m}^2 \end{aligned}$$

$$\begin{aligned} \text{Total cross-sectional area of tubes} \\ &= 5000 \times \pi/4 \times (1.125 \times 0.0254)^2 \\ &= 3.59129 \text{ m}^2 \end{aligned}$$

$$\begin{aligned} \text{Cross-sectional area of downcomer} \\ &= \pi/4 \times (36 \times 0.0254)^2 \\ &= 0.65669 \text{ m}^2 \end{aligned}$$

$$\begin{aligned} &= 8.82887 - 3.59129 - 0.65669 \\ &= 4.58089 \text{ m}^2 \end{aligned}$$

AREAEL(38) The flow area of the outlet pipe from the IHX is assumed to be 36" pipe with a 0.5" wall thickness.
 $= \pi/4 \times \{(36 - 2 \times 0.5) \times 0.0254\}^2$
 $= 0.62072 \text{ m}^2$

AREAEL(39) The flow area of the horizontal pipe connecting the steam generator to the IHX is based on the same 36" pipe with a 0.5" wall thickness.
 $= 0.62072 \text{ m}^2$

AREAEL(40) The flow area of the steam generator is set to provide the appropriate pressure drop for the intermediate loop. This is established from a PRIMAR-4 calculation.
 $= 0.882 \text{ m}^2$

AREAEL(41) The flow area of the vertical return pipe from the steam generator is based on a 36" pipe with a 0.5" wall thickness.
 $= \pi/4 \times \{(36 - 2 \times 0.5) \times 0.0254\}^2$
 $= 0.62072 \text{ m}^2$

AREAEL(42) The flow area of the horizontal pipe connecting the pipe tee with the pump volume is based on a 36" pipe with a 0.5" wall thickness.
 $= \pi/4 \times \{(36 - 2 \times 0.5) \times 0.0254\}^2$
 $= 0.62072 \text{ m}^2$

AREAEL(43) The flow area of the pipe connecting the pipe tee with the sodium expansion tank is based on a pipe diameter of 18" with a wall thickness of 0.25".
 $= \pi/4 \times \{(18 - 2 \times 0.25) \times 0.0254\}^2$
 $= 0.15518 \text{ m}^2$

AREAEL(44) The flow area of the horizontal pipe from the sodium expansion tank to the air dump heat exchanger in DRACS-I is based on a pipe inside diameter of 14.883", Fig. D-8.
 $= \pi/4 \times (14.883 \times 0.0254)^2$
 $= 0.11224 \text{ m}^2$

AREAEL(45) The flow area through the air dump heat exchanger of DRACS-I is taken from Fig. D-9, with the pipe OD of 1.9". Assume the wall thickness is 0.075".
 $= 63 \times \pi/4 \times [(1.9 - 2 \times 0.075) \times 0.0254]^2$
 $= 0.097763 \text{ m}^2$

AREAEL(46) Same as element 44.
 $= 0.11224 \text{ m}^2$

AREAEL(47) Same as element 44.
 $= 0.11224 \text{ m}^2$

AREAEL(48) The flow area on the tube side of the DRACS-I heat exchanger is the same as for element 32.
 $= 0.072119 \text{ m}^2$

AREAEL(49) Same as element 44.
 $= 0.11224 \text{ m}^2$

AREAEL(50) Same as element 44.
 $= 0.11224 \text{ m}^2$

The designs of DRACS-I and DRACS-II are identical, and the corresponding flow areas are the same. Note that in DRACS-I the primary flow is on the shell side while in DRACS-II the primary flow is on the tube side.

AREAEL(51) Same as element 44.
 $= 0.11224 \text{ m}^2$

AREAEL(52) Same as element 45.
 $= 0.097763 \text{ m}^2$

AREAEL(53) Same as element 44.
 $= 0.11224 \text{ m}^2$

AREAEL(54) Same as element 44.
 $= 0.11224 \text{ m}^2$

AREAEL(55) Same as element 29.
 $= 0.2452 \text{ m}^2$

AREAEL(56) Same as element 44.
 $= 0.11224 \text{ m}^2$

AREAEL(57) Same as element 44.
 $= 0.11224 \text{ m}^2$

- #582-721 DHELEM (IELL) The hydraulic diameter of the liquid elements.
For the following locations the hydraulic diameter of an element in the oxide core is the same as the corresponding element in the metallic core.
- DHELEM(1) The hydraulic diameter in the core is not used in the calculation, so a non-zero number is input.
= 1.0000
- DHELEM(2) The hydraulic diameter of the lower region of bypass channel 1 is the same as for the internal blanket assemblies in the core channels.
= 3.06025D-03 m
- DHELEM(3) The hydraulic diameter of the central region of bypass channel 1 is the same as for the lower region.
= 3.06025D-03 m
- DHELEM(4) The hydraulic diameter of the upper region of bypass channel 1 is the same as for the lower region.
= 3.06025D-03 m
- DHELEM(5) The hydraulic diameter of the lower region of bypass channel 2 is based on the sodium and steel fractions for the radial shields and comparing with the same values for driver and blanket assemblies. The estimate is 2.5×10^{-3} ,
= 0.00250 m
- DHELEM(6) Same as for element 5
= 0.00250 m
- DHELEM(7) Same as for element 5
= 0.00250 m
- DHELEM(8) The hydraulic diameter of the lower region of bypass channel 3 is set to 0.5 for consistency with the heat transfer and flow calculations.
= 0.5 m
- DHELEM(9) Same as for element 8.
= 0.5 m
- DHELEM(10) Same as for element 8.
= 0.5 m
- DHELEM(11) The hydraulic diameter of the inlet section of the IHX. Using the gap between the shroud and the outside of the IHX of 9", and an IHX outside diameter of 142";
Wetted perimeter
= $\pi \times (142 + 2 \times 9) + \pi \times (142)$
= 948.76"
= 24.0985 m

$$\begin{aligned}\text{Flow area} &= \text{AREAEL}(11) \\ &= 2.7545 \text{ m}^2\end{aligned}$$

$$\begin{aligned}\text{Hydraulic Diameter (DH)} \\ &= 4 \times 2.7545 / 24.0985 \\ &= 0.4572 \text{ m}\end{aligned}$$

DHELEM(12) The hydraulic diameter of the tubes in the IHX is based on information in Ref. D2.

$$\begin{aligned}\text{Tube ID} &= 1.125 - 2 \times 0.045 \\ &= 1.035" \\ &= 0.02629 \text{ m}\end{aligned}$$

DHELEM(13) The hydraulic diameter of the outlet section of the IHX.

$$\begin{aligned}\text{From Fig. D-6,} \\ &= 70 - 2 \times 0.75 = 68.5" \\ &= 1.7399 \text{ m}\end{aligned}$$

DHELEM(14) The hydraulic diameter of the inlet section of a primary loop pump is taken from Fig. D-7,

$$\begin{aligned}&= 31.5" \\ &= 0.8001 \text{ m}\end{aligned}$$

DHELEM(15) The hydraulic diameter of the impeller region of a primary loop pump is estimated from Fig. D-7.

$$\begin{aligned}&= (69" - 56") \times 0.0254 \\ &= 0.3302 \text{ m}\end{aligned}$$

DHELEM(16) The hydraulic diameter of a discharge pipe from a primary loop pump is based on a 24" diameter pipe.

$$= 0.60960 \text{ m}$$

DHELEM(17) Same as for element 16.

$$= 0.60960 \text{ m}$$

DHELEM(18) Same as for element 16.

$$= 0.60960 \text{ m}$$

DHELEM(19) Same as for element 16.

$$= 0.60960 \text{ m}$$

DHELEM(20) Same as for element 14.

$$= 0.8001 \text{ m}$$

DHELEM(21) Same as for element 15.

$$= 0.3302 \text{ m}$$

DHELEM(22) Same as for element 16.

$$= 0.6096 \text{ m}$$

DHELEM(23) Same as for element 16.

$$= 0.6096 \text{ m}$$

DHELEM(24) Same as for element 16.
 $= 0.6096 \text{ m}$

DHELEM(25) Same as for element 16.
 $= 0.6096 \text{ m}$

DHELEM(26) The hydraulic diameter of the pipe connecting the hot pool and the region between the upper baffle plate and the redan is based on an assumed flow area of 0.10 m^2 .
 $= (4 \times 0.1 / \pi)^{1/2}$
 $= 0.35682 \text{ m}$

DHELEM(27) Same as for element 26.
 $= 0.35682 \text{ m}$

DHELEM(28) The hydraulic diameter of the inlet section of the shell side of the DRACS-I heat exchanger. From Fig. D-8,
Wetted perimeter
 $= 2 \times (12 + \pi \times 32 \times 0.8)$
 $= 184.85"$
 $= 4.69518 \text{ m}$
Flow Area = AREAEL(28) = 0.62264 m^2
DHELEM(28) = $4 \times 0.62264 / 4.69519$
 $= 0.53045 \text{ m}$

DHELEM(29) The hydraulic diameter of the shell side of the DRACS-I heat exchanger. From Fig. D-8,
Wetted perimeter inside the shell
 $= \pi \times \{(32 - 2 \times 0.9) \times 0.0254\}$
 $= 2.40985 \text{ m}$
Wetted perimeter of the tubes
 $= \pi \times 306 \times (0.75 \times 0.0254)$
 $= 18.31328 \text{ m}$
Wetted perimeter of the downcomer
 $= \pi \times 16 \times 0.0254$
 $= 1.27674 \text{ m}$
Total wetted perimeter
 $= 2.40985 + 18.31328 + 1.27674$
 $= 21.99987 \text{ m}$
Flow area = AREAEL(29) = 0.24520 m^2
DHELEM(29) = $4 \times 0.24520 / 21.99987$
 $= 0.04458 \text{ m}$

DHELEM(30) The hydraulic diameter of the outlet section of the shell side of the DRACS-I heat exchanger. This is assumed to be the same as for the inlet section, DHELEM(28),
 $= 0.53045 \text{ m}$

DHELEM(31) The hydraulic diameter of the inlet section of the tube side of the DRACS-II heat exchanger. For the inlet section, it is assumed that the hydraulic diameter is the same as for DRACS-I, even though the primary side of

DRACS-I is the shell side.
 = DHELEM(28)
 = 0.53045 m

DHELEM(32) The hydraulic diameter of the tubes in the DRACS-II heat exchanger. From Fig. D-8,
 = $(0.75" - 2 \times 0.034") \times 0.0254$
 = 0.01732 m

DHELEM(33) The hydraulic diameter of the outlet section of the tube side of the DRACS-II heat exchanger is assumed to be the same as for the inlet section;
 = DHELEM(31)
 = 0.53045 m

DHELEM(34) The hydraulic diameter of the impeller region of the intermediate loop pump is assumed to be the same as for the primary pump;
 = DHELEM(15)
 = 0.3302 m

DHELEM(35) The hydraulic diameter of the horizontal cold return pipe from the intermediate loop pump to the IHX. From Fig. D-4, the diameter of this pipe is 36". Assume the wall thickness is 0.5".
 = $(36" - 2 \times 0.5") \times 0.0254$
 = 0.88900 m

DHELEM(36) The hydraulic diameter of the vertical pipe into the bottom of the IHX is the same as for element 35.
 = DHELEM(35)
 = 0.88900 m

DHELEM(37) The hydraulic diameter of the shell side of the IHX. The inside diameter is estimated at 11' from Ref. D2 and D3.

Wetted perimeter of the shell side
 = $\pi \times (11' \times 12 \times 0.0254)$
 = 10.53313 m

Wetted perimeter of the tube bundle
 = $5600 \times \pi \times (1.125" \times 0.0254)$
 = 502.7177 m

Wetted perimeter of downcomer
 = $\pi \times (36 \times 0.0254)$
 = 2.87267 m

Total wetted perimeter
 = $0.53313 + 502.71770 + 2.87267$
 = 516.1235

Flow area = AREAEL(37)
 = 4.58089 m**2

DHELEM(37) = $4 \times 4.58089 / 516.1235$
 = 0.03502 m

- DHELEM(38) The hydraulic diameter of the vertical outlet pipe from the IHX is the same as for element 35.
= 0.88900 m
- DHELEM(39) The hydraulic diameter of the horizontal pipe from the IHX to the steam generator is the same as for element 35.
= 0.88900 m
- DHELEM(40) The hydraulic diameter of the shell side of the steam generator is assumed to be the same as for the IHX, with the flow area adjusted to give the proper pressure drop.
= 0.03502 m
- DHELEM(41) The hydraulic diameter of the vertical pipe connecting the steam generator with the pipe tee is the same as for element 35.
= 0.88900 m
- DHELEM(42) The hydraulic diameter of the horizontal pipe connecting the pipe tee with the intermediate loop pump volume is the same as for element 35.
= 0.88900 m
- DHELEM(43) The hydraulic diameter of the pipe connecting the pipe tee and the sodium expansion tank is based on the pipe dimensions assumed for calculating the flow area.
= $(18" - 2 \times 0.25") \times 0.0254$
= 0.44450 m
- DHELEM(44) The hydraulic diameter of the horizontal pipe from the sodium expansion tank to the air dump heat exchanger in the DRACS-I loop. Based on the pipe dimensions used for calculating the flow area;
= $14.833" \times 0.0254$
= 0.37803 m
- DHELEM(45) The hydraulic diameter of the tubes in the air dump heat exchanger in DRACS-I. Based on the pipe dimensions used for calculating the flow area;
= $(1.9" - 2 \times 0.075") \times 0.0254$
= 0.04445 m
- DHELEM(46) The hydraulic diameter of the horizontal pipe connecting the air dump heat exchanger with the DRACS-I heat exchanger is the same as for element 44.
= 0.37803 m
- DHELEM(47) The hydraulic diameter of the vertical pipe connecting to the bottom of the DRACS-I heat exchanger is the same as for element 44.
= 0.37803 m

- DHELEM(48) The hydraulic diameter of the tubes in the DRACS-I heat exchanger. From Fig. D-8,
 $= (0.75" - 2 \times 0.034") \times 0.0254$
 $= 0.017323 \text{ m}$
- DHELEM(49) The hydraulic diameter of the vertical pipe exiting the heat exchanger is the same as for element 44.
 $= 0.37803 \text{ m}$
- DHELEM(50) The hydraulic diameter of the horizontal pipe connecting to the sodium expansion tank is the same as for element 44.
 $= 0.37803 \text{ m}$

Since the designs of DRACS-I and DRACS-II are assumed to be identical for modelling purposes, the hydraulic diameters of the corresponding elements are equal. In DRACS-I, the primary flow is through the shell side while in DRACS-II, the primary flow is through the tube side.

- DHELEM(51) Same as element 44.
 $= 0.37803 \text{ m}$
- DHELEM(52) Same as element 45.
 $= 0.04445 \text{ m}$
- DHELEM(53) Same as element 44.
 $= 0.37803 \text{ m}$
- DHELEM(54) Same as element 44.
 $= 0.37803 \text{ m}$
- DHELEM(55) Same as element 29.
 $= 0.04458 \text{ m}$
- DHELEM(56) Same as element 44.
 $= 0.37803 \text{ m}$
- DHELEM(57) Same as element 44.
 $= 0.37803 \text{ m}$

#722-778 ROUGHL (IELL) The pipe surface roughness for the friction factor in the pipe pressure drop calculation. Assume a 0.00001 relative roughness for a relatively smooth surface in all pipes.
 $= 0.00001$ for all elements

#862-917 BENDNM (IELL) The number of bends in each liquid element. The number of bends in each element is input to account for the additional pressure drop associated with each bend. Only those elements which contain bends are reviewed.

There is one bend in each horizontal and vertical section of the outlet pipes from the primary pumps.
 BENDNM(16) thru BENDNM(19) = 1.0
 BENDNM(22) thru BENDNM(25) = 1.0

It is assumed that there are three bends in the pipes connecting the intermediate loop pump and the IHX, the IHX and the steam generator, and the steam generator and the pipe tee to allow for thermal expansion and changing the pipe direction.

BENDNM(35) = 3.0

BENDNM(39) = 3.0

BENDNM(41) = 3.0

It is assumed that there is one bend in the pipe from the pipe tee to the sodium expansion tank.

BENDNM(43) = 1.0

DRACS-I and DRACS-II are assumed to be the same, so the corresponding elements have the same number of bends.

From Fig. D-8, there are 4 passes per tube in the air dump heat exchanger of DRACS-I. This requires 3 180 degree bends or 6 equivalent 90 deg bends per tube.

BENDNM(45) = 6.0

BENDNM(52) = 6.0

There is one bend to change the direction of the air dump heat exchanger outlet pipe from horizontal to vertical.

BENDNM(46) = 1.0

BENDNM(53) = 1.0

There is one bend to change the direction of the air dump heat exchanger outlet pipe from vertical to horizontal.

BENDNM(49) = 1.0

BENDNM(56) = 1.0

#1002-1141 G2PRDR
(IELL)

Orifice coefficients which are added to the elements to provide additional pressure drop. These are normally zero as input, with PRIMAR-4 calculating these coefficients during the steady-state to initialize the model. Some values need to be set to provide the correct free surface levels in the volumes. These values are determined by trial-and-error using the PRIMAR-4 steady-state calculation.

G2PRDR(12) = 7.25840

G2PRDR(16) = 1.63623

G2PRDR(18) = 1.63623

G2PRDR(22) = 1.63623

G2PRDR(24) = 1.63623

G2PRDR(37) = 43.66180

#1042 BNDLOD

The equivalent length of pipe, in diameters, which provides the same pressure drop as a smooth 90 degree bend.
= 12.0

#1143-1199 WALLMC

The pipe wall mass x heat capacity per unit length. The following values are used for density x specific heat capacity calculated at 725 deg K. ($\rho \times C_p$)

HT-9	4.9460D+6 J/M**3-K
316-SS	4.4970D+6 J/M**3-K
304-SS	4.4190D+6 J/M**3-K
9Cr-1Mo	4.5800D+6 J/M**3-K

WALLMC(1) The value for the core is not used by the code, so a non-zero value is input.
= 1.0000

WALLMC(2) This element represents the lower region of a radial blanket subassembly. The value is estimated as follows:
Area fraction of pins and duct = $0.495 + 0.275 = 0.770$
Subassembly pitch is 6.131"
Area per subassembly is 32.5532 sq in.
Area of pins and duct per subassembly is 25.0660 sq in.
Assume that this area all has the properties of HT-9:
HT-9 density = 7760 kg/m**3 at 298 K
Subassembly steel mass/length = 125.491 kg/m
HT-9 heat capacity = 615 J/kg-K at 673 K

$$\begin{aligned} \text{WALLMC}(2) &= (125.491)(615) \\ &= 7.71770\text{D}+4 \text{ J/m-K} \end{aligned}$$

WALLMC(3) Same as for element 2.
= 7.71770D+4 J/m-K

WALLMC(4) Same as for element 2.
= 7.71770D+4 J/m-K

WALLMC(5) This element represents the lower section of the radial shield and control assemblies. Using the same method as for element 2,
Area fraction of steel in the shields = 0.815
Area of steel per shield assembly = 26.5309 sq in.
Subassembly steel mass/length = 132.825 kg/m
Shield mass x heat capacity/length
= $306 (132.825)(615)$
= $2.49963 \times 10^7 \text{ kg/m}$
Area fraction of steel in the control assy's = 0.602
Area of steel per control assembly = 19.5970 sq in.
Subassembly steel mass/length = 98.1114 kg/m
Control assembly mass x heat capacity/length
= $36 (98.1114)(615)$
= $2.17219 \times 10^6 \text{ kg/m}$

$$\begin{aligned} \text{WALLMC}(5) &= 2.49963 \times 10^7 + 2.17219 \times 10^6 \\ &= 2.71657 \times 10^7 \text{ kg/m-K} \end{aligned}$$

WALLMC(6) Same as for element 5.
= 2.71657D+7 J/m-K

WALLMC(7) Same as for element 5.
= 2.71657D+7 J/m-K

WALLMC(8) This element represents the core bypass flow region which has uncertain associated structure. An arbitrary value of 1.0×10^5 is selected for compatibility with the heat transfer calculations.
 $= 1.000+5 \text{ J/m-K}$

WALLMC(9) Same as for element 8.
 $= 1.000+5 \text{ J/m-K}$

WALLMC(10) Same as for element 8.
 $= 1.000+5 \text{ J/m-K}$

WALLMC(11) This element represents the inlet section of the IHX. Using the outside diameter of the IHX of 142", assume the shell thickness is 2":

Shell O.D. = 142" = 3.6038 m

Shell I.D. = 138" = 3.5052 m

Inlet section height = XLENEL(11) = 1.885 m

Shell volume in the inlet section

$$= \pi/4 \times (3.6038^{**2} - 3.5052^{**2}) \times 1.885$$

$$= 1.03774 \text{ m}^{**3}$$

Tubesheet I.D. = 56.842" = 1.44373 m, from [D3]

Tubesheet thickness = 15" = 0.38100 m, from [D3]

Tubesheet volume

$$= \pi/4 \times (3.6038^{**2} - 1.44378^{**2}) \times 0.381$$

$$= 2.10443 \text{ m}^{**3}$$

(Note that the flow area of the tubes is included, as an approximation for the other miscellaneous steel in the inlet section)

Total volume of steel in the inlet section of the IHX

$$= 1.03774 + 2.10443$$

$$= 3.14167 \text{ m}^{**3}$$

WALLMC(11) = Volume $\times$ (rho $\times$ Cp (9Cr-1Mo))/Length

$$= 3.14167 \times 4.58000+6/1.885$$

$$= 7.633340+6 \text{ J/m-K}$$

WALLMC(12) This element represents the tube region of the IHX. Using the dimensions for calculating the flow areas,

Tube O.D. = 1.125"

Tube wall thickness = 0.045"

Wall cross-sectional area per tube

$$= \pi/4 \times (1.125^{**2} - 1.035^{**2}) \times 0.0254^{**2}$$

$$= 9.8504 \times 10^{-5} \text{ m}^{**2} \text{ per tube}$$

$$\text{WALLMC(12)} = (9.8504 \times 10^{-5})(5600 \text{ tubes})(4.58 \times 10+6)$$

$$= 2.5243 \times 10+6$$

WALLMC(13) This element represents the exit section of the IHX.

Using the outside diameter of the IHX of 142", assume the shell thickness is 2":

Height of the outlet section = XLENEL(13) = 2.8702 m

Shell Volume = $\pi/4 \times (3.6038^{**2} - 3.5052^{**2}) \times 2.8702$

$$= 1.5801 \text{ m}^{**3}$$

Since the shell diameter reduces to the discharge pipe diameter in the outlet section, the volume is reduced by 20% to approximate this effect.

$$\text{Adjusted shell volume} = 0.8 \times 1.5801 = 1.26408 \text{ m}^3$$

Tubesheet thickness = 13.125" = 0.33337 m, from [D3]

Tubesheet volume

$$= \pi/4 \times (3.6032^2 - 1.44378^2) \times 0.33337$$

$$= 1.84135 \text{ m}^3$$

(Note that the flow area of the tubes is included, as an approximation for the other miscellaneous steel in the inlet section)

$$\text{Total volume of steel in the outlet section of the IHX}$$

$$= 1.26408 + 1.84135$$

$$= 3.10543 \text{ m}^3$$

$$\text{WALLMC(13)} = \text{Volume} \times (\rho \times C_p (9\text{Cr-1Mo})) / \text{Length}$$

$$= 3.10543 \times 4.5800 \times 10^6 / 2.8702$$

$$= 4.95536 \times 10^6 \text{ J/m-K}$$

WALLMC(14) This element represents the inlet pipe for the primary pumps. From Fig. D-7,

$$\text{Nozzle inlet I.D.} = 31.5"$$

$$\text{Wall thickness} = 1.0"$$

$$\text{Nozzle inlet O.D.} = 33.5"$$

Wall cross-sectional area

$$= \pi/4 \times \{(33.5 \times 0.0254)^2 - (31.5 \times 0.0254)^2\}$$

$$= 0.065872 \text{ m}^2$$

$$\text{WALLMC(14)} = \text{Area} \times (\rho \times C_p (304\text{-SS}))$$

$$= 0.065872 \times 4.4190 \times 10^6$$

$$= 2.91088 \times 10^5 \text{ J/m-K}$$

WALLMC(15) This element represents the impeller region of the primary pump. Detailed information on the pump is not available, so the following estimate was made.

$$\text{Weight of pump internals} = 69 \text{ tons} = 62595.2 \text{ kg}$$

$$\text{Length from Fig. D-7} = 32' = 9.7536 \text{ m}$$

$$\text{WALLMC(15)} = \text{mass} \times C_p (304\text{-SS}) / \text{Length}$$

$$= 3.6003 \times 10^6 \text{ J/m-K}$$

WALLMC(16) This element represents the vertical section of the discharge pipe of the primary pump. Using the dimensions for calculating the flow area,

Pipe wall cross-sectional area

$$= \pi/4 \times \{(24.0 \times 0.0254)^2 - (23.5 \times 0.0254)^2\}$$

$$= 0.012034 \text{ m}^2$$

$$\text{WALLMC(16)} = \text{Area} \times (\rho \times C_p (304\text{-SS}))$$

$$= 0.012034 \times 4.4190 \times 10^6$$

$$= 5.31795 \times 10^4 \text{ J/m-K}$$

WALLMC(17) Same as for element 16.
 $= 5.31795D+4 \text{ J/m-K}$

WALLMC(18) Same as for element 16.
 $= 5.31795D+4 \text{ J/m-K}$

WALLMC(19) Same as for element 16.
 $= 5.31795D+4 \text{ J/m-K}$

The second primary pump group is the same as the first primary pump group.

WALLMC(20) Same as for element 14.
 $= 2.91088D+5 \text{ J/m-K}$

WALLMC(21) Same as for element 15.
 $= 3.6003D+6 \text{ J/m-K}$

WALLMC(22) Same as for element 16.
 $= 5.31795D+4 \text{ J/m-K}$

WALLMC(23) Same as for element 16.
 $= 5.31795D+4 \text{ J/m-K}$

WALLMC(24) Same as for element 16.
 $= 5.31795D+4 \text{ J/m-K}$

WALLMC(25) Same as for element 16.
 $= 5.31795D+4 \text{ J/m-K}$

WALLMC(26) This element represents a fictitious pipe connecting the hot pool and the region between the upper baffle plate and the redan. A nominal value is input which is compatible with the heat transfer calculation.
 $= 1.000 \text{ J/m-K}$

WALLMC(27) This element represents a fictitious pipe connecting the cold pool and the region between the lower baffle plate and the redan. A nominal value is input which is compatible with the heat transfer calculation.
 $= 1.000 \text{ J/m-K}$

WALLMC(28) This element represents the inlet section of the DRACS-I heat exchanger. From Fig. D-8, the shell outside diameter is 32" and its wall thickness is 0.9".

$$\begin{aligned} \text{Shell wall cross-sectional area} \\ &= \pi/4 \times [(32 \times 0.0254)^2 - (30.2 \times 0.0254)^2] \\ &= 0.05673 \text{ m}^2 \end{aligned}$$

$$\begin{aligned} \text{WALLMC(28)} &= \text{Area} \times (\rho \times C_p (304\text{-SS})) \\ &= 2.50694D+5 \text{ J/m-K} \end{aligned}$$

WALLMC(29) This element represents the shell side in the heat transfer region of the DRACS-I heat exchanger. It is assumed that this is the same as for the inlet section.
 $= 2.50694 \times 10^5 \text{ J/m-K}$

WALLMC(30) Same as element 28.
 $= 2.50694 \times 10^5 \text{ J/m-K}$

WALLMC(31) Same as element 28.
 $= 2.50694 \times 10^5 \text{ J/m-K}$

WALLMC(32) This element represents the tube section of the DRACS-II heat exchanger (which is the same as the DRACS-I heat exchanger). From Fig. D-8,
 Tube O.D. = 0.75"
 Wall thickness = 0.034"
 Total wall cross-sectional area
 $= 306 \text{ tubes} \times \pi/4 (0.75^2 - 0.682^2) \times 0.0254$
 $= 0.015098 \text{ m}^2$

$$\begin{aligned} \text{WALLMC(32)} &= \text{Area} \times (\rho \times C_p \text{ (304-SS)}) \\ &= 0.015098 \times 4.419 \times 10^6 \\ &= 6.67180 \times 10^4 \text{ J/m-K} \end{aligned}$$

WALLMC(33) Same as for element 28.
 $= 2.50694 \times 10^5 \text{ J/m-K}$

WALLMC(34) This element represents the intermediate loop pump. It is assumed to have the same characteristics as the primary loop pumps, element 15.
 $= 3.60030 \times 10^6 \text{ J/m-K}$

WALLMC(35) This element represents the horizontal pipe from the pump to the IHX. Using the dimensions for calculating the flow area,
 Pipe wall cross-sectional area
 $= \pi/4 \times \{(36 \times 0.0254)^2 - (35 \times 0.0254)^2\}$
 $= 0.035976 \text{ m}^2$

$$\begin{aligned} \text{WALLMC(35)} &= \text{Area} \times (\rho \times C_p \text{ (304-SS)}) \\ &= 0.035976 \times 4.419 \times 10^6 \\ &= 1.58978 \times 10^5 \text{ J/m-K} \end{aligned}$$

WALLMC(36) Same as for element 35.
 $= 1.58978 \times 10^5 \text{ J/m-K}$

WALLMC(37) This element represents the shell side of the IHX. Using the dimensions for calculating the inlet section, element 11,
 Shell cross-sectional area
 $= \pi/4 ((142 \times 0.0254)^2 - (138 \times 0.0254)^2)$
 $= 0.5675 \text{ m}^2$

- WALLMC(37) = Area x (rho x Cp (304-SS))
 = 0.5674 x 4.4190D+6
 = 2.50783D+6 J/m-K
- WALLMC(38) Same as for element 35.
 = 1.58978D+5 J/m-K
- WALLMC(39) Same as for element 35.
 = 1.58978D+5 J/m-K
- WALLMC(40) This element represents the shell side of the steam generator. The shell side diameter is 120" [D3], and the wall thickness is assumed to be 2".
 Shell cross-sectional area
 = $\pi/4 ((120 \times 0.0254)^{**2} - (116 \times 0.0254)^{**2})$
 = 0.47833 m**2
- WALLMC(40) = Area x (rho x Cp (304-SS))
 = 0.47833 x 4.4190D+6
 = 2.11374D+6 J/m-K
- WALLMC(41) Same as for element 35.
 = 1.58978D+5 J/m-K
- WALLMC(42) Same as for element 35.
 = 1.58978D+5 J/m-K
- WALLMC(43) This element represents the pipe connecting the pipe tee with the sodium expansion tank. Using the dimensions for calculating the flow area,
 Pipe wall cross-sectional area
 = $\pi/4 ((18 \times 0.0254)^{**2} - (17 \times 0.0254)^{**2})$
 = 0.017734 m**2
- WALLMC(43) = Area x (rho x Cp (304-SS))
 = 0.017734 x 4.4190D+6
 = 7.83699D+4 J/m-K
- WALLMC(44) This element represents the pipe from the sodium expansion tank to the air dump heat exchanger. Using the dimensions for calculating the flow area,
 Pipe wall cross-sectional area
 = $\pi/4 ((16 \times 0.0254)^{**2} - (14.833 \times 0.0254)^{**2})$
 = 0.018232 m**2
- WALLMC(44) = Area x (rho x Cp (304-SS))
 = 0.018232 x 4.4190D+6
 = 8.05690D+4 J/m-K
- WALLMC(45) This element represents the pipes in the air dump heat exchanger. Using the dimensions for calculating the flow area, ignoring the mass of the fins,
 Pipe O.D. = 1.90"
 Pipe I.D. = 1.75"

Pipe wall cross-sectional area
 $= 63 \text{ tubes} \times (\pi/4)(1.9^{**2} - 1.75^{**2}) \times 0.0254^{**2}$
 $= 0.017478 \text{ m}^{**2}$
WALLMC(45) = Area x (rho x Cp (304-SS))
 $= 0.017478 \times 4.419 \times 10^6$
 $= 7.72335 \times 10^4 \text{ J/m-K}$

WALLMC(46) Same as for element 44.
 $= 8.05690 \times 10^4 \text{ J/m-K}$

WALLMC(47) Same as for element 44.
 $= 8.05690 \times 10^4 \text{ J/m-K}$

WALLMC(48) Same as for element 32.
 $= 6.67181 \times 10^4 \text{ J/m-K}$

WALLMC(49) Same as for element 44.
 $= 8.05690 \times 10^4 \text{ J/m-K}$

WALLMC(50) Same as for element 44.
 $= 8.05690 \times 10^4 \text{ J/m-K}$

WALLMC(51) Same as for element 44.
 $= 8.05690 \times 10^4 \text{ J/m-K}$

WALLMC(52) Same as for element 45.
 $= 7.72335 \times 10^4 \text{ J/m-K}$

WALLMC(53) Same as for element 44.
 $= 8.05690 \times 10^4 \text{ J/m-K}$

WALLMC(54) Same as for element 44.
 $= 8.05690 \times 10^4 \text{ J/m-K}$

WALLMC(55) Same as for element 28.
 $= 2.50594 \times 10^5 \text{ J/m-K}$

WALLMC(56) Same as for element 44.
 $= 8.05690 \times 10^4 \text{ J/m-K}$

WALLMC(57) Same as for element 44.
 $= 8.05690 \times 10^4 \text{ J/m-K}$

#1283-1339 WALLH (IELL) The pipe wall conduction heat-transfer coefficient.

WALLH(1) The heat transfer coefficient for the core region is not used by the code. It is set to a non-zero value.
 $= 1.00$

WALLH(2) This element represents the lower section of the radial blanket subassembly. The thermal conductivity of steel is estimated at 28.32 W/m-K at nominal operating

temperatures, and the pin diameter is 0.392",

$$\text{WALLH}(2) = k/l = 28.32 / ((0.392/2)(0.0254))$$

$$= 5.68800\text{D}+3 \text{ W/m}^{**2}\text{-K}$$

- WALLH(3) Same as for element 2.
 $= 5.68800\text{D}+3 \text{ W/m}^{**2}\text{-K}$
- WALLH(4) Same as for element 2.
 $= 5.68800\text{D}+3 \text{ W/m}^{**2}\text{-K}$
- WALLH(5) This element represents the lower section of the radial shield and control assemblies. No specific geometry is available. Since the steel fraction is about the same as the pin and duct fraction for the blanket assemblies, the values are nominally set to the radial blanket values.
 $= 5.00000\text{D}+3 \text{ W/m}^{**2}\text{-K}$
- WALLH(6) Same as for element 5.
 $= 5.00000\text{D}+3 \text{ W/m}^{**2}\text{-K}$
- WALLH(7) Same as for element 5.
 $= 5.00000\text{D}+3 \text{ W/m}^{**2}\text{-K}$
- WALLH(8) This element represents the lower section of the core bypass flow region. The geometry is indeterminate for this region, so it is assumed to be the same as for the radial blanket assemblies.
 $= 5.00000\text{D}+3 \text{ W/m}^{**2}\text{-K}$
- WALLH(9) Same as for element 5.
 $= 5.00000\text{D}+3 \text{ W/m}^{**2}\text{-K}$
- WALLH(10) Same as for element 5.
 $= 5.00000\text{D}+3 \text{ W/m}^{**2}\text{-K}$
- WALLH(11) The wall heat transfer coefficient for the inlet section of the IHX. Using the dimensions for calculating the wall heat capacity,
 $= 28.32 \text{ W/m-K} / ((2"/2) \times 0.0254)$
 $= 1114.5 \text{ W/m}^{**2}\text{-K}$
- WALLH(12) The wall heat transfer coefficient for the tube wall in the IHX. Using the dimensions for calculating the wall heat capacity,
 $= 28.32 \text{ W/m-K} / (0.045" \times 0.0254)$
 $= 24776.9 \text{ W/m}^{**2}\text{-K}$
- WALLH(13) The wall heat transfer coefficient for the outlet section of the IHX. Using the dimensions for calculating the wall heat capacity,
 $= 28.32 \text{ W/m-K} / (0.75" \times 0.0254)$
 $= 1486.7 \text{ W/m}^{**2}\text{-K}$

- WALLH(14) The wall heat transfer coefficient for the inlet pipe of a primary pump. Assume the wall thickness of the inlet pipe is 1".
 $= 28.32 \text{ W/m-K} / (1.0" \times 0.0254)$
 $= 1114.5 \text{ W/m}^2\text{-K}$
- WALLH(15) The wall heat transfer coefficient for the impeller region of the primary pump is estimated by assuming that the flow area is in a "pipe" surrounded by the mass of the pump.
 Flow area = AREAEL(15) = 0.8234 m^2
 Equivalent diameter for the flow area
 $= (0.8234 / (\pi/4))^{1/2}$
 $= 1.0239 \text{ m}$
 Area associated with the pump steel
 $= \text{mass} / (\text{density} \times \text{length})$
 $= 62595.2 \text{ kg} / ((7876.3 \text{ kg/m}^3)(9.7536 \text{ m}))$
 $= 0.8148 \text{ m}^2$
 Equivalent diameter for the flow area and steel
 $= ((0.8234 + 0.8148) / (\pi/4))^{1/2}$
 $= 1.4443 \text{ m}$
 $\text{WALLH}(15) = 28.32 \text{ W/m-K} / ((1.4443 - 1.0239) / 2)$
 $= 1.34750\text{D}+02 \text{ W/m}^2\text{-K}$
- WALLH(16) The wall heat transfer coefficient for a discharge pipe from a primary pump. Using the dimensions for calculating the wall heat capacity,
 $= 28.32 \text{ W/m-K} / (0.25" \times 0.0254)$
 $= 4459.8 \text{ W/m}^2\text{-K}$
- WALLH(17) Same as for element 16.
 $= 4459.8 \text{ W/m}^2\text{-K}$
- WALLH(18) Same as for element 16.
 $= 4459.8 \text{ W/m}^2\text{-K}$
- WALLH(19) Same as for element 16.
 $= 4459.8 \text{ W/m}^2\text{-K}$
- The two primary pump groups are the same so the wall heat transfer coefficients of corresponding elements are the same.
- WALLH(20) Same as for element 14.
 $= 1114.5 \text{ W/m}^2\text{-K}$
- WALLH(21) Same as for element 15.
 $= 134.75 \text{ W/m}^2\text{-K}$
- WALLH(22) Same as for element 16.
 $= 4459.8 \text{ W/m}^2\text{-K}$
- WALLH(23) Same as for element 16.
 $= 4459.8 \text{ W/m}^2\text{-K}$

WALLH(24) Same as for element 16.
 $= 4459.8 \text{ W/m}^{**2}\text{-K}$

WALLH(25) Same as for element 16.
 $= 4459.8 \text{ W/m}^{**2}\text{-K}$

WALLH(26) The wall heat transfer coefficient for the fictitious pipe connecting the hot pool with the region between the upper baffle plate and the redan is given an arbitrary large value.
 $= 1.0000\text{D}+5 \text{ W/m}^{**2}\text{-K}$

WALLH(27) The wall heat transfer coefficient for the fictitious pipe connecting the cold pool with the region between the lower baffle plate and the redan is given an arbitrary large value.
 $= 1.0000\text{D}+5 \text{ W/m}^{**2}\text{-K}$

WALLH(28) The wall heat transfer coefficient for the inlet section of the DRACS-I heat exchanger. Using the dimensions for calculating the wall heat capacity of the shell side,
 $= 28.32 \text{ W/m-K} / (0.9" \times 0.0254)$
 $= 1238.8 \text{ W/m}^{**2}\text{-K}$

WALLH(29) Same as for element 28.
 $= 1238.8 \text{ W/m}^{**2}\text{-K}$

WALLH(30) Same as for element 28.
 $= 1238.8 \text{ W/m}^{**2}\text{-K}$

The heat exchangers in DRACS-I and DRACS-II are assumed to be the same.

WALLH(31) Same as for element 28.
 $= 1238.8 \text{ W/m}^{**2}\text{-K}$

WALLH(32) The wall heat transfer coefficient for the tube section of the DRACS-II heat exchanger. The tube wall thickness is 0.034", so that
 $= 28.32 \text{ W/m-K} / (0.034" \times 0.0254)$
 $= 3.27930\text{D}+04 \text{ W/m}^{**2}\text{-K}$

WALLH(33) Same as for element 28.
 $= 1238.8 \text{ W/m}^{**2}\text{-K}$

WALLH(34) The wall heat transfer coefficient for the intermediate loop pump is assumed to be the same as for the primary loop pump, element 15.
 $= 134.75 \text{ W/m}^{**2}\text{-K}$

WALLH(35) The wall heat transfer coefficient for the horizontal pipe in the cold return section of the intermediate loop. Using the dimensions for calculating the wall heat capacity,
 $= 28.32 \text{ W/m-K} / (0.5" \times 0.0254)$
 $= 2229.9 \text{ W/m}^{**2}\text{-K}$

- WALLH(36) Same as for element 35.
= 2229.9 W/m**2-K
- WALLH(37) The wall heat transfer coefficient for the shell side of the IHX is based on the shell thickness of 2".
= 28.32 W/m-K / (2" x 0.0254)
= 557.480 W/m**2-K
- WALLH(38) Same as for element 35.
= 2229.9 W/m**2-K
- WALLH(39) Same as for element 35.
= 2229.9 W/m**2-K
- WALLH(40) Same as for element 37.
= 557.48 W/m**2-K
- WALLH(41) Same as for element 35.
= 2229.9 W/m**2-K
- WALLH(42) Same as for element 35.
= 2229.9 W/m**2-K
- WALLH(43) Same as for element 35.
= 2229.9 W/m**2-K
- WALLH(44) The wall heat transfer coefficient for the pipe connecting the sodium expansion tank and the air dump heat exchanger is based on the dimensions for calculating the wall heat capacity.
= 28.32 W/m-K / (0.5585" x 0.0254)
= 1996.35 W/m**2-K
- WALLH(45) The wall heat transfer coefficient for the air dump heat exchanger tubes is based on the wall thickness of 0.15".
= 28.32 W/m-K / (0.15" x 0.0254)
= 7433.10 W/m**2-K
- WALLH(46) Same as for element 44.
= 1996.35 W/m**2-K
- WALLH(47) Same as for element 44.
= 1996.35 W/m**2-K
- WALLH(48) Same as for element 32.
= 3.2793D+4 W/m**2-K
- WALLH(49) Same as for element 44.
= 1996.35 W/m**2-K
- WALLH(50) Same as for element 44.
= 1996.35 W/m**2-K

The DRACS-I and DRACS-II loops are almost the same, with the exception that the secondary side is the tube side in DRACS-I and the shell side in DRACS-II.

WALLH(51) Same as for element 44.
= $1996.35 \text{ W/m}^{**2}\text{-K}$

WALLH(52) Same as for element 45.
= $7433.10 \text{ W/m}^{**2}\text{-K}$

WALLH(53) Same as for element 44.
= $1996.35 \text{ W/m}^{**2}\text{-K}$

WALLH(54) Same as for element 44.
= $1996.35 \text{ W/m}^{**2}\text{-K}$

WALLH(55) Same as for element 31.
= $1238.80 \text{ W/m}^{**2}\text{-K}$

WALLH(56) Same as for element 44.
= $1996.35 \text{ W/m}^{**2}\text{-K}$

WALLH(57) Same as for element 44.
= $1996.35 \text{ W/m}^{**2}\text{-K}$

#1423-1460 VOLLGC The total volume of each compressible volume.
VOLLGC(1) The volume of the inlet plenum. The inlet plenum consists of the core barrel and the distribution plenum. From Fig. D-3, and Ref. D3,
Core barrel: I.D. = $20'4.25'' = 244.25''$
Height for inlet plenum = $4'8''$
Inside height = $4'8'' - 6'' = 50''$
Wall thickness = $3''$
Volume of the core barrel region
= $(\pi) * ((244.25 * 0.0254)^{**2}) * (50 * 0.0254)$
= 38.391 m^{**3}

Distribution plenum (a torus shape):
O.D. of base circle = $2'4''$
Wall thickness = $2''$ (assumed)
I.D. of the base circle of the distribution plenum
= $2'4'' - 2(2'') = 2' = 24''$
Volume of the distribution plenum
= $2 * ((\pi)^{**2}) * (\text{core barrel I.D.}/2) * (\text{base circle})^{**2}$
= $2 * ((\pi)^{**2}) * ((244.25/2) * 0.0254) * ((24 * 0.0254)^{**2})$
= 22.375 m^{**3}

VOLLGC(1) = $38.391 + 22.735$
= 61.126 m^{**3}

VOLLGC(2) The volume of the hot pool, or outlet plenum. From Figs. D-2 and D-3, the following dimensions were estimated:

Height of the closure deck above the top of the core
 = 23'5.9"
 = 281.9"

I.D. of the reactor vessel = 62'

Total volume inside the reactor vessel above the top of the core
 = $(\pi/4) * ((62 * 12 * 0.0254)^2) * (281.9 * 0.0254)$
 = 2008.3 m³

O.D. of the IHX = 142"

Volume of 1 IHX

= $(\pi/4) * ((142 * 0.0254)^2) * (281.9 * 0.0254)$
 = 73.16 m³.

Volume of 4 IHXs

= 4 * 73.16
 = 292.64 m³

O.D. of a primary pump = 89"

Volume of 1 primary pump

= $(\pi/4) * ((89 * 0.0254)^2) * (23.492 * 12 * 0.0254)$
 = 28.75 m³

Volume of 4 primary pumps

= 4 * 28.75
 = 115.0 m³

DRACS O.D. = 36"

Length of the DRACS-I heat exchanger from the closure deck to the bottom of the heat exchanger
 = 20.2909'

Volume of the DRACS-I heat exchanger

= $(\pi/4) * ((36 * 0.0254)^2) * (20.2909 * 12 * 0.0254)$
 = 4.06129 m³

Volume of the DRACS-II heat exchanger

= $(\pi/4) * ((36 * 0.0254)^2) * (23.492 * 12 * 0.0254)$
 = 4.70216 m³

Length of the UIS from the closure deck to the bottom of the UIS

= 20.1016' = 6.1269 m

O.D. of the UIS = 14.53125' = 4.429 m

Volume of the UIS

= $(\pi/4) * ((4.429)^2) * (6.1269)$
 = 94.4 m³

VOLLGC(2) = 2008.3 - 292.64 - 115.0 - 4.06129
 - 4.70216 - 94.4
 = 1497.5 m³

This value is increased by 1000 m³ to prevent pressurization of the reactor vessel during a transient since the cover gas system involves pressure control.

VOLLGC(2) = 2497.5 m³

VOLLGC(3) The volume of the cold pool is estimated from Figs. D-2 and D-3 and Ref. D3. The volume of the cold pool consists of 4 regions: the cylindrical region, the toroidal region, the conical region, and the spherical cap region.

The height of each region is estimated as:

$$\text{Cylindrical region} = 16.46875' = 5.01968 \text{ m}$$

$$\text{Toroidal region} = 4.35938' = 1.32874 \text{ m}$$

$$\text{Conical region} = 10.65625' = 3.24803 \text{ m}$$

$$\text{Spherical cap region} = 1.45313' = 0.44266 \text{ m}$$

$$\text{I.D. of the reactor vessel} = 62'$$

Volume of the cylindrical region

$$= (\pi/4) * ((62 * 12 * 0.0254)^2) * 5.01968$$

$$= 1407.9 \text{ m}^3$$

The toroidal region is treated as a short cylinder.

$$\text{Average I.D. of this region} = 60.5467' = 18.45469 \text{ m}$$

Volume of the toroidal region

$$= (\pi/4) * ((18.45469)^2) * 1.32874$$

$$= 355.4 \text{ m}^3$$

$$\text{I.D. of the base conical region} = 55.703' = 16.9783 \text{ m}$$

Base area of the conical region

$$= (\pi/4) * (16.9783^2)$$

$$= 226.4013 \text{ m}^2$$

Volume of the conical region

$$= (1/3) * 226.403 * 3.24803$$

$$= 245.1 \text{ m}^3$$

$$\text{Radius of the spherical cap region} = 31' = 9.4488 \text{ m}$$

Volume of the spherical cap region

$$= (2/3) * (\pi) * (9.4488^2) * (0.44266)$$

$$= 87.8 \text{ m}^3$$

Total volume inside the reactor vessel below the lower baffle plate

$$= 1407.9 + 355.4 + 245.1 + 87.8$$

$$= 2091.2 \text{ m}^3$$

Height of the shield barrel below the lower baffle plate = 12.4' = 3.77952 m

Diameter of the shield barrel = 22'8" = 6.9088 m

Volume of the shield barrel below the lower baffle plate

$$= (\pi/4) * (6.9088^2) * (3.77952)$$

$$= 141.7 \text{ m}^3$$

Height of an IHX below the lower baffle plate

$$= 11.83636' = 3.6077 \text{ m}$$

O.D. of an IHX = 142" = 3.6068 m

Volume of 1 IHX in the cold pool

$$= (\pi/4) * (3.6063^2) * 3.6077$$

$$= 36.9 \text{ m}^3$$

Height of a primary pump in the cold pool
 $= 10.709' = 3.2641 \text{ m}$

Diameter of a primary pump $= 89" = 2.2606 \text{ m}$

Volume of 1 primary pump in the cold pool
 $= (\pi/4) * (2.2606^2) * 3.2641$
 $= 13.1 \text{ m}^3$

Length of the suction pipe of a primary pump
 $= 11.14285' = 3.39639 \text{ m}$

Diameter of the suction pipe of a primary pump
 $= 31.5" = 0.8001 \text{ m}$

Volume of 1 suction pipe
 $= (\pi/4) * (0.8001^2) * 3.39634$
 $= 1.7076 \text{ m}^3$

Length of a discharge pipe of a primary pump
 $= 23.8776' = 7.27789 \text{ m}$

Diameter of a discharge pipe of a primary pump
 $= 24" = 0.6096 \text{ m}$

Volume of 1 discharge pipe
 $= (\pi/4) * (0.6096^2) * 7.27789$
 $= 2.124 \text{ m}^3$

$$\begin{aligned} \text{VOLLGC}(3) &= 2091.2 - 141.7 - 4*36.9 - 4*13.1 - 4*1.7076 \\ &\quad - 8*2.124 - 61.126 \text{ (inlet plenum volume)} \\ &= 1664.6 \text{ m}^3 \end{aligned}$$

This value is increased by 1000 m^3 to prevent
 pressurization of the reactor vessel during a transient
 since the cover gas system involves pressure control.
 $\text{VOLLGC}(3) = 2664.6 \text{ m}^3$

$\text{VOLLGC}(4)$ The pump outlet plenum volume for pump group 1 is
 assumed to be 6.0 m^3 ,
 $= 6.0 \text{ m}^3$

$\text{VOLLGC}(5)$ The pump outlet plenum volume for pump group 2 is
 assumed to be 6.0 m^3 ,
 $= 6.0 \text{ m}^3$

$\text{VOLLGC}(6)$ The volume of the region between the upper baffle plate
 and the redan. From Fig. D-2,
 Height of the upper thermal baffle region
 $= 30" = 0.762 \text{ m}$
 Diameter of the baffle region $= 62' = 18.8976 \text{ m}$
 Total volume within the reactor vessel of the upper
 thermal baffle region
 $= (\pi/4) * (18.8976^2) * 0.762$
 $= 213.7 \text{ m}^3$

The volume of 1 IHX in this region
 $= (\pi/4) * ((142*0.0254)^2) * 0.762$
 $= 7.9 \text{ m}^3$

The volume of 1 pump in this region
 $= (\pi/4) * ((89 * 0.0254)^2) * 0.762$
 $= 3.1 \text{ m}^3$

The volume of the DRACS2 heat exchanger in this region
 $= (\pi/4) * ((36 * 0.0254)^2) * 0.762$
 $= 0.5 \text{ m}^3$

The volume of the shield barrel in this region
 $= (\pi/4) * ((272 * 0.0254)^2) * 0.762$
 $= 28.7 \text{ m}^3$

$$\text{VOLLGC}(6) = 213.7 - 4 * 7.9 - 4 * 3.1 - 0.5 + 28.7$$

$$= 140.5 \text{ m}^3$$

VOLLGC(7) The volume of the region between the lower baffle plate and the redan is the same as for volume 6,
 $= 140.5 \text{ m}^3$

VOLLGC(8) The volume of the pump bowl associated with the intermediate loop pump.
 $\text{XLENEL}(34) = 1.0 \text{ m}$
 Diameter of the intermediate pump = 85" = 2.759 m
 $\text{VOLLGC}(8) = (\pi/4) * (2.759^2) * 1.0$
 $= 3.67 \text{ m}^3$

VOLLGC(9) The volume of the pipe tee is assumed to 1.0 m³,
 $= 1.0 \text{ m}^3$

VOLLGC(10) The volume of the sodium expansion tank is estimated from Fig. D-4 and Ref. D3.
 Length of the sodium expansion tank = 120.0"
 Outside diameter of the sodium expansion tank = 164"
 Treating the sodium expansion tank as a cylinder,
 $\text{VOLLGC}(10) = (\pi/4) * ((120 * 0.0254)^2) * (164 * 0.0254)$
 $= 30.395 \text{ m}^3$

VOLLGC(11) The volume of the sodium storage tank associated with the DRACS-I loop is assumed to be 20.0 m³,
 $= 20.0 \text{ m}^3$

VOLLGC(12) Same as for volume 11,
 $= 20.0 \text{ m}^3$

#1461-1498 PRES GO The initial gas pressure in the compressible volumes which contain cover gas. Atmospheric pressure is assumed for the reactor vessel and the DRACS tanks.
 $\text{PRES GO}(2) = 1.01325\text{D}+05 \text{ Pa}$
 $\text{PRES GO}(3) = 1.01325\text{D}+05 \text{ Pa}$
 $\text{PRES GO}(11) = 1.01325\text{D}+05 \text{ Pa}$
 $\text{PRES GO}(12) = 1.01325\text{D}+05 \text{ Pa}$

The pressure in the pump bowl at nominal conditions is based on an estimated static head of 105 ft. This results in a static head of $2.73646 \times 10^5 \text{ Pa}$.
 $\text{PRES GO}(8) = 2.73646\text{D}+05 \text{ Pa}$
 $\text{PRES GO}(10) = 2.72646\text{D}+05 \text{ Pa}$

- #1499-1536 ALPHAP The volume pressure expansion coefficient is set to 2.0×10^{-8} for all compressible volumes based on previous work.
 $= 2.0D-08$ (1/Pa)
- #1537-1574 ALPHAT The volume thermal expansion coefficient is based on the thermal expansion of steel. A typical value is 6.0×10^{-5} , and this is used for all compressible volumes.
 $= 6.0D-05$ (1/T)
- #1575-1612 ZCVL The reference height for the printout of liquid pressure in each compressible volume.
 Reference height for the inlet plenum
 $ZCVL(1) = ZINL(1) = -1.7018$ m for the oxide core
 $ZCVL(1) = ZINL(1) = -1.4986$ m for the metallic core
- Reference height for the hot pool above the core
 $ZCVL(2) = ZINL(2) = 2.9972$ m for the oxide core
 $ZCVL(2) = ZINL(2) = 3.2004$ m for the metallic core
- Reference height at the inlet of a primary pump
 $ZCVL(3) = ZINL(3) = -3.0480$ m for the oxide core
 $ZCVL(3) = ZINL(3) = -2.8448$ m for the metallic core
- Reference height at the outlet of a primary pump
 $ZCVL(4) = ZINL(4) = 0.1016$ m for the oxide core
 $ZCVL(4) = ZINL(4) = 0.3048$ m for the metallic core
- $ZCVL(5) = 0.1016$ m for the oxide core
 $ZCVL(5) = 0.3048$ m for the metallic core
- Mid-height of the region between the upper thermal baffle plate and the redan. From Fig. D-2,
 Elevation of the upper region mid-height
 $= 64'4'' - 1'3'' = 757''$
 $ZCVL(6) = 757'' - 654'' = 103''$
 $= 2.6162$ m for the oxide core
 $ZCVL(6) = 2.6162 + 0.2032$
 $= 2.8194$ m for the metallic core
- Mid-height of the region between the lower thermal baffle plate and the redan. From Fig. D-2,
 Elevation of the lower region mid-height
 $= 64'4'' - 3'9'' = 60'7'' = 727''$
 $ZCVL(7) = 727'' - 654'' = 73''$
 $= 1.8542$ m for the oxide core
 $ZCVL(7) = 1.8542 + 0.2032$
 $= 2.0574$ m for the metallic core
- The outlet elevation of the intermediate loop pump
 $ZCVL(8) = ZOUTEL(42) = 19.05$ m for the oxide core
 $ZCVL(8) = ZOUTEL(42) = 19.2532$ m for the metallic core

The elevation of the pipe tee

ZCVL(9) = ZINL(17) = 19.05 m for the oxide core

ZCVL(9) = ZINL(17) = 19.2532 m for the metallic core

The elevation of intermediate loop sodium expansion tank

ZCVL(10) = ZOUTEL(43) = 23.876 m for the oxide core

ZCVL(10) = ZOUTEL(43) = 24.0792 m for the metallic core

The elevation of the DRACS-I sodium storage tank

ZCVL(11) = ZOUTEL(50) = 33.0708 m for the oxide core

ZCVL(12) = ZOUTEL(50) = 33.2740 m for the metallic core

The elevation of the DRACS-II sodium storage tank

ZCVL(12) = ZOUTEL(12) = 33.0708 m for the oxide core

ZCVL(12) = ZOUTEL(12) = 33.2740 m for the metallic core

#1613-1650 AREAIN

The area of the liquid-gas interface in compressible volumes which have cover gas. For those volumes without cover gas, AREAIN is set to 1.0.

AREAIN(1) AREAIN(1) = 1.0

AREAIN(2) Interface area of the hot pool

Cross-sectional area of the hot pool

= $(\pi/4) * (62 * 12 * 0.0254)^2$

= 280.5 m²

The cross-sectional areas of all of the components penetrating the hot pool surface

Cross-sectional of 1 IHX

= $(\pi/4) * (142.) * 9.9254^2$

= 10.2 m²

Cross-sectional area of 1 primary pump

= $(\pi/4) * (89.0 * 0.0254)^2$

= 4.0 m²

Cross-sectional area of the UIS

= $(\pi/4) * (16.65 * 12 * 0.0254)^2$

= 20.0 m²

Cross-sectional area of the DRACS-I heat exchanger

= $(\pi/4) * (36 * 0.0254)^2$

= 0.7 m²

Cross-sectional area of the DRACS-II heat exchanger

= $(\pi/4) * (36 * 0.0254)^2 = 0.7 \text{ m}^2$,

AREAIN(2) = 280.5 - 4*10.2 - 4*4.0 - 20.2 - 2*0.7
= 202.3 m²

AREAIN(3) The interface area in the cold pool around the primary pump and the DRACS-II heat exchanger. As calculated for AREAIN(2), the cross-sectional areas of a primary pump and the heat exchanger are 4.0 m² and 0.7 m², respectively.

AREAIN(3) = 4*4.0 + 0.7
= 16.7 m²

AREAIN(4) AREAIN(4) = 1.0

AREAIN(5) AREAIN(5) = 1.0

AREAIN(6) AREAIN(6) = 1.0

AREAIN(7) AREAIN(7) = 1.0

AREAIN(8) The interface area of the pump bowl is assumed to 0.5,
AREAIN(8) = 0.5 m^2

AREAIN(9) AREAIN(9) = 1.0

AREAIN(10) The interface area in the sodium expansion tank assuming
that the tank is half full. From Ref. D3, the length of
the tank is 164" and the diameter is 120".
AREAIN(10) = $(120 \times 0.0254) \times (164 \times 0.0254)$
= 12.697 m^2

AREAIN(11) The interface area in the sodium storage tank in the
DRACS-I loop is assumed to be 4.0 m^2 ,
AREAIN(11) = 4.0 m^2

AREAIN(12) The interface area in the sodium storage tank in the
DRACS-II loop is assumed to be 4.0 m^2 ,
AREAIN(12) = 4.0 m^2

#1651-1688	TREFCV	The steady-state temperature of the cover gas in the compressible volumes is set based on the liquid temperature. For volumes with shared cover gas, the temperature is set to the average of the liquid temperatures weighted by the interfacial area. TREFCV(2) = 758.3 K TREFCV(3) = 758.3 K
#1689	GAMGSC	The ratio of the specific heats, C_p/C_v , for the cover gas assumed to be helium = 1.6
#1690	RGASC	The gas constant for the cover gas is set to the recommended value for helium, = $2078.0 \text{ Pa-m}^3/\text{Kg-K}$
#1691	UOCVGS	The viscosity of the cover gas at the reference temperature, TRFU, is set to the recommended value for helium, = $1.94 \times 10^{-5} \text{ Pa-s}$
#1692	TRFU	The reference temperature for the cover gas viscosity = 273 K
#1693-1732	XLENGG	The length of the gas segments The length of the gas pipe connecting the pump and the sodium expansion tank. From Fig. D-2, XLENGG(1) = 6' = 1.828 m.

- #1721-1748 AREASG The flow area of the gas segments
For the pipe connecting the pump and the sodium expansion tank, from Fig. D-2, and assuming a wall thickness of 0.0625",

$$\text{AREASG}(1) = (\pi/4) * ((2 - 2 * 0.0625) * 0.0254)^2$$

$$= 1.780-3 \text{ m}^2$$
- #1749-1804 CVGMLT (M,ISGG) The multiplicity factors at the gas segment ends
There is one gas segment connecting the pump and the sodium expansion tank,

$$\text{CVGMLT}(1,1) = 1.0$$

$$\text{CVGMLT}(2,1) = 1.0$$
- #1805-1832 DHSEGG The hydraulic diameter of the gas segments
Using the dimensions for calculating the flow area,

$$\text{DHSEGG}(1) = (2 - 2 * 0.0625) * 0.0254$$

$$= 4.76250-02 \text{ m}$$
- #1833-1860 ROUGHG The surface roughness of the gas segments

$$\text{ROUGHG}(1) = 1.00-05 \text{ (for a smooth pipe).}$$
- #1861-1898 TAUGAS The heat transfer time constant for the cover gas. For compressible volumes with cover gas, the time constant is set to 1.0×10^8 sec, which essentially maintains the cover gas temperature constant. This is intended to simulate the continued operation of the cover gas recirculating system, presuming that some kind of temperature control is available in this system.

$$\text{TAUGAS}(2) = 1.00+08 \text{ s}$$

$$\text{TAUGAS}(3) = 1.00+08 \text{ s}$$

$$\text{TAUGAS}(8) = 1.00+08 \text{ s}$$

$$\text{TAUGAS}(10) = 1.00+08 \text{ s}$$

$$\text{TAUGAS}(11) = 1.00+08 \text{ s}$$

$$\text{TAUGAS}(12) = 1.00+08 \text{ s}$$
- #1899-1910 PMPINR The moment of inertia of the primary and intermediate loop pumps were determined from test calculations which resulted in an initial flow-halving time of 8 seconds for the primary pumps and 6 seconds for the intermediate loop pumps.

$$\text{PMPINR}(1) = 18650.0 \text{ Kg-m}^2$$

$$\text{PMPINR}(2) = 18650.0 \text{ Kg-m}^2$$

$$\text{PMPINR}(3) = 400.0 \text{ Kg-m}^2$$
- #1911-1922 HEADR The rated pump head is determined from information contained in Ref. D3.
For a primary pump
Pump suction head = 20 psig
Pump discharge head = 130 psig
Rated pump head = $130 - 20$

$$= 110 \text{ psi}$$

$$= 7.584230+06 \text{ Pa}$$

$$\text{HEADR}(1) = 7.584230+05 \text{ Pa}$$

$$\text{HEADR}(2) = 7.584230+05 \text{ Pa}$$

For an intermediate pump

Total head at design flow = 350 ft

Minimum NPSH required at pump inlet for 110% flow
= 236 ft

Estimated difference at 100% flow = 94.2 ft

$$\begin{aligned} \text{HEADR}(3) &= (0.3048 \text{ ft/m}) * (94.2 \text{ ft}) * (9.807 \text{ m/s}^2) \\ &\quad * (860.63 \text{ Kg/m}^3) \\ &= 2.42338 \times 10^5 \text{ Pa} \end{aligned}$$

#1923-1934 PMPSPR

The rated pump speed is taken from Ref. D3.

For a primary pump

Pump design speed = 495 rpm = 51.8 rad/s

PMPSPR(1) = 51.8 rad/s

PMPSPR(2) = 51.8 rad/s

For an intermediate pump

Pump speed = 1115 rpm = 116.76 rad/s

PMPSPR(3) = 116.76 rad/s

#1935-1946 PMPFLR

The rated pump flow is taken from Ref. D3.

PMPFLR(1) = FLOSSL(6) = 4492.3 Kg/s

PMPFLR(2) = FLOSSL(9) = 4492.3 Kg/s

PMPFLR(3) = FLOSSL(34) = 4460.4 Kg/s

#1947-1958 PMPEFR

The rated pump efficiency is taken from Ref. D3 for the primary pumps and assumed for the intermediate loop pump.

Primary pump motor torque = 86000 ft-lb
= $1.1653 \times 10^5 \text{ J}$

Power input = $(1.1653 \times 10^5 \text{ J}) * (51.8 \text{ rad/s})$
= $6.04 \times 10^6 \text{ W}$

Work done = $(4492.3 \text{ kg/s}) * (7.9738 \times 10^5 \text{ Pa}) / 860.5$
= $4.1628 \times 10^6 \text{ W}$

Efficiency = $4.1628 \times 10^6 / 6.04 \times 10^6$
= 68.9%

PMPEFR(1) = 68.9%

PMPEFR(2) = 68.9%

PMPEFR(3) = 83.0%

#1983-2223 APMPHD

The pump head coefficients are based on using the homologous pump model in PRIMAR-4 with coefficients derived by comparison with the pump behavior for a prototype Clinch River Breeder Reactor Pump. The same data is used for all three pumps in the model.

APMPHD(1) = 0.01

APMPHD(2) = 0.00

APMPHD(3) = -73.13

APMPHD(4) = 0.00268

APMPHD(5) = 0.00

APMPHD(6) = 0.07

APMOHD(7) = 0.00383

APMPHD(8) = 0.01071

APMPHD(9) = 0.01406

APMPHD(10) = 0.01

APMPHD(11) = 0.268
 APMPHD(12) = 0.0
 APMPHD(13) = 0.0
 APMPHD(18) = 1.0
 APMPHD(19) = 0.03
 APMPHD(20) = 0.03

#2223-2462 AMOTTK The normalized motor torque verses time tables. These are set according to the accident being investigated. For cases with a loss of pumping power, the normalized motor torque is set to 0.0 at the start of the transient. For cases where the pumps continue to run the value is set at 1.0.

For the steady-state initialization,

Time	Normalized Torque
0.0	1.0
10000.0	1.0

#2463 GRAVTY The acceleration due to gravity
= 9.807 m/s**2

#2464-2501 BTAPNA The sodium isothermal compressibility for the compressible volumes, 1/Pa, at the reference temperature, TREFCV. These are all set to 2.13D-10 1/Pa as suggested by the input description.

#2578-2615 HWALL The wall heat-transfer coefficient for the compressible volumes.

Wall heat-transfer coefficient for the inlet plenum

Assuming a wall thickness of 3", with a thermal conductivity of 21.5 W/m-K for 304-SS,

$$\begin{aligned}
 \text{HWALL}(1) &= 21.5 / (3 * 0.0254) \\
 &= 282.2 \text{ W/m**2-K}
 \end{aligned}$$

The wall heat-transfer coefficient for the hot pool

Using the same assumptions as for volume 1,

$$\begin{aligned}
 \text{HWALL}(2) &= 21.5 / (3 * 0.0254) \\
 &= 282.2 \text{ W/m**2-K}
 \end{aligned}$$

The wall heat-transfer coefficient of the cold pool

Using the same assumptions as for volume 1,

$$\begin{aligned}
 \text{HWALL}(3) &= 21.5 / (3 * 0.0254) \\
 &= 282.2 \text{ W/m**2-K}
 \end{aligned}$$

The wall heat-transfer coefficient of a primary pump outlet plenum wall, from Fig. D-7,

Pump wall thickness = 1.96" = 0.049866 m

$$\begin{aligned}
 \text{HWALL}(4) &= 21.5 / 0.049866 \\
 &= 431.2 \text{ W/m**2-K}
 \end{aligned}$$

$$\text{HWALL}(5) = 431.2 \text{ W/m**2-K}$$

The wall heat-transfer coefficient of the thermal baffle regions. Assume the wall thickness is 2" and of

304-SS material,
 $HWALL(6) = 21.5 / (2 * 0.0254)$
 $= 423.2 \text{ W/m}^{**2}\text{-K}$

$HWALL(7) = 423.2 \text{ W/m}^{**2}\text{-K}$

The wall heat-transfer coefficient of the intermediate pump is assumed to be same as for the primary pumps.
 $HWALL(8) = 431.2 \text{ W/m}^{**2}\text{-K}$

The wall heat-transfer of the pipe tee is set to a value of $50.0 \text{ W/m}^{**2}\text{-K}$.
 $HWALL(9) = 50.0 \text{ W/m}^{**2}\text{-K}$.

The wall heat-transfer coefficient of the sodium expansion tank is based on a wall thickness of 2". The tank is of 9Cr-1Mo material of which the thermal conductivity is $k = 28.32 \text{ W/m-K}$.
 $HWALL(10) = 28.32 / (2 * 0.0254)$
 $= 557.5 \text{ W/m}^{**2}\text{/K}$

The wall heat-transfer coefficient for the sodium storage tanks in the DRACS loops is based on a wall thickness of 0.075" and the material is 304-SS.
 $HWALL(11) = 21.5 / (0.075 * 0.0254)$
 $= 1128.6 \text{ W/m}^{**2}\text{-K}$

$HWALL(12) = 1128.6 \text{ W/m}^{**2}\text{-K}$

#2616-2653 AWALL

The wall surface area for each compressible volume.
 Surface area of the inlet plenum.

Diameter of the core barrel $= 244.25"$

Diameter of the toroidal region $= 24"$

Surface area of the core barrel bottom

$= (\pi/4) * ((244.25 * 0.0254) ** 2)$

$= 30.229 \text{ m}^{**2}$

Surface area of the toroidal region

$= 4 * (\pi ** 2) * ((244.25/2) * 0.0254) * ((24.0/2) * 0.0254)$

$= 37.33 \text{ m}^{**2}$

$AWALL(1) = 30.229 + 37.33$

$= 67.56 \text{ m}^{**2}$

Surface area of the walls surrounding the hot pool

Height of sodium surface above core top

$= 85' - 64'4" = 248" = 6.2992 \text{ m}$

I.D. of the hot pool $= 62' = 744" = 18.8976 \text{ m}$

$AWALL(2) = (\pi) * (18.8976) * (6.2992)$

$= 374.0 \text{ m}$

Surface area of the walls surrounding the cold pool.

The surface area of the cold pool consists of 4

regions: the cylindrical region, the toroidal region, the conical region and the spherical cap region.

Using the same dimensions for calculating the volume,

Average ID of the toroidal section

$$= 60.5467' = 18.45469 \text{ m}$$

Base radius of the conical region = $56.1875'/2$

$$= 8.563 \text{ m}$$

Radius of the top toroidal region = $20.58594'/2$

$$= 3.137 \text{ m}$$

I.D. of the reactor vessel = $62' = 18.8976 \text{ m}$

Surface area of the cylindrical region

$$= (\pi) * (18.8976) * (5.01968)$$

$$= 298.0 \text{ m}^2$$

Surface area of the toroidal region

$$= (\pi) * (18.45469) * (1.32874)$$

$$= 77.02 \text{ m}^2$$

Surface area of the conical section

$$= (\pi) * (8.563 + 3.137) * (((3.24803)^2) + (3.24803 - 3.137)^2)^{1/2}$$

$$= 232.4 \text{ m}^2$$

Surface area of the spherical cap region

$$= (\pi) * (0.51678) * (2 * 3.137)$$

$$= 10.186 \text{ m}^2$$

$$AWALL(3) = 298.0 + 77.02 + 232.4 + 10.186$$

$$= 617.6 \text{ m}^2$$

Surface area of a primary pump outlet plenum.

O.D. of a primary pump = 89"

Assume the plenum wall is 2" thick

I.D. of a primary pump = $89" - 4" = 85" = 2.159 \text{ m}$

XLENEL(15) = 3.4036 m

$$AWALL(4) = (\pi) * (2.159) * (3.4036)$$

$$= 23.086 \text{ m}^2$$

$$AWALL(5) = 23.086 \text{ m}^2$$

Surface area of the thermal baffle regions.

With the dimensions used for calculating the volume,

I.D. of the reactor vessel = 62'

O.D. of an IHX = 142"

O.D. of a primary pump = 89"

O.D. of DRACS-I = 36"

O.D. of DRACS-II = 36"

O.D. of the core barrel = 20.975'

Total horizontal area of this region

$$= (\pi/4) * ((62 * 12 * 0.0254)^2)$$

$$= 280.48 \text{ m}^2$$

Cross-sectional area of 1 IHX

$$= (\pi) * ((142 * 0.0254)^2)$$

$$= 10.22 \text{ m}^2$$

Cross-sectional area of 1 primary pump

$$= (\pi/4) * ((89 * 0.0254)^2)$$

$$= 4.01 \text{ m}^2$$

Cross-sectional area of the DRACS-I heat exchanger

$$= (\pi/4) * ((36 * 0.0254)^2)$$

$$= 0.66 \text{ m}^2$$

Cross-sectional area of the DRACS-II heat exchanger
 $= (\pi/4) * ((36 * 0.0254) ** 2)$
 $= 0.66 \text{ m}^2$

Cross-sectional area of core
 $= (\pi/4) * (((20 * 12 + 9.75) * 0.0254) ** 2)$
 $= 31.61 \text{ m}^2$

Net horizontal area in this region
 $= 280.48 - (4 * (10.22 + 4.401) + 2 * 0.66 + 31.61)$
 $= 190.63 \text{ m}^2$

Height of this region = 2.5' = 0.762 m

I.D. of the reactor vessel = 62' = 18.8976 m

O.D. of the core barrel = 20'9.75" = 6.34363 m

$= (\pi) * (18.8976 + 6.34363) * (0.762)$
 $= 60.43 \text{ m}^2$

$AWALL(6) = 190.63 + 60.43$
 $= 251.1 \text{ m}^2$

$AWALL(7) = 251.1 \text{ m}^2$

Surface area of the pump bowl of the intermediate pump.

I.D. of the intermediate pump wall = 85"

$XLENEL(34) = 1.0 \text{ m}$

$AWALL(8) = (\pi) * (85 * 0.0254) * (1.0)$
 $= 6.7827 \text{ m}^2$

Surface area of the pipe tee is set to 1.0 m².

$AWALL(9) = 1.0 \text{ m}^2$

Surface area of the sodium expansion tank.

O.D. of the tank = 120"

Thickness of the tank = 2"

Length of the tank = 164"

Assume the tank is 50% filled.

$AWALL(10) = 0.5 * (\pi) * (120 * 0.0254) * (164 * 0.0254)$
 $+ (\pi) * (((120/2) * 0.0254) ** 2)$
 $= 27.24 \text{ m}^2$

Surface area of the DRACS sodium storage tanks is set to 1.0 m².

$AWALL(11) = 1.0 \text{ m}^2$

$AWALL(12) = 1.0 \text{ m}^2$

#2654-2691 CMWALL

The mass times the specific heat of the wall of each compressible volume.

Inlet plenum.

$AWALL(1) = 67.56 \text{ m}^2$

$Rho * Cp = 4.419D+06 \text{ J/m}^3\text{-K}$

Thickness = 3"

$CMWALL(1) = (67.56 * 3 * 0.0254) * (4.419D+06)$
 $= 2.27493D+07 \text{ J/K}$

Outlet plenum

Thickness = 3"

AWALL(2) = 374.0 m**2

Rho*Cp = 4.419D+06 J/m**3/K

$$\text{CMWALL}(2) = (374.0 * 3 * 0.0254) * (4.419D+06)$$

$$= 1.2594D+08 \text{ J/K}$$

Cold pool

The cold pool consists of 4 regions: the cylindrical region, the toroidal region, the conical region and the spherical cap region. Assume a wall thickness of 3".

Surface area of the cylindrical region = 298.0

Surface area of the toroidal region = 77.02

Surface area of the conical region = 232.4

Surface area of the spherical cap region = 30.68

Steel volume of the cylindrical region

= 298.0*2.75*0.0254

= 20.82 m**3

Steel volume of the toroidal region

= 77.02*5*0.0254

= 9.78 m**3

Steel volume of the conical region

= 232.4*3.5*0.0254

= 20.66 m**3

Steel volume of spherical cap region

= 30.68*3.5*0.0254

= 2.73 m**3

Volume = 20.82 + 9.78 + 20.66 + 2.73

= 53.99 m**3

Rho*Cp = 4.419D+06 J/m**3-K

CMWALL(3) = 53.99 * 4.419D+06

= 2.3858D+08 J/K

Outlet plenum of a primary pump

Plenum wall thickness = 2"

AWALL(4) = 23.086 m

Rho*Cp = 4.419D+06 J/m**3-K

CMWALL(4) = 23.086*2*0.0254*4.419D+06

= 5.18247D+06 J/K

CMWALL(5) = 5.18247D+06 J/K

Baffle regions

Wall thickness = 2"

AWALL(6) = 251.1 m**2

Rho *Cp = 4.419D+06 J/m**3-K

CMWALL(6) = 2*0.0254*251.1*4.419D+06

= 5.63681D+07 J/K

CMWALL(7) = 5.63681D+07 J/K

Intermediate loop pump bowl

I.D. of the intermediate pump is 85"

```
XLENEL(34) = 1.0 m
Surface area = (pi)*(85*0.0254)*(1.0) = 6.7827 m**2
Wall thickness = 2"
Volume of this region = (2*0.0254)*(6.7827)
                        = 0.34456 m**3
```

Rho *Cp = 4.419D+06 J/m**3/K
CMWALL(8) = 0.34456*4.419D+06
= 1.52262D+06 J/K

```
Pipe tee
AWALL(9) = 1.0 m
Wall thickness is 0.25"
Rho *Cp = 4.419D+06 J/m**3-K
CMWALL(9) = (0.25*0.0254*1.0)*4.419D+06
           = 2.80606D+04 J/K
```

Sodium expansion tank
 Wall thickness of the tank = 2"
 $AWALL(10) = 27.24 \text{ m}^{**2}$
 $Rho * Cp = 4.58D+06 \text{ J/m}^{**3}\text{-K}$ for 9Cr-1Mo steel
 $CMWALL(10) = 27.24 * 2 * 0.0254 * 4.58D+06$
 $= 6.33777D+06 \text{ J/K}$

```

DRACS loops sodium storage tanks
Wall thickness = 0.25"
CMWALL(11) = 2.80606D+04 J/K
CMWALL(12) = 2.80606D+04 J/K

```

#2694	C1BY	Coefficients in Lyon-Martinelli convective heat-transfer correlation for bypass channel. =0.025
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#2695	C2BY	=0.8
#2696	C3BY	=4.8

#2697-2704	XKALBY	Thermal conductivity of bypass wall A. This is set according to the material in the bypass wall. For the oxide core, a thermal conductivity of 4.7 W/m-K is used for UO2, and 26.2 for steel. In the metallic core, 26.2 is used for all channels.
#2705-2712	XKAUBY	

Bypass	channel	XKALBY	XKAUBY
1	(oxide)	4.7	4.7
1	(metal)	26.2	26.2
2		26.2	26.2
3		26.2	26.2

#2713-2720	XKBLBY	Thermal conductivity of bypass walls B and D. This is to the value of steel for all cases, 26.2 W/m-K.
#2721-2728	XKBUBY	
#2729-2736	XKDBY	

#2737-2744	DABY	Thickness of bypass wall A in the bypass channel. The thickness is calculated to preserve the thickness and mass of the contents of the fuel pins in the radial blankets, in the pins in the shields, and to an arbitrary value in the core bypass flow region.
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Cladding cross-sectional area = $1.6498 \times 10^{-5} \text{ m}^2$

Cladding thickness = $0.22" = 5.588 \times 10^{-4} \text{ m}$

Cladding perimeter = $2.9524 \times 10^{-2} \text{ m}$

Fuel cross-sectional area = $(\pi)(0.348"/2)^2$
 $= 6.13643 \times 10^{-5} \text{ m}^2$

Fuel thickness = $6.13643 \times 10^{-5} / 2.9524 \times 10^{-2}$
 $= 2.07845 \times 10^{-3} \text{ m}$

DABY(1) = $2.07845 \times 10^{-3} \text{ m}$

Shield hydraulic diameter = $2.5 \times 10^{-3} \text{ m}$

Shield flow area = $2.98305 \times 10^{-3} \text{ m}^2$

Shield perimeter = 4.773 m

Steel area per shield = $1.32292 \times 10^{-2} \text{ m}^2$

Steel thickness = $2.77167 \times 10^{-3} \text{ m}$

Divide this between bypass walls A and B equally,

DABY(2) = $1.38584 \times 10^{-3} \text{ m}$

DABY(3) = $5.00000 \times 10^{-4} \text{ m}$

#2745-2752 DBBY The thickness of bypass wall B in the bypass channels was determined with DABY.

DBBY(1) = $5.58800 \times 10^{-4} \text{ m}$

DBBY(2) = $1.38584 \times 10^{-3} \text{ m}$

DBBY(3) = $5.00000 \times 10^{-4} \text{ m}$

#2753-2760 DDBY The thickness of bypass wall D in the bypass channels is the hexcan thickness for bypass channels 1 and 2, and is arbitrary for channel 3.

DDBY(1) = $3.81000 \times 10^{-3} \text{ m}$

DDBY(2) = $3.81000 \times 10^{-3} \text{ m}$

DDBY(3) = $5.00000 \times 10^{-4} \text{ m}$

#2761-2768 RCALBY The density times specific heat of bypass walls A, B, and D is given by the materials; for UO₂, $2.9 \times 10^6 \text{ J/m}^3\text{-K}$, and for steel, $4.709 \times 10^6 \text{ J/m}^3\text{-K}$.
 #2769-2776 RCAUBY
 #2777-2784 RCBLBY
 #2785-2792 RCBUBY
 #2793-2800 RCDBY
 Metallic fuel is assumed to be the same as steel.

	1	2	3
RCALBY (oxide)	2.9×10^6	4.709×10^6	4.709×10^6
RCALBY (metal)	4.709×10^6	4.709×10^6	4.709×10^6
RCAUBY (oxide)	2.9×10^6	4.709×10^6	4.709×10^6
RCAUBY (metal)	4.709×10^6	4.709×10^6	4.709×10^6
RCBLBY	4.709×10^6	4.709×10^6	4.709×10^6
RCBUBY	4.709×10^6	4.709×10^6	4.709×10^6
RCDBY	4.709×10^6	4.709×10^6	4.709×10^6

#2801-2856 PSHPBY The power shape by node for each bypass channel is assumed to be uniform. Values are input for bypass channel 1 only, since power is generated only in this channel.
 PSHPBY = 1.0

#2857-2880 GAMNBY The fraction of total reactor power by region for each bypass channel. Bypass channel 1 is set to 1-FRPR, block 12. Since there is a multiplicity of 90 in

bypass channel 1, the input value is
 = (1-FRPR)/90
 = 3.51967D-04 for region A only

#2921-2928 PERABY

The perimeter of walls B and D in the bypass channels.

#2929-2936 PERDBY

The pin perimeters were determined as part of DABY.

Duct wall perimeter = 0.49546 m

Bypass channel	PERABY	PERDBY
1	4.98955	0.49546
2	1632.37	169.45
3	1.00	0.10

#2932-3104 DTMPTB

The table of normalized temperature drop for an IHX or a steam generator.

#2937-2938 Set to 1.0 for normal temperature drop in the steam generator. Set to 0.0 for the loss-of-heat-sink at the steam generator.

#3105-3272 ZCENTR

The height of the thermal center for the steam generator.

ZCENTR(SG) = ZOUTEL(40) + XLENEL(40)/2
 = -6.9088 + 22.6060/2
 = 4.3942 m for the oxide core
 ZCENTR(SG) = 4.3942 + 0.2032
 = 4.5974 m for the metallic core

#3105-3106 Set to 4.3942 m for oxide core.

#3105-3106 Set to 4.5974 m for metallic core.

#3273-3440 TMPMTB

Time for DTMPTB and ZCENTR tables.

For steady-state conditions:

#3273-3274 Set to 0.0 and 10000.0 s.

#3287-3288 Set to 0.0 and 10000.0 s.

For transient conditions, refer to Appendices A and B.

#3441-3444 DTEVPF

Fraction of total steam generator sodium temperature drop. This is set to 0.9999 for the table look-up option as suggested in the input description.

#3445-3448 PRSIHX

Steady-state intermediate side IHX inlet pressure.

This value was adjusted using SASSYS calculatoins to give the proper interface elevations in the intermediate loops, including the DRACS loops. This is used in conjunction with PRES0.

PRSIHX(1) = 7.0300D+05

PRSIHX(2) = 3.5000D+05

PRSIHX(3) = 3.5000D+05

#3455-3515 DZSPHX

The fractional axial node heights in the IHX. The usual number is 40 full size nodes, but for numerical reasons, a half-size node is placed at each end of each IHX.

DZSPHX(1) = 0.5

DZSPHX(2) = 1.0

. . . .

$$\begin{aligned} \text{DZSHPX}(40) &= 1.0 \\ \text{DZSHPX}(41) &= 0.5 \end{aligned}$$

- #3516-3519 PERSPX The perimeter between the shell and the shell side coolant in IHX.
 IHX
 Diameter of the outer shell = 3.28246 m
 Diameter of the center pipe = 1.40677 m
 $\text{PERSPX}(1) = (\pi)(3.28246 + 1.40677)$
 $= 14.73165 \text{ m}$
- DRACS-I heat exchanger
 I.D. of the shell = $32" - 2 \times 0.9" = 30.2" = 0.76709 \text{ m}$
 O.D. of the center tube = $14.883" + 2 \times 0.25" = 15.383"$
 $= 0.39073 \text{ m}$
 $\text{PERSPX}(2) = (\pi)(0.76709 + 0.39073)$
 $= 3.63736 \text{ m}$
- DRACS-II heat exchanger
 $\text{PERSPX}(3) = \text{PERSPX}(2)$
 $= 3.63736 \text{ m}$
- #3524-3527 PERPTX The perimeter between the shell side coolant and the tubes in the IHX.
 IHX
 O.D. of the tubes = 1.125"
 Number of tubes = 5600
 $\text{PERPTX}(1) = (\pi)(1.125 \times 0.0254)(5600)$
 $= 502.7 \text{ m}$
- DRACS-I heat exchanger
 O.D. of the tubes = 0.75"
 Number of tubes = 306
 $\text{PERPTX}(2) = (\pi)(0.75 \times 0.0254)(306)$
 $= 18.31328 \text{ m}$
- DRACS-II heat exchanger
 $\text{PERPTX}(3) = \text{PERPTX}(2)$
 $= 18.31328 \text{ m}$
- #3532-3535 PERTIX The perimeter between the tubes and the tube side coolant in an IHX.
 IHX
 O.D. of the tubes = 1.125"
 Thickness of the tube wall = 0.045"
 Number of tubes = 5600
 $\text{PERTIX}(1) = (5600)(\pi)((1.125 - 2 \times 0.045) \times 0.0254)$
 $= 462.5 \text{ m}$
- DRACS-I heat exchanger
 O.D. of the tubes = 0.75"
 Thickness of the tube wall = 0.034"
 Number of tubes = 306

$$\begin{aligned}\text{PERTIX}(2) &= (306) * (\pi) * ((0.75 - 2 * 0.034) * 0.0254) \\ &= 16.653 \text{ m}\end{aligned}$$

DRACS-II heat exchanger

$$\begin{aligned}\text{PERTIX}(3) &= \text{PERTIX}(2) \\ &= 16.653 \text{ m}\end{aligned}$$

#3540-3543 DSHIHX

The shell thickness in an IHX.
IHX

$$\begin{aligned}\text{Shell thickness} &= 2" = 0.0508 \text{ m} \\ \text{DSHIHX}(1) &= 0.0508 \text{ m}\end{aligned}$$

DRACS-I heat exchanger

$$\begin{aligned}\text{Shell thickness} &= 0.9" = 0.02286 \text{ m} \\ \text{DSHIHX}(2) &= 0.02286 \text{ m}\end{aligned}$$

DRACS-II heat exchanger

$$\begin{aligned}\text{DSHIHX}(3) &= \text{DSHIHX}(2) \\ &= 0.02286 \text{ m}\end{aligned}$$

#3548-4551 DTUIHX

The tube thickness in an IHX.
IHX

$$\begin{aligned}\text{Tube thickness} &= 0.045" = 1.143 \times 10^{-3} \text{ m} \\ \text{DTUIHX}(1) &= 1.143 \times 10^{-3} \text{ m}\end{aligned}$$

DRACS-I heat exchanger

$$\begin{aligned}\text{Tube thickness} &= 0.034" = 8.636 \times 10^{-4} \text{ m} \\ \text{DTUIHX}(2) &= 8.636 \times 10^{-4} \text{ m}\end{aligned}$$

DRACS-II heat exchanger

$$\begin{aligned}\text{DTUIHX}(3) &= \text{DTUIHX}(2) \\ &= 8.636 \times 10^{-4} \text{ m}\end{aligned}$$

#3556-3559 RCSHHX

The density times the specific heat for the shell
of an IHX.

IHX

$$\begin{aligned}\text{The IHX is fabricated of 9Cr-1Mo steel of which} \\ \text{the density*specific heat is } 4.58 \times 10^6 \text{ J/m}^3\text{-K.} \\ \text{RCSHHX}(1) &= 4.58 \times 10^6 \text{ J/m}^3\text{-K}\end{aligned}$$

DRACS-I heat exchanger

$$\begin{aligned}\text{The shell of the DRACS-I heat exchanger is fabricated} \\ \text{of 304-SS, which has a density*specific heat of} \\ 4.419 \times 10^6 \text{ J/m}^3\text{-K} \\ \text{RCSHHX}(2) &= 4.419 \times 10^6 \text{ J/m}^3\text{-K}\end{aligned}$$

DRACS-II heat exchanger

$$\begin{aligned}\text{RCSHHX}(3) &= \text{RCSHHX}(2) \\ &= 4.419 \times 10^6 \text{ J/m}^3\text{-K}\end{aligned}$$

#3564-3567 RCTUHX

The density times the specific heat of the tubes in an IHX.
IHX

$$\begin{aligned}\text{The tubes in the IHX are of 9Cr-1Mo alloy steel which} \\ \text{has a density*specific heat of } 4.58 \times 10^6 \text{ J/m}^3\text{-K.} \\ \text{RCTUHX}(1) &= 4.58 \times 10^6 \text{ J/m}^3\text{-K}\end{aligned}$$

DRACS-I heat exchanger

The tubes in the DRACS-I heat exchanger are of 304-SS which has a density*specific heat of $4.419\text{D}+06 \text{ J/m}^{**3}\text{-K}$.

$$\text{RCSHHX}(2) = 4.419\text{D}+06 \text{ J/m}^{**3}\text{-K}$$

DRACS-II heat exchanger

$$\begin{aligned}\text{RCSHHX}(3) &= \text{RCSHHX}(2) \\ &= 4.419\text{D}+06 \text{ J/m}^{**3}\text{-K}\end{aligned}$$

#3572-3575 XKSHHX

The thermal conductivity of the shell in an IHX.
IHX

The shell of the IHX is of 9Cr-1Mo alloy steel which has a thermal conductivity of 28.32 W/m-K .

$$\text{XLSHHX}(1) = 28.32 \text{ W/m-K}$$

DRACS-I heat exchanger

The shell of the DRACS-I heat exchanger is of 304-SS which has a thermal conductivity of 21.5 W/m-K .

$$\text{XKSHHX}(2) = 21.5 \text{ W/m-K}$$

DRACS-II heat exchanger

$$\begin{aligned}\text{XKSHHX}(3) &= \text{XKSHHX}(2) \\ &= 21.5 \text{ W/m-K}\end{aligned}$$

#3580-3583 XKTUHX

The thermal conductivity of the tubes in an IHX.
IHX

The tubes in the IHX are of 9Cr-1Mo alloy steel which has a thermal conductivity of 28.32 W/m-K .

$$\text{XKTUHX}(1) = 28.32 \text{ W/m-K}$$

DRACS-I heat exchanger

The tubes in the DRACS-I heat exchanger are of 304-SS which has a thermal conductivity of 21.5 W/m-K .

$$\text{XKTUHX}(2) = 21.5 \text{ W/m-K}$$

DRACS-II heat exchanger

$$\begin{aligned}\text{XKTUHX}(3) &= \text{XKTUHX}(2) \\ &= 21.5 \text{ W/m-K}\end{aligned}$$

#3588-3591 HFPIHX

The fouling heat transfer coefficient for the shell-side flow in an IHX. Based on previous work, this is set to $1.0\text{D}+09$ for the IHX, and the DRACS-I and DRACS-II heat exchangers.

$$\text{HFPIHX}(1) = 1.0\text{D}+09$$

$$\text{HFPIHX}(2) = 1.0\text{D}+09$$

$$\text{HFPIHX}(3) = 1.0\text{D}+09$$

#3596-3599 HFIIHX

The fouling heat transfer coefficient for tube-side flow in an IHX. Based on previous work, this is set to $1.0\text{D}+09$ for the IHX, and the DRACS-I and DRACS-II heat exchangers.

$$\text{HFIIHX}(1) = 1.0\text{D}+09$$

$$\text{HFIIHX}(2) = 1.0\text{D}+09$$

$$\text{HFIIHX}(3) = 1.0\text{D}+09$$

- #3604-3607 SLANTX The slant-height factor of an IHX to account for non-vertical orientation. Since all of the tubes in the IHX and the DRACS-I and DRACS-II heat exchangers are vertically oriented, the value is 1.0.
 $SLANTX(1) = 1.0$
 $SLANTX(2) = 1.0$
 $SLANTX(3) = 1.0$
- #3612-3649 VOLGSO The initial gas volume for each of the compressible volumes that have a cover gas. For compressible volumes without cover gas, the value is 0.0.
- Outlet plenum
 $AREAIN(2) = 202.3 \text{ m}^2$
 Height of the gas space from Fig. D-2 = 3' = 0.9144 m
 $VOLGSO(2) = (202.3) * (0.9144)$
 $= 184.983 \text{ m}^3$
 As discussed for the total volume of this compressible volume, an extra 1000 m^3 was added to prevent over-pressurization of the reactor vessel, which is not realistic. For this reason,
 $VOLGSO(2) = 1184.98 \text{ m}^3$
- Cold pool
 $AREAIN(3) = 16.7 \text{ m}^2$
 Height of gas space from Fig. D-2 = 7' = 2.1336 m
 $VOLGSO(3) = (16.7) * (2.1336)$
 $= 35.63116 \text{ m}^3$
 As discussed for the total volume of this compressible volume, an extra 1000 m^3 was added to prevent over-pressurization of the reactor vessel, which is not realistic. For this reason,
 $VOLGSO(3) = 1035.63 \text{ m}^3$
- Intermediate loop pump bowl
 Assumed to be 1.0 m^3
 $VOLGSO(8) = 1.0 \text{ m}^3$
- Intermediate loop sodium expansion tank
 O.D. = 120"
 Overall length = 164"
 $\text{Tank volume} = (\pi) * ((120 * 0.0254)^2) * (164 * 0.0254)$
 $= 30.395 \text{ m}^3$
 For the calculation of the interface area, it was assumed that 50% of the tank is occupied by gas.
 $VOLGSO(10) = 0.50 * 30.395$
 $= 15.197 \text{ m}^3$
- DRACS loops sodium storage tanks
 Assume that each tank is half full,
 $VOLGSO(11) = 10.0 \text{ m}^3$
 $VOLGSO(12) = 10.0 \text{ m}^3$

#3650	C1PIPE	Coefficients in Lyon-Martinelli convective heat-transfer correlation for pipe elements. =0.025 (recommended in input description)
#3651	C2PIPE	=0.8 (recommended in input description)
#3652	C3PIPE	=4.8 (recommended in input description)
#3653	C1IHx	Coefficients in Lyon-Martinelli convective heat-transfer correlation for heat exchangers. =0.025 (recommended in input description)
#3654	C2IHx	=0.8 (recommended in input description)
#3655	C3IHx	=4.8 (recommended in input description)
#3801-3804	RITB	The inner radius of the tubes in the air dump heat exchangers. DRACS-I air dump heat exchanger O.D. of the finned tubes = 1.9" Tube wall thickness = 0.075" $RITB(1) = ((1.9 - 2 \times 0.075) / 2) \times 0.0254$ = 2.2225D-02 m DRACS-II heat exchanger $RITB(2) = RITB(1)$ = 2.2225D-02 m
#3805-3808	ROTB	The outer radius of the tubes in the air dump heat exchangers. DRACS-I air dump heat exchanger O.D. of the finned tubes = 1.9" $ROTB(1) = (1.9 / 2) \times 0.0254$ = 2.413D-02 m DRACS-II air dump heat exchanger $ROTB(2) = ROTB(1)$ = 2.413D-02 m
#3809-3812	XKTB	The thermal conductivity of the tubes in the air dump heat exchangers. DRACS-I air dump heat exchanger $XKTB(1) = 21.5 \text{ W/m-K (for 304-SS tubes)}$ DRACS-II air dump heat exchanger $XKTB(2) = XKTB(1)$ = 21.5 W/m-K
#3813-3816	HOTB	The air dump heat exchanger tube-to-air heat transfer coefficient. Values are based on previous work. $HOTB(1) = 37.3 \text{ W/m-K}$ $HOTB(2) = 37.3 \text{ W/m-K}$
#3817-3820	HITB	The air dump heat exchanger tube-to-Na heat transfer coefficient. Values are based on previous work. $HITB(1) = 8.6D+04 \text{ W/m-K}$ $HITB(2) = 8.6D+04 \text{ W/m-K}$

- #3821-3824 AHT The air dump heat exchanger air-side heat transfer area.
 DRACS-I air dump heat exchanger
 O.D. of the finned tubes = 1.9"
 Fin height = 1"
 Number of finned tubes = 63
 Number of fins per inch = 9
 Tube length per pass = 9.2' = 110.4"
 Number of passes per tube = 4
 Number of fins per pass = 110.4 * 9 = 994
 Surface area per fin

$$= 2 * (\pi / 4) * ((3.9 * 0.0254)^2 - (1.9 * 0.0254)^2)$$

$$= 0.011756 \text{ m}^2$$
 Surface area per finned tube = 4 * 994 * 0.011756

$$= 46.7403 \text{ m}^2$$
 Total fin surface area = 63 * 46.7403

$$= 2944.66 \text{ m}^2$$
 Total outer surface area of tubes

$$= 63 * 4 * (\pi) * (1.9 * 0.0254) * (110.4 * 0.0254)$$

$$= 114.5 \text{ m}^2$$

$$\text{AHT}(1) = 2944.66 + 114.5$$

$$= 3059.16 \text{ m}^2$$

 DRACS-II air dump heat exchanger

$$\text{AHT}(2) = \text{AHT}(1)$$

$$= 3059.16 \text{ m}^2$$

 #3825-3828 SRLN The stack riser length for the air dump heat exchanger.
 DRACS-I air dump heat exchanger

$$\text{XLENEL}(48) = 3.7084 \text{ m}$$

$$\text{XLENEL}(49) = 25.6794 \text{ m}$$

$$\text{SRLN}(1) = 3.7084 + 25.6794$$

$$= 29.3878 \text{ m}$$

 DRACS-II air dump heat exchanger

$$\text{SRLN}(2) = \text{SRLN}(1)$$

$$= 29.3878 \text{ m}$$

 #3829-3832 XKHXL The loss coefficient in the air dump heat exchanger for the air side of the heat exchanger. Based on previous work, a value of 0.84 is used.

$$\text{XKHXL}(1) = 0.84$$

$$\text{XKHXL}(2) = 0.84$$

 #3833-3836 XKRLS The loss coefficient in the riser of the air dump heat exchanger. Based on previous work, a value of 0.01 is used.

$$\text{XKRLS}(1) = 0.01$$

$$\text{XKRLS}(2) = 0.01$$

 #3837-3840 XKSOLS The stack outlet loss coefficient for the air dump heat exchanger. No additional losses are assumed here.

$$\text{XKSOLS}(1) = 0.0$$

$$\text{XKSOLS}(2) = 0.0$$

- #3841-3844 AHX The air flow area of the air dump heat exchanger.
 DRACS-I air dump heat exchanger
 Finned tube O.D. = 1.9"
 Fin height = 1.0"
 Number of fins per inch = 9
 Number of tubes = 63
 Number of passes per tube = 4
 Tube length per pass = 9.2'
 Number of fins per pass = $9.2 \times 12 \times 9 = 994$
 Fin thickness = $1/32$ " (assumed)
 Inside area = $(118 \times 0.0254) \times (32 \times 12 \times 0.0254)$
 $= 29.2335 \text{ m}^2$
 Projected area of tubes and fins
 $= 63 \times (118 \times 1.9 + 2 \times (1 \times 1/32) \times 994) \times (0.0254^2)$
 $= 11.6377 \text{ m}^2$
 $\text{AHX}(1) = 29.2335 - 11.6377$
 $= 17.595 \text{ m}^2$

 DRACS-II air dump heat exchanger
 $\text{AHX}(2) = \text{AHX}(1)$
 $= 17.595 \text{ m}^2$
- #3845-3848 AR The air dump heat exchanger riser cross-sectional area.
 DRACS-I air dump heat exchanger
 This is inside area calculated for AHX.
 $\text{AR}(1) = 29.2335 \text{ m}^2$

 DRACS-II air dump heat exchanger
 $\text{AR}(2) = \text{AR}(1)$
 $= 29.2335 \text{ m}^2$
- #3849-3852 ASI The air dump heat exchanger stack inlet cross-sectional area. From Fig. D-9,
 DRACS-I air dump heat exchanger
 Stack inlet opening = $118" \times 20'$
 $\text{ASI}(1) = (118 \times 0.0254) \times (20 \times 12 \times 0.0254)$
 $= 18.27093 \text{ m}^2$

 DRACS-II air dump heat exchanger
 $\text{ASI}(2) = \text{ASI}(1)$
 $= 18.27093 \text{ m}^2$
- #3853-3892 XKSILS Table of the stack inlet loss coefficient versus time.
 Based on calculations for the steady-state conditions
 and the rated capacity of 14 MW for each DRACS loop,
 the following values were obtained.
 DRACS-I heat exchanger
 $\text{XKSILS} = 143.6$

 DRACS-II heat exchanger
 $\text{XKSILS} = 130.8$
- #3893-3932 XKSITM Time for XKSILS table.
 Set to 0.0 and 10000.0 s for both DRACS-I and DRACS-II

#3933 AIRTMP The inlet air temperature is assumed to be 300 K.

#4174-4183 VSLEXP Length times thermal expansion coefficient for the compressible volume walls used for calculating the motion of the core with respect to the control rod driveline due to reactor vessel expansion. Assume the thermal expansion coefficient is 2.0×10^{-5} per K.

Oxide Fuel		Metallic Fuel	
CV	Length	VSLEXP	Length
2	6.3 m	1.2598D-04	6.3 m
6	0.762 m	1.524D-05	0.762 m
7	0.762 m	1.524D-05	0.762 m
3	0.1016 m	2.032D-061	0.661 m
			VSLEXP
			1.2598D-04
			1.524D-05
			1.524D-05
			1.3208D-05

#4284-4287 DTIHX The steady-state temperature drop across the primary side of a heat exchanger. This is set to 168 K for both DRACS-I and DRACS-II. This option was required due to the DRACS-I heat exchanger having both the primary side inlet and outlet connected to the same compressible volume, CV2.

REFERENCES

- D1. G. Birgersson et al., "The SAS4A LMFBR Accident Analysis Code System," ANL/RAS 83-38, Revision 2 (February 1988).
- D2. R. B. Vilim and R. N. Hill, "The Steady-State Thermal-Hydraulic Performance of Interchangeable 3500 MWth Metal and Oxide Fueled LMRs," ANL-FRA-164 (March 1989).
- D3. LSPB - Large Scale Prototype Breeder, CDS 400-8P, Revision 1, Electric Power Research Institute (September 1986).

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APPENDIX E. DEFORM-4 INPUT DATA FOR THE OXIDE FUEL CORE

The SAS4A code contains a detailed fuel behavior model called DEFORM-4 for oxide fuel. The model calculates the effects of irradiation on oxide fuel with 316 SS cladding, including fuel pellet growth, porosity formation and migration, cracking, and fuel/cladding stress. The following description provides the data to be added to several of the input data blocks. The listing in Appendix F includes the DEFORM-4 data as a separate listing, so that the data can be included when desired.

I. Block 1, INPCOM

Location, Input Variable

26	IYLD	The cladding flow stress is based on a table of values, input in Block 13, Loc. 910-969; IYLD = 3.
28	IMELTV	The assumption for the axial extent of the molten cavity pressure calculation is for a cavity which includes all of the central cavity; IMELTV = 1.

II. Block 13, PMATCM

Location, Input Variable

171-183	RHOTAB (L,5)	The fuel type used for the internal blanket assemblies is in table 5. This table originally contained data for UO ₂ with 93.3% of the theoretical density. The DEFORM-4 models start with 100% theoretical densities, so the values originally in table 3, Loc. 131-143, were copied into table 5, Loc. 171-183.
600	FGMM	The molecular weight of a fission gas atom is set to the suggested value of 131.0.
601	GATPF	The number of gas atoms generated per fission is set to the suggested value of 0.246.
602	ENPF	The energy released per fission is set to the suggested value of 197.0.
603	RLEQ	The fraction of the as-fabricated fuel porosity which defines the equiaxed/columnar fuel boundary is set to the suggested value of 0.6.
604	RUEQ	The fuel grain size ratio at the equiaxed/as-fabricated boundary which defines the boundary is set to the suggested value of 1.2.
605	PRSMIN	The minimum attainable fuel porosity which occurs during fuel restructuring is set to the suggested value of 0.02.

910-920 YLDTAB The cladding yield point as a function of temperature is
970-980 YLDTEM set according to the following table;

Cladding Yield Point, Pa	Temperature, K
8.48000D+08	255.35
7.58000D+08	366.45
7.14000D+08	477.55
6.96000D+08	588.75
6.76000D+08	699.85
6.36000D+08	810.95
4.62000D+08	922.05
2.62000D+08	1033.15
1.38000D+08	1144.25
0.34500D+08	1255.35
0.0	1648.15

1073-1076 PRSTY As-fabricated porosity for each fuel type.

Fuel Type	Porosity
Mixed Oxide 86.8% TD	0.132
Mixed Oxide 82.5% TD	0.175
UO2 86.8% TD	0.132
UO2 82.5% TD	0.175
UO2 93.3% TD	0.067

1077 PRSTY (5) The as-fabricated fuel porosity for the internal blanket fuel, table 5, is 0.043.

1083 APORE The pre-exponential factor in the pore velocity equation is set to the suggested value of 20.704.

1084 QPORE The pore migration activation energy in set to the suggested value of 4.5281×10^5 .

1085 ABC The exponent of the temperature factor in the pore velocity equation is set to the suggested value of 1.5.

1086 RGASSI The ideal gas constant for use in DEFORM-4 is set to the suggested value of 8.31434.

1087 GAMMA The fuel surface tension is set to the suggested value of 0.45.

1088 APG The pre-exponential factor in the bubble radius equation is set to the suggested value of 5.0×10^4 .

1089 QPG The temperature dependence of bubble radius in the bubble radius equation is set to the suggested value of 5.65065×10^4 .

1090 GK The pre-exponential factor for grain growth in the equation for unlimited grain growth rate is set to the suggested value of 1.717×10^{10} .

1091	QV	The activation energy of unlimited grain growth is set to the suggested value of 3.87×10^5 .
1092	GK1	The pre-exponential factor for grain growth in the equation for limited grain growth rate is set to the suggested value of 1.45556×10^{-8} .
1093	QV1	The temperature dependence in the equation for limited grain growth rate is set to the suggested value of 2.67×10^5 .
1094	GRAINK	The pre-exponential factor in the equation for calculating the maximum grain size is set to the suggested value of 2.23×10^{-3} .
1095	GRAINQ	The temperature dependence in the equation for calculating the maximum grain size is set to the suggested value of 6.3375×10^4 .
1096	CVXE	The specific heat of xenon at constant volume is set to the suggested value of 94.69.
1097	CVHE	The specific heat of helium at constant volume is set to the suggested value of 3.13×10^3 .
1098	ROFF	The surface roughness of the fuel is set to the suggested value of 3.3×10^{-6} .
1099	ROFC	The surface roughness of the inner surface of the cladding is set to the suggested value of 1.78×10^{-6} .
1101	GAMGS	The ratio of specific heats for the gas, c_p/c_v , in the jump distance calculation is set to the suggested value of 1.66.
1103	ET	The exponent of the pressure dependence in the solid/solid conductance equation is set to the suggested value of 1.0.
1104	ACCHE	The accommodation coefficient of helium is set to the suggested value of 0.15.
1105	ACCXE	The accommodation coefficient of xenon is set to the suggested value of 0.805.
1106	CZERO	The calibration constant for the surface roughness in the gap conductance calculation is set to 1.0.
1107	STEBOL	The Stefan-Boltzmann constant is set to the suggested value of 5.67×10^{-8} .
1108	EMSF	The fuel thermal emissivity is set to the suggested value of 0.9.
1109	EMSC	The cladding thermal emissivity is set to the suggested value of 0.8.

1110	QA1	This parameter in the Weisman fission gas release model is set to the suggested value of 6.92×10^3 .
1111	QA2	This parameter in the Weisman fission gas release model is set to the suggested value of 33.95.
1112	QA3	This parameter in the Weisman fission gas release model is set to the suggested value of 0.338.
1113	QA4	This parameter in the Weisman fission gas release model is set to the suggested value of 1.48×10^4 .
1114	QA5	This parameter in the Weisman fission gas release model is set to the suggested value of 9.575.
1115	ALFSS	The pre-exponential factor in the equation for isotropic fission gas release is set to the suggested value of 2.0×10^{-4} .
1116	BETSS	The temperature dependence in the equation for isotropic fission gas release is set to the suggested value of 1.1×10^4 .
1117	CNU	The Poisson ratio for the cladding is set to the suggested value of 0.3.
1118	FNU	The Poisson ratio for the fuel is set to the suggested value of 0.3265.
1119	AM	The exponent of stress in the fuel creep stress relaxation calculation is set to the suggested value of 1.0.
1120	QLAX	The activation energy for diffusion creep is set to the suggested value of 3.77×10^5 .
1124	RGAS	The gas constant per kilogram for fission gas is set to the suggested value of 65 for high burnup fuel.
1225	HEMM	The molecular mass of a helium atom is set to the suggested value of 4.
1226	EPSSFP	The fraction of volume swelling due to solid fission products is set to the suggested value of 3.0×10^{-3} .
1258	FAXIAL	The fraction of the calculated axial expansion used by DEFORM-4 is set to the suggested value of 1.0.
1259	FCLDWK	The initial cold-work strain of the cladding is set to 0.2.
1261	FSTRAN	The maximum allowable cladding strain is set to 0.10
1262	FTMPCH	The fraction of fuel or blanket melting temperature at which crack healing occurs is set to 0.60.

- 1263 EXPCOF The fraction of the axial expansion reactivity calculated by DEFORM-4 to be used for feedback effects is set to 1.0.
- 1275 FGFI The mole fraction of fission gas in the initial fill gas is set to 0.10.

III. Block 51, INPCHN

Location, Variable

- 3 IRHOK The model selection for fuel density and thermal conductivity chooses the functional form for fuel conductivity based on COMETH3J-FBR, which is temperature and porosity dependent. The fuel thermal conductivity tables are ignored; IRHOK = 3.
- 15 IFUELV The property table for the core fuel selects either mixed oxide or UO₂, depending on the channel. For use with DEFORM-4, table with properties based on 100% theoretical density are used; see Appendix A.
Channel	Core Fuel Table
1	1
2	1
3	5
- 16 IFUELB The property table for the axial blanket fuel selects UO₂ in this case. For use with DEFORM-4, tables with properties based on 100% theoretical density are used; see Appendix A.
Channel	Axial Blanket Table
1	3
2	3
3	5
- 20 NAXOP The model selection for axial expansion and crack volume chooses the model with crack volume and axial plane strain, but not axial swelling; NAXOP = 5.
- 21 MSTEP The number of main steady-state power change and constant power intervals for DEFORM-4 fuel characterization. For the current calculations, this is set to 4 intervals, with the data listed in Block 62.
- 22 ITAU The nominal value of irradiation induced cladding swelling incubation is used; ITAU = 0.
- 23 IRATE The nominal value for the irradiation induced cladding swelling rate is used; IRATE = 0.
- 24 IHGAP The SAS4A method for calculating the fuel/cladding gap conductance is used; IHGAP = 1.

- 31 NGRAIN The suggested value of 4 is used for the model selection for grain growth theory.
- 32 ISSFUE This value is set to 1 to activate the DEFORM-4 calculation in the steady-state. If this value is 1, the value of ISSFU2, Block 51 Loc. 122, must also be 1.
- 37-40 NPLIN
(MSTEP) The number of subdivisions in each of the MSTEP (Block 1, Loc. 21) divisions. The suggested arrangement for 292 days is 10 days from zero power to full power, MSTEP(1), 30 days at full power, MSTEP(2), 60 days at full power, MSTEP(3), and 192 days at full power, MSTEP(4), for a total of 292 days. For the first 2 divisions, the length of each subdivision is 1 day, for the third division it is 3 days, and for the last division it is 9.6 days:
MSTEP	NPLIN	MSTEP Length	Length per subdivision
1	10	10 days	1 day
2	30	30 days	1 day
3	20	60 days	3 days
4	20	192 days	9.6 days
- 51-54 NSKIP
(MSTEP) The printing control for DEFORM-4 results during the steady-state irradiation. For NSKIP > NPLIN, no results are printed.
MSTEP	NSKIP	Results
1	99	No results printed
2	99	No results printed
3	99	No results printed
4	30	No results printed
- 95 IPSIG Selection of the equation to be used for the hydrostatic pressure for fuel swelling; IPSIG = 3.
- 122 ISSFU2 Activates DEFORM-4 for the transient calculation. This must be set to 1 if ISSFUE = 1; ISSFU2 = 1.
- 123 IHEALC It is assumed that there is no crack healing, so IHEALC = 1.
- 124 IAXTHF Both thermal and force effects are assumed to be active in the calculation of axial expansion; IAXTHF = 1.
- 181 IAXEXP The simple axial expansion model is deactivated when DEFORM-4 is used; IAXEXP = 0.

IV. Block 61, GEOMIN

Location, Input Variable

- 128 ROUTFP The fuel outer radius in each axial segment in the core is determined as follows;
Fuel pin smear density:	82.5%	
Fuel pellet density:	86.8%	
Fuel pellet radius =	$(0.825/0.868)^{1/2}$	clad inner radius
	$= 2.98392 \times 10^{-3} \text{ m}$	

Blanket pin smear density: 93.3%
 Blanket pellet density: 95.7%
 Blanket pellet radius = $(0.933/0.957)^{1/2}$ clad inner radius
 = 4.36383×10^{-3} m

V. Block 62, POWINC

Location, Input Variable

45-48 PLIN The relative power at the end of each division and the
 (MSTEP) cumulative time at the end of each division. See Block
 53-56 TPLIN 51, Loc. 21, 37-44. The irradiation history is assumed
 (MSTEP) to be the same for each channel.

MSTEP	PLIN	TPLIN
1	1.0	$8.64000 \times 10+5$
2	1.0	$3.45600 \times 10+6$
3	1.0	$8.64000 \times 10+6$
4	1.0	$5.04576 \times 10+7$

61 FLTPOW The fast flux to linear power ratio is obtained from
 Ref. 6 in the main report; FLTPOW = $8.50624 \times 10+12$.

VI. Block 63, PMATCH

Location, Input Variable

5 HBMAX The maximum value of the gap conductance when a gap
 exists is set to $3.0 \times 10+4$.

6 HBMIN The minimum value of the gap conductance is set to
 100.0.

26 DGO The initial grain size is set to the suggested value of
 $1.0 \times 10-5$.

27 POGAS The initial gas pressure in the fission gas plenum is
 set to atmospheric pressure; POGAS = $1.01325 \times 10+5$ Pa.

60 FGMIN The fraction of the retained fission gas that assists
 nonequilibrium fission gas bubble induced fuel swelling
 is set to 0.30.

69 TSEP1 The coefficients for the equation of the fuel plasticity
 70 TSEP2 temperature are set to the suggested values:
 TSEP1 = 2700.0
 TSEP2 = $1.0 \times 10-8$

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APPENDIX F. SAS4A INPUT DATA LISTING FOR THE OXIDE FUEL CORE

This appendix contains the listing of the SAS4A input data for the oxide fuel core. The first section covers the basic data required for calculating the steady-state conditions without DEFORM-4. The input data for running the DEFORM-4 models is included in the second section, while the third section has the additional data required for running the transients described in this report.

I. Steady-State Input Data

The steady-state input data contains all the data required for initializing the core conditions prior to any particular transient. The data also include values for models which are not used in the steady-state calculation, but which are identical for all accidents. The generation of this input data is discussed in detail in Appendix A for the core modelling, Appendix C for the radial core expansion reactivity feedback model, and Appendix D for the modelling of the remainder of the reactor plant. The resulting input data listing is as follows:

3500 MWE OXIDE FUEL CORE, LSPB PLANT, ANL/EUROPEAN COMPARISON STUDY
3 CHANNEL DECK FOR ATWS TRANSIENTS, ARGONNE NATL LAB., 4/17/89

3 0 0 0

XNOTES

THREE CHANNEL OXIDE FUEL DECK, FIVE FUEL TYPES, ONE CLADDING
TYPE. SIX DELAYED NEUTRON PRECURSOR GROUPS, SIX DECAY HEAT
GROUPS. PRIMAR-4 REPRESENTATION OF THE REACTOR SYSTEM.
DETAILED RADIAL CORE EXPANSION REACTIVITY FEEDBACK MODEL.

-1

INPCOM	1	1	0								
1	9	3	0	5	1	1	0	0	0	0	
11	10	0	0	0	0	100	6	6	0	0	0
21	9	0	1	0	11	0	0	4	0	0	
31	1	2									
36	4	-4	0	0	-5						
45	2	1	1								
51	6	595	1	2	2	0	1				
59	7	991	1	0	1	2	2	1			

-1

OPCIN	11	1	0							
1	1	1.00000D-01								
3	1	1.00000D-05								
5	2	2.50000D-01	1.00000D-02							
9	3	1.00000D+01	5.00000D+01	3.00000D+01						
26	2	20.	0.001							

-1

XNOTES

NEUTRONICS FOR OXIDE CORE DESIGN AS GIVEN IN DATA PACKAGE
SAS CHANNELS REPRESENT DRIVERS AND INTERNAL BLANKETS
DECAY HEAT REPEATED FROM EARLIER DESIGN
CORE LOAD PAD DIMENSIONS RESULT IN GRID PLATE PITCH EQUALLING
THE ABOVE-CORE LOAD PAD PITCH AT NOMINAL OPERATING CONDITIONS
CORE RESTRAINT CLEARANCE FOR DEFINED CORE GEOMETRY FOR P/F GT .7

-1

POWINA	12	1	0			
2	2	4.05400D-07	3.50000D+09			
4	4	7.76170D-05	7.06290D-04	6.13530D-04	1.19000D-03	
8	2	5.44450D-04	1.69620D-04			
10	4	1.29750D-02	3.13260D-02	1.35070D-01	3.46130D-01	
14	2	1.37920D+00	3.74970D+00			
69	2	9.68323D-01	9.15444D-01			
71	4	6.29920D+00	2.00000D-05	-7.12000D+00	0.00000D+00	
75	3	5.87927D+04	2.41470D+03	2.62467D+01		
260	5	2.20054E-02	1.90590E-02	9.93659E-03	5.25396E-03	1.93225E-03
265	1	1.67409E-03				
290	5	6.56251E-02	1.77911E-03	1.37667E-04	6.08268E-06	2.25624E-07
295	1	1.89684E-08				
320	1	6.22500D-02				
325	1	1.0				
365	1	2.52288E+07				
408	4	1.00000D+00	1.54827D-01	1.54887D-01	1.54887D-01	
412	2	-6.40000D+01	-7.00000D-05			
414	2	1.00000D-03	1.00000D-03			
423	2	1.00000D-01	3.50000D-01			
426	1	1.62400D-03				
428	1	5.00000D+02				
-1						

XNOTES

FUEL PROPERTY TABLES

FUEL DENSITY TABLES

TABLE 1: MIXED OXIDE, THEORETICAL DENSITY

TABLE 2: MIXED OXIDE, FUEL PIN SMEAR DENSITY (82.5%)

TABLE 3: UO₂, THEORETICAL DENSITYTABLE 4: UO₂, AXIAL BLANKET SMEAR DENSITY (82.5%)TABLE 5: UO₂, INTERNAL AND RADIAL BLANKET SMEAR DENSITY (93.3%)

THERMAL CONDUCTIVITY TABLES

TABLES 1-4, THEORETICAL VALUE REDUCED TO 82.1% FOR ASSUMED 10%
AVERAGE FUEL POROSITYTABLE 5, THEORETICAL VALUE REDUCED TO 91.7% FOR ASSUMED 5%
AVERAGE FUEL POROSITY

HEAT CAPACITY TABLES

TABLES 1-5, HEAT CAPACITY FOR UO₂ USED FOR ALL FUEL

SOLIDUS AND LIQUIDUS TEMPERATURES, HEAT OF FUSION

TABLES 1-2, MIXED OXIDE

TABLES 3-5, UO₂

CLADDING TABLES

TABLE 1, HT-9

-1						
PMATCM	13	1	0			
1	3	1.10800D 04	2.04000D-05	8.70000D-09		
4	5	2.10000D 00	2.88000D-03	2.52000D-05	5.83000D-10	5.75000D-02
9	2	5.03000D-04	2.91000D-11			
11	5	2.49000D+01	2.54000D+01	2.59000D+01	2.63000D+01	2.68000D+01

16	3	2.72000D+01	2.76000D+01	2.80000D+01				
71	5	3.73000D+02	4.73000D+02	5.73000D+02	6.73000D+02	7.73000D+02		
76	3	8.73000D+02	9.73000D+02	1.07300D+03				
91	5	1.09606D 04	1.08782D 04	1.07805D 04	1.06674D 04	1.05401D 04		
96	5	1.03974D 04	1.02415D 04	1.00702D 04	9.88690D 03	9.75626D 03		
101	5	9.29485D 03	9.29485D 03	8.92916D 03	8.88086D 03	8.35182D 03		
111	5	9.04246D 03	8.97453D 03	8.89392D 03	8.80064D 03	8.69558D 03		
116	5	8.57784D 03	8.44923D 03	8.30794D 03	8.15669D 03	8.04892D 03		
121	5	7.66825D 03	7.66825D 03	7.36656D 03	7.32671D 03	7.17385D 03		
131	5	1.09387D 04	1.08569D 04	1.07596D 04	1.06470D 04	1.05194D 04		
136	5	1.03774D 04	1.02211D 04	1.00510D 04	9.86748D 03	9.67080D 03		
141	3	9.63883D 03	8.86000D 03	8.74092D 03				
151	5	9.02443D 03	8.95694D 03	8.87667D 03	8.78378D 03	8.67851D 03		
156	5	8.56136D 03	8.43241D 03	8.29208D 03	8.14067D 03	7.97841D 03		
161	3	7.95203D 03	8.86000D 03	8.74092D 03				
171	5	1.02058D 04	1.01295D 04	1.00387D 04	9.93365D 03	9.81460D 03		
176	5	9.68211D 03	9.53629D 03	9.37758D 03	9.20636D 03	9.02286D 03		
181	3	8.99303D 03	8.86000D 03	8.74092D 03				
251	5	3.73150D 02	6.73150D 02	9.73150D 02	1.27315D 03	1.57315D 03		
256	5	1.87315D 03	2.17315D 03	2.47315D 03	2.77315D 03	2.97515D 03		
261	5	3.00000D 03	3.00000D 03	3.02515D 03	3.07315D 03	3.67315D 03		
271	5	3.73150D 02	6.73150D 02	9.73150D 02	1.27315D 03	1.57315D 03		
276	5	1.87315D 03	2.17315D 03	2.47315D 03	2.77315D 03	2.97515D 03		
281	5	3.00000D 03	3.00000D 03	3.02515D 03	3.07315D 03	3.67315D 03		
291	5	3.73150D 02	6.73150D 02	9.73150D 02	1.27315D 03	1.57315D 03		
296	5	1.87315D 03	2.17315D 03	2.47315D 03	2.77315D 03	3.07315D 03		
301	3	3.12015D 03	3.12015D 03	3.25015D 03				
311	5	3.73150D 02	6.73150D 02	9.73150D 02	1.27315D 03	1.57315D 03		
316	5	1.87315D 03	2.17315D 03	2.47315D 03	2.77315D 03	3.07315D 03		
321	3	3.12015D 03	3.12015D 03	3.25015D 03				
331	5	3.73150D 02	6.73150D 02	9.73150D 02	1.27315D 03	1.57315D 03		
336	5	1.87315D 03	2.17315D 03	2.47315D 03	2.77315D 03	3.07315D 03		
341	3	3.12015D 03	3.12015D 03	3.25015D 03				
419	1	3.00150D 02						
420	5	1.07460D 01	6.18860D 00	4.36000D 00	3.38800D 00	2.80100D 00		
425	5	2.42800D 00	2.19200D 00	2.05400D 00	1.99400D 00	1.99900D 00		
430	5	2.06400D 00	2.18500D 00	2.36100D 00	2.59300D 00	2.88000D 00		
440	5	1.07460D 01	6.18960D 00	4.36000D 00	3.38800D 00	2.80100D 00		
445	5	2.42800D 00	2.19200D 00	2.05400D 00	1.99400D 00	1.99900D 00		
450	5	2.06400D 00	2.18500D 00	2.36100D 00	2.59300D 00	2.88000D 00		
460	5	7.06022D 00	5.31637D 00	4.20760D 00	3.44427D 00	2.88957D 00		
465	5	2.47247D 00	2.15777D 00	1.93512D 00	1.81340D 00	1.81595D 00		
470	5	1.97569D 00	2.33061D 00	2.91999D 00	2.87022D 00	2.83327D 00		
480	5	7.06022D 00	5.31637D 00	4.20760D 00	3.44427D 00	2.88957D 00		
485	5	2.47247D 00	2.15777D 00	1.93512D 00	1.81340D 00	1.81595D 00		
490	5	1.97569D 00	2.33061D 00	2.91999D 00	2.87022D 00	2.83327D 00		
500	5	7.88149D 00	5.93479D 00	4.69705D 00	3.84492D 00	3.22570D 00		
505	5	2.76008D 00	2.40877D 00	2.16022D 00	2.02434D 00	2.02718D 00		
510	5	2.20551D 00	2.60171D 00	3.25966D 00	3.20409D 00	3.16285D 00		
580	5	2.73150D 02	4.73150D 02	6.73150D 02	8.73150D 02	1.07315D 03		
585	5	1.27315D 03	1.47315D 03	1.67315D 03	1.87315D 03	2.07315D 03		
590	5	2.27315D 03	2.47315D 03	2.67315D 03	2.87315D 03	3.07315D 03		
606	5	2.24688D 02	2.76104D 02	2.95143D 02	3.06360D 02	3.14853D 02		
611	5	3.22211D 02	3.29373D 02	3.37970D 02	3.51533D 02	3.76340D 02		

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616 5 4.21481D 02 4.98140D 02 6.18439D 02 6.18667D 02 4.85000D 02
626 5 2.24688D 02 2.76104D 02 2.95143D 02 3.06360D 02 3.14853D 02
631 5 3.22211D 02 3.29373D 02 3.37970D 02 3.51533D 02 3.76340D 02
636 5 4.21481D 02 4.98140D 02 6.18439D 02 6.18667D 02 4.85000D 02
646 5 2.24688D 02 2.76104D 02 2.95143D 02 3.06360D 02 3.14853D 02
651 5 3.22211D 02 3.29373D 02 3.37970D 02 3.51533D 02 3.76340D 02
656 5 4.21481D 02 4.98140D 02 6.18439D 02 6.18667D 02 4.85000D 02
666 5 2.24688D 02 2.76104D 02 2.95143D 02 3.06360D 02 3.14853D 02
671 5 3.22211D 02 3.29373D 02 3.37970D 02 3.51533D 02 3.76340D 02
676 5 4.21481D 02 4.98140D 02 6.18439D 02 6.18667D 02 4.85000D 02
686 5 2.24688D 02 2.76104D 02 2.95143D 02 3.06360D 02 3.14853D 02
691 5 3.22211D 02 3.29373D 02 3.37970D 02 3.51533D 02 3.76340D 02
696 5 4.21481D 02 4.98140D 02 6.18439D 02 6.18667D 02 4.85000D 02
766 5 2.73150D 02 4.73150D 02 6.73150D 02 8.73150D 02 1.07315D 03
771 5 1.27315D 03 1.47315D 03 1.67315D 03 1.87315D 03 2.07315D 03
776 5 2.27315D 03 2.47315D 03 2.67315D 03 3.12000D 03 3.12000D 03
786 5 2.97515D 03 2.97515D 03 3.11015D 03 3.11015D 03 3.11015D 03
794 5 3.02115D 03 3.02115D 03 3.13015D 03 3.13015D 03 3.13015D 03
802 5 2.64896D 05 2.64896D 05 2.77037D 05 2.77037D 05 2.77037D 05
810 1 1690.15
813 1 1710.15
816 1 2.683 D 05
819 5 4.82500D+02 5.21480D+02 5.63370D+02 6.15380D+02 6.84680D+02
824 5 7.84960D+02 8.66080D+02 1.02150D+03 1.21000D+03 8.68340D+02
829 2 7.47970D+02 7.62140D+02
879 5 3.73000D+02 4.73000D+02 5.73000D+02 6.73000D+02 7.73000D+02
884 5 8.73000D+02 9.23000D+02 9.73000D+02 9.98000D+02 1.02300D+03
889 2 1.07300D+03 1.08856D+03
990 5 3.73566D+06 4.02379D+06 4.33085D+06 4.71178D+06 5.22021D+06
995 5 5.95863D+06 6.55969D+06 7.71948D+06 9.13369D+06 6.54728D+06
1000 2 5.62698D+06 5.72962D+06
1050 5 3.73000D+02 4.73000D+02 5.73000D+02 6.73000D+02 7.73000D+02
1055 5 8.73000D+02 9.23000D+02 9.73000D+02 9.98000D+02 1.02300D+03
1060 2 1.07300D+03 1.08856D+03
1086 1 8.31434D+00

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-1

XNOTES

COVER GAS ASSUMED TO BE AT ATMOSPHERIC PRESSURE AT OPERATION

-1

PRIMIN

14 1 0

1 1 1.52599D 05

45 1 6.30150D 02

65 1 0.

85 1 7.83000D 02

87 2 -1.7018 2.9972

90 2 0.1 0.25

-1

XNOTES

CHANNEL 1 - PEAK DRIVER ASSEMBLY, MAX. POWER AND POWER-TO-FLOW

CHANNEL 2 - AVERAGE OF THE REMAINING DRIVER ASSEMBLIES

CHANNEL 3 - AVERAGE OF THE INTERNAL BLANKET ASSEMBLIES

-1

INPCHN	51	1	0			
4	3	6	1	2		
7	4	3	30	2	4	
14	4	11	2	4	1	
25	4	271	12	5	4	
90	5	12	29	29	29	12
118	1	1				
181	1	1				
203	1	1				
-1						
INPCHN	51	2	1			
25	2	271	390			
-1						
INPCHN	51	3	2			
15	2	5	5			
25	2	169	157			
-1						
GEOMIN	61	1	0			
1	4	2.29993D-05	2.29993D-05	2.29993D-05	2.29993D-05	
8	4	1.27000D-01	1.27000D-01	5.08000D-02	5.08000D-02	
12	4	6.77335D-02	6.77335D-02	6.77335D-02	6.77335D-02	
16	4	6.77335D-02	6.77335D-02	6.77335D-02	6.77335D-02	
20	4	6.77335D-02	6.77335D-02	6.77335D-02	6.77335D-02	
24	4	6.77335D-02	6.77335D-02	6.77335D-02	5.08000D-02	
28	4	5.08000D-02	8.46668D-02	8.46668D-02	8.46668D-02	
32	4	3.23022D-03	3.23022D-03	3.23022D-03	3.23022D-03	
39	4	2.20261D-03	2.20261D-03	2.20261D-03	2.20261D-03	
46	4	2.20261D-03	2.20261D-03	2.20261D-03	2.20261D-03	
53	1	1.0160				
54	1	3.06070D-03				
78	1	3.61950D-03				
102	2	3.06070D-03	3.61950D-03			
104	1	1.00000D-06				
128	1	3.06070D-03				
152	4	6.85800D-01	2.74320D+00	4.06400D-01	8.63600D-01	
159	4	2.16085D-03	2.16085D-03	2.16085D-03	2.16085D-03	
166	1	6.41566D-05				
169	4	9.04875D-04	5.58800D-04	2.79400D-04	9.04875D-04	
180	2	3.06070D-03	3.61950D-03			
182	4	2.27420D-02	2.27420D-02	2.27420D-02	2.27420D-02	
189	4	9.04875D-04	0.00000D+00	2.79400D-04	9.04875D-04	
-1						
GEOMIN	61	2	1			
-1						
GEOMIN	61	3	2			
1	4	2.63808D-05	2.63808D-05	2.63808D-05	2.63808D-05	
8	4	1.27000D-01	1.27000D-01	5.08000D-02	5.08000D-02	
12	4	6.77335D-02	6.77335D-02	6.77335D-02	6.77335D-02	
16	4	6.77335D-02	6.77335D-02	6.77335D-02	6.77335D-02	
20	4	6.77335D-02	6.77335D-02	6.77335D-02	6.77335D-02	
24	4	6.77335D-02	6.77335D-02	6.77335D-02	5.08000D-02	
28	4	5.08000D-02	8.46668D-02	8.46668D-02	8.46668D-02	
32	4	2.85779D-03	2.85779D-03	2.85779D-03	2.85779D-03	
39	4	2.20261D-03	2.20261D-03	2.20261D-03	2.20261D-03	

46	4	2.20261D-03	2.20261D-03	2.20261D-03	2.20261D-03
53	1	1.0160			
54	1	4.41960D-03			
78	1	4.97840D-03			
102	2	4.41960D-03	4.97840D-03		
104	1	1.00000D-06			
128	1	4.41960D-03			
152	4	6.85800D-01	2.74320D+00	4.06400D-01	8.63600D-01
159	4	3.15494D-03	3.15494D-03	3.15494D-03	3.15494D-03
166	1	1.04244D-04			
169	4	1.24460D-03	5.58800D-04	2.79400D-04	1.24460D-03
180	2	4.41960D-03	4.97840D-03		
182	4	3.12802D-02	3.12802D-02	3.12802D-02	3.12802D-02
189	4	1.24460D-03	0.00000D+00	2.79400D-04	1.24460D-03
-1					
POWINC	62	1	0		
2	1	7.80000D-03			
4	2	4.80000D-03	1.14000D-02		
6	5	5.20982D-02	9.21300D-02	7.53205D-02	9.55035D-02
11	5	6.70660D-01	7.73180D-01	8.38034D-01	9.03268D-01
16	5	9.71596D-01	9.78619D-01	9.67967D-01	9.45560D-01
21	5	8.24075D-01	7.55349D-01	6.48849D-01	5.34861D-01
26	4	6.56648D-02	5.30372D-02	3.39961D-02	2.20395D-02
30	5	1.00000D+00	1.00000D+00	1.00000D+00	1.00000D+00
35	5	1.00000D+00	1.00000D+00	1.00000D+00	1.00000D+00
40	1	1.00000D+00			
45	1	1.00000D+00			
62	1	-1.96886D-04			
63	1	-1.52839D-04			
64	5	7.19610D-03	2.18275D-02	1.86557D-02	2.72274D-02
69	5	3.93763D-02	4.95811D-02	5.94463D-02	6.80290D-02
74	5	7.89058D-02	7.98959D-02	7.83367D-02	7.38466D-02
79	5	5.73876D-02	4.70589D-02	3.65141D-02	2.71253D-02
84	4	1.50466D-02	1.21726D-02	5.25961D-03	2.25946D-03
112	5	-1.05779D-07	-4.38348D-07	-1.11615D-06	-1.94766D-06
117	5	1.97624D-06	5.35145D-06	9.36494D-06	1.21464D-05
122	5	1.60731D-05	1.64355D-05	1.58951D-05	1.39977D-05
127	5	8.74852D-06	4.68446D-06	1.35091D-06	-1.53800D-06
132	4	-1.33233D-06	-6.37658D-07	-2.62388D-07	-1.23322D-07
160	5	4.31728D-08	1.91981D-07	4.22791D-07	6.63598D-07
165	5	-1.52652D-06	-3.07553D-06	-4.81760D-06	-6.04264D-06
170	5	-7.74243D-06	-7.89584D-06	-7.66379D-06	-6.85936D-06
175	5	-4.54029D-06	-2.76824D-06	-1.23071D-06	1.78688D-07
180	4	5.23145D-07	2.76646D-07	1.15633D-07	5.02132D-08
208	5	5.64802D-07	1.92023D-06	4.00704D-06	6.06383D-06
213	5	1.14597D-05	1.43700D-05	1.64604D-05	1.87583D-05
218	5	2.13181D-05	2.15990D-05	2.11767D-05	2.03284D-05
223	5	1.60134D-05	1.38565D-05	1.09304D-05	8.24123D-06
228	4	3.59488D-06	1.87123D-06	8.23201D-07	3.30770D-07
256	1	1.56065D+00			
-1					
POWINC	62	2	1		
6	5	4.35771D-02	7.51871D-02	5.96170D-02	7.50598D-02
11	5	5.50006D-01	6.32307D-01	6.87107D-01	7.39967D-01

16	5	7.96288D-01	8.01982D-01	7.93256D-01	7.73898D-01	7.31044D-01
21	5	6.74540D-01	6.14352D-01	5.24986D-01	4.28666D-01	6.20221D-02
26	4	4.66609D-02	3.66878D-02	2.28548D-02	1.43483D-02	
62	1	4.37996D-03				
63	1	3.28633D-03				
64	5	8.12885D-03	2.35243D-02	1.96334D-02	2.84407D-02	3.10794D-02
69	5	4.00741D-02	5.01402D-02	6.02054D-02	6.87556D-02	7.56602D-02
74	5	7.97466D-02	8.07460D-02	7.91636D-02	7.44852D-02	6.70303D-02
79	5	5.78794D-02	4.66719D-02	3.55109D-02	2.52942D-02	2.03811D-02
84	4	1.25077D-02	9.48347D-03	3.87932D-03	1.57900D-03	
112	5	8.09972D-08	2.91873D-07	7.25090D-07	1.29044D-06	8.50012D-07
117	5	9.97299D-07	3.05357D-06	5.50650D-06	7.21790D-06	8.59298D-06
122	5	9.63486D-06	9.85770D-06	9.52200D-06	8.35037D-06	6.88237D-06
127	5	5.08196D-06	2.49872D-06	3.92145D-07	1.40745D-06	1.71485D-06
132	4	9.73747D-07	4.40982D-07	1.69065D-07	7.27666D-08	
160	5	3.35648D-08	1.30520D-07	2.78601D-07	4.43281D-07	1.22743D-08
165	5	8.92744D-07	1.84617D-06	2.92217D-06	3.68310D-06	4.29426D-06
170	5	4.73900D-06	4.83423D-06	4.68835D-06	4.18580D-06	3.53025D-06
175	5	2.72414D-06	1.57836D-06	5.91752D-07	3.00442D-07	6.64762D-07
180	4	4.10576D-07	2.02907D-07	7.92882D-08	3.16595D-08	
208	5	4.39637D-07	1.41220D-06	2.86397D-06	4.27339D-06	6.13942D-06
213	5	8.03591D-06	1.00755D-05	1.16284D-05	1.32463D-05	1.45254D-05
218	5	1.50888D-05	1.52861D-05	1.49865D-05	1.43413D-05	1.29779D-05
223	5	1.12938D-05	9.68314D-06	7.59944D-06	5.69759D-06	3.83960D-06
228	4	2.42493D-06	1.22336D-06	5.19164D-07	2.00638D-07	
256	1	1.27188D+00				
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POW INC	62	3	2			
6	5	9.13086D-02	1.61980D-01	1.32220D-01	1.64939D-01	3.36985D-01
11	5	4.06112D-01	4.69508D-01	6.16046D-01	6.65009D-01	7.01632D-01
16	5	7.58826D-01	7.64529D-01	7.55709D-01	6.91281D-01	6.51687D-01
21	5	5.99523D-01	4.44350D-01	3.78299D-01	3.06741D-01	1.40393D-01
26	4	1.08359D-01	8.51345D-02	5.34140D-02	3.35461D-02	
62	1	4.33673D-03				
63	1	3.51787D-03				
64	5	4.29068D-03	1.25425D-02	1.05842D-02	1.53432D-02	2.98970D-02
69	5	4.04873D-02	5.18224D-02	6.43131D-02	7.40364D-02	8.18426D-02
74	5	8.70234D-02	8.82215D-02	8.63474D-02	8.04344D-02	7.20115D-02
79	5	6.16858D-02	4.83963D-02	3.64807D-02	2.54635D-02	1.18925D-02
84	4	7.53348D-03	5.86103D-03	2.45912D-03	1.03236D-03	
112	5	1.65610D-08	1.41321D-08	3.43729D-08	2.82236D-07	1.42082D-06
117	5	3.38753D-06	5.59333D-06	8.23165D-06	1.02432D-05	1.18209D-05
122	5	1.29551D-05	1.32091D-05	1.28174D-05	1.15314D-05	9.83938D-06
127	5	7.72011D-06	4.99498D-06	2.78508D-06	8.80527D-07	1.31266D-07
132	4	2.49453D-07	1.71238D-07	8.16389D-08	4.83311D-08	
160	5	9.13372D-09	2.99601D-08	3.72566D-10	1.28720D-07	6.84579D-07
165	5	1.53925D-06	2.47132D-06	3.60881D-06	4.45304D-06	5.11269D-06
170	5	5.60296D-06	5.71124D-06	5.54444D-06	4.98766D-06	4.27819D-06
175	5	3.38481D-06	2.20203D-06	1.26383D-06	4.33415D-07	6.60476D-08
180	4	1.32177D-07	1.00731D-07	4.58262D-08	2.18497D-08	
208	5	2.45097D-08	1.18560D-07	1.11624D-07	2.98340D-07	4.28112D-07
213	5	1.09961D-06	1.84085D-06	1.69641D-06	2.20503D-06	2.60023D-06
218	5	2.41073D-06	2.46843D-06	2.38055D-06	2.58185D-06	2.15114D-06
223	5	1.60372D-06	1.72699D-06	9.47050D-07	2.60121D-07	1.28314D-07

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228      4 1.76877D-08-3.66264D-08-2.36845D-09 7.72082D-09
256      1 7.24075D-01
-1
PMATCH  63      1      0
2        3 1.80000D 03 6.10000D-05 1.32000D-06
5        3 8.00000D 03 8.00000D 03 2.86000D-02
11       4 2.72000D 01 2.72000D 01 2.72000D 01 2.72000D 01
18       4 2.72000D 01 2.72000D 01 2.72000D 01 2.72000D 01
25       3 4.818 0-08 0.0          1.01325D 05
28       4 2.72000D 01 2.72000D 01 2.72000D 01 2.72000D 01
35       2 7.76000D 03 8.45000D 02
37       4 5.98560D 06 5.98560D 06 5.98560D 06 5.98560D 06
44       4 5.98560D 06 5.98560D 06 5.98560D 06 5.98560D 06
51       4 5.98560D 06 5.98560D 06 5.98560D 06 5.98560D 06
58       2 5.57000D 05 3.00000D-02
73       4 1.10000D-05 1.40000D-05 1.50000D+11 1.52056D+11
77       3 1.10000D-05 1.40000D-05 1.00000D 00
-1
PMATCH  63      2      1
-1
PMATCH  63      3      1
-1
COOLIN  64      1      0
1        5 4.08900D-01-2.50000D-01 1.44000D-02 8.60000D-01 4.48000D 00
7        2 1.50900D 03 9.90000D 01
47       1 1.26907D-01
67       2 5.00000D-01 5.00000D-01
69       5 1.50000D 01 5.00000D 01 1.00000D-01 6.30000D 04 2.00000D-02
75       3 1.00000D-07 1.00000D-07 1.42700D-04
84       1 1.42700D-04
165      1 1.00000D 01
-1
COOLIN  64      2      1
47       1 1.18135D-01
-1
COOLIN  64      3      1
47       1 1.33845D-01
-1
FUELIN  65      1      0
2        1 0.2
-1
FUELIN  65      2      1
-1
FUELIN  65      3      2
-1
INPMR4   3      1      0
1       10      7      5      0      15      5      0      0      1      0      57
11      10      1      7      8      4      4      4      4      9      4      8
21       2      8      8
49      10      1      3      2      3      3      2      3      3      2      3
59      10      3      7      3      3      5      3      3      3      3      3
69      10      5      3      3      3      3      3      3      3      6      3
79      10      3      7     11      5      3      3      6      3      3      8
89      10      3      3      3      3     12      3      3      7      3      3

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99	7	3	12	3	3	6	3	3			
189	10	1	2	1	2	1	2	1	2	2	3
199	10	3	4	4	1	4	1	3	5	5	1
209	10	5	1	2	6	3	7	2	2	2	3
219	10	8	9	9	8	9	10	11	11	12	12
269	2	8	10								
325	10	1	3	3	3	3	2	2	2	2	2
335	10	2	1	1	3	3	8	1	1	7	7
365	10	1	2	5	8	11	14	16	18	20	22
375	10	24	26	27	28	31	34	42	43	44	51
405	1	3									
406	3	15	21	34							
418	3	2	2	2							
470	1	3									
473	3	12	29	32							
481	3	37	48	55							
497	1	1									
512	1	40									
513	10	4	7	4	4	7	4	4	7	4	4
523	10	41	4	4	4	4	4	4	4	4	4
533	10	4	41	4	4	41	4	20	20	41	10
543	10	4	4	4	2	10	10	4	2	10	10
613	10	2	3	4	5	6	7	8	9	10	11
623	10	12	13	14	16	18	20	22	24	26	27
633	10	28	29	30	31	32	33	34	38	40	41
643	10	42	43	44	45	46	49	51	52	53	56
713	10	2	3	4	5	6	7	8	9	10	11
723	10	12	13	15	17	19	21	23	25	26	27
733	10	28	29	30	31	32	33	36	39	40	41
743	10	42	43	44	45	47	50	51	52	54	57
813	1	3									
814	3	7	7	7							
822	3	1	1	1							
830	3	3	6	9							
839	2	1	40								
852	1	1									
864	1	3									
973	2	1	33								
982	1	2									
991	3	2	45	52							
1008	5	4	-2	-6	-7	-3					
1052	3	-2	-3	-6							
1082	3	-6	-7	-7							
1153	2	2	2								
1156	2	1	1								
1171	1	0									
1288	2	200	20								
-1											
PMR4IN	18	1	0								
2	5	16449.7	10.3644		335.0		251.6		4475.85		
7	5	4492.3	2246.15		2246.15		4492.3		2246.15		
12	5	2246.15	0.0		0.0		63.0		65.8		
17	5	4460.4	4460.4		0.0		65.8		63.0		
42	5	-1.7018	-1.7018		-1.7018		-1.7018		8.1534		

47	5-3.048	0.1016	0.1016	-3.048	0.1016
52	5 0.1016	2.9972	1.4732	7.3914	9.29642
57	5 19.05	19.05	19.05	33.0708	33.0708
82	5 1.	1.	90.	90.	1.
87	5 1.	1.	1.	4.	4.
92	5 2.	1.	1.	2.	1.
97	5 2.	2.	1.	1.	2.
102	5 1.	2.	1.	1.	1.
107	5 1.	1.	1.	1.	1.
112	5 1.	1.	1.	1.	1.
117	5 1.	1.	1.	1.	1.
162	5 2.9972	0.0	1.7272	2.9972	0.0
167	5 1.7272	2.9972	0.0	1.7272	2.9972
172	5 8.05182	0.127	-2.7432	3.5052	6.9088
177	5-3.5052	-3.5052	-3.5052	-3.5052	3.5052
182	5 6.9088	-3.5052	-3.5052	-3.5052	-3.5052
187	5 2.8972	1.5732	7.3914	3.683	3.683
192	5 7.3914	3.683	3.583	19.05	19.05
197	5 0.127	8.05182	15.6972	15.6972	-6.9088
202	5 19.05	19.05	23.876	33.0708	31.8516
207	5 31.8516	3.683	7.3914	33.0708	33.0708
212	5 33.0708	31.8516	31.8516	3.683	7.3914
217	2 33.0708	33.0708			
302	5 4.699	1.7018	1.7272	1.27	1.7018
307	5 1.7272	1.27	1.7018	1.7272	1.27
312	5 1.885	7.92482	2.87020	6.5532	3.4036
317	5 3.6068	4.03861	3.6068	4.03861	6.5532
322	5 3.4036	3.6068	4.03861	3.6068	4.03861
327	5 0.1	0.1	0.2159	3.7084	0.2159
332	5 1.90502	3.7084	0.1	1.0	17.5970
337	5 18.9230	7.92482	7.64538	22.7076	22.6060
342	5 25.9588	6.47701	4.826	3.6069	11.2167
347	5 4.3282	28.1686	3.7084	25.6794	1.0
352	5 3.6069	11.2167	4.3282	28.1686	3.7084
357	2 25.6794	1.0			
442	5 1.0	0.00445836	0.00445836	0.00445836	1.14510
447	5 1.14510	1.14510	0.860023	0.860023	0.860023
452	5 2.7545	3.0397	2.3776	0.50278	0.8234
457	5 0.29186	0.29186	0.29186	0.29186	0.50278
462	5 0.8234	0.29186	0.29186	0.29186	0.29186
467	5 0.1000	0.1000	0.62264	0.24520	0.62264
472	5 0.62264	0.072119	0.62264	0.8234	0.62072
477	5 0.62072	4.58089	0.62072	0.62072	0.882
482	5 0.62072	0.62072	0.15518	0.11224	0.097763
487	5 0.11224	0.11224	0.072119	0.11224	0.11224
492	5 0.11224	0.097763	0.11224	0.11224	0.2452
497	2 0.11224	0.11224			
582	5 1.0	3.06025D-03	3.06025D-03	3.06025D-03	0.0025
587	5 0.0025	0.0025	0.5	0.5	0.5
592	5 0.4572	0.026289	1.7399	0.8001	0.3302
597	5 0.6096	0.6096	0.6096	0.6096	0.8001
602	5 0.3302	0.6096	0.6096	0.6096	0.6096
607	5 0.35682	0.35682	0.53045	0.04458	0.53045
612	5 0.53045	0.01732	0.53045	0.3302	0.889

617	5	0.889	0.03502	0.889	0.889	0.03502
622	5	0.889	0.889	0.4445	0.37803	0.04445
627	5	0.37803	0.37803	0.017323	0.37803	0.37803
632	5	0.37803	0.04445	0.37803	0.37803	0.04458
637	2	0.37803	0.37803			
722	5	0.00001	0.00001	0.00001	0.00001	0.00001
727	5	0.00001	0.00001	0.00001	0.00001	0.00001
732	5	0.00001	0.00001	0.00001	0.00001	0.00001
737	5	0.00001	0.00001	0.00001	0.00001	0.00001
742	5	0.00001	0.00001	0.00001	0.00001	0.00001
747	5	0.00001	0.00001	0.00001	0.00001	0.00001
752	5	0.00001	0.00001	0.00001	0.00001	0.00001
757	5	0.00001	0.00001	0.00001	0.00001	0.00001
762	5	0.00001	0.00001	0.00001	0.00001	0.00001
767	5	0.00001	0.00001	0.00001	0.00001	0.00001
772	5	0.00001	0.00001	0.00001	0.00001	0.00001
777	2	0.00001	0.00001			
862	5	0.0	0.0	0.0	0.0	0.0
867	5	0.0	0.0	0.0	0.0	0.0
872	5	0.0	0.0	0.0	0.0	0.0
877	5	1.0	1.0	1.0	1.0	0.0
882	5	0.0	1.0	1.0	1.0	1.0
887	5	0.0	0.0	0.0	0.0	0.0
892	5	0.0	0.0	0.0	0.0	3.0
897	5	0.0	0.0	0.0	3.0	0.0
902	5	3.0	0.0	1.0	0.0	6.0
907	5	1.0	0.0	0.0	1.0	0.0
912	5	0.0	6.0	1.0	0.0	0.0
917	2	1.0	0.0			
1013	1	7.25840D+00				
1017	1	1.63623D 00				
1019	1	1.63623D 00				
1023	1	1.63623D 00				
1025	1	1.63623D 00				
1038	1	4.36618D+01				
1142	1	12.0				
1143	5	1.	7.71770D+04	7.71770D+04	7.71770D+04	2.71657D+07
1148	5	2.71657D+07	2.71657D+07	1.00000D+05	1.00000D+05	1.00000D+05
1153	5	7.63334D+06	2.52430D+06	4.95536D+06	2.91088D+05	3.60030D+06
1158	5	5.31795D+04	5.31795D+04	5.31795D+04	5.31795D+04	2.91088D+05
1163	5	3.60030D+06	5.31795D+04	5.31795D+04	5.31795D+04	5.31795D+04
1168	5	1.0	1.0	2.50694D+05	2.50694D+05	2.50694D+05
1173	5	2.50694D+05	6.67180D+04	2.50694D+05	3.60030D+06	1.58978D+05
1178	5	1.58978D+05	2.50783D+06	1.58978D+05	1.58978D+05	2.11374D+06
1183	5	1.58978D+05	1.58978D+05	7.83699D+04	8.05690D+04	7.72335D+04
1188	5	8.05690D+04	8.05690D+04	6.67181D+04	8.05690D+04	8.05690D+04
1193	5	8.05690D+04	7.72335D+04	8.05690D+04	8.05690D+04	2.50694D+05
1198	2	8.05690D+04	8.05690D+04			
1283	5	1.	5.68800D+03	5.68800D+03	5.68800D+03	5.0 D+03
1288	5	5.0 D+03	5.0 D+03	5.0 D+03	5.0 D+03	5.0 D+03
1293	5	1.11450D+03	2.47769D+04	1.48670D+03	1.11450D+03	1.34750D+02
1298	5	4.45980D+03	4.45980D+03	4.45980D+03	4.45980D+03	1.11450D+03
1303	5	1.34750D+02	4.45980D+03	4.45980D+03	4.45980D+03	4.45980D+03
1308	5	1.0 D+05	1.0 D+05	1.23880D+03	1.23880D+03	1.23880D+03

1313	5	1.23880D+03	3.27930D+04	1.23880D+03	1.34750D+02	2.22990D+03
1318	5	2.22990D+03	5.57480D+02	2.22990D+03	2.22990D+03	5.57480D+02
1323	5	2.22990D+03	2.22990D+03	2.22990D+03	1.99635D+03	7.43310D+03
1328	5	1.99635D+03	1.99635D+03	3.27930D+04	1.99635D+03	1.99635D+03
1333	5	1.99635D+03	7.43310D+03	1.99635D+03	1.99635D+03	1.23880D+03
1338	2	1.99635D+03	1.99635D+03			
1423	5	61.126	2497.5	2664.6	6.0	6.0
1428	5	140.5	140.5	3.67	1.0	30.395
1433	2	20.0	20.0			
1461	5	0.0	1.01325D+05	1.01325D+05	0.0	0.0
1466	5	0.0	0.0	2.73646D+05	0.0	2.73646D+05
1471	2	1.01325D+05	1.01325D+05			
1499	5	2.0 D-08	2.0 D-08	2.0 D-08	2.0 D-08	2.0 D-08
1504	5	2.0 D-08	2.0 D-08	2.0 D-08	2.0 D-08	2.0 D-08
1509	2	2.0 D-08	2.0 D-08			
1537	5	6.00000D-05	6.00000D-05	6.00000D-05	6.00000D-05	6.00000D-05
1542	5	6.00000D-05	6.00000D-05	6.00000D-05	6.00000D-05	6.00000D-05
1547	2	6.00000D-05	6.00000D-05			
1575	5	-1.7018	2.9972	-3.048	0.1016	0.1016
1580	5	2.6162	1.8542	19.05	19.05	23.876
1585	2	33.0708	33.0708			
1613	5	1.0	202.3	16.7	1.0	1.0
1618	5	1.0	1.0	0.5	1.0	12.697
1623	2	4.0	4.0			
1651	5	0.0	758.3	758.3	0.0	0.0
1689	4	1.6	2078.	1.94 D-05	273.	
1693	1	1.828				
1721	1	1.78000D-03				
1749	2	1.	1.			
1805	1	4.76250D-02				
1833	1	1.00000D-05				
1861	5	0.0	1.00000D+08	1.00000D+08	0.0	0.0
1866	5	0.0	0.0	1.00000D+08	0.0	1.00000D+08
1871	2	1.00000D+08	1.00000D+08			
1899	3	18650.0	18650.0	400.0		
1911	3	7.58423D+05	7.58423D+05	2.42338D+05		
1923	3	51.84	51.84	116.76		
1935	3	4492.3	4492.3	4460.4		
1947	3	0.689	0.689	0.83		
1983	5	0.01	0.0	-73.13	0.00268	0.0
1988	5	0.07	0.00383	0.01071	0.01406	0.01
1993	1	0.268				
2000	3	1.0	0.03	0.03		
2003	5	0.01	0.0	-73.13	0.00268	0.0
2008	5	0.07	0.00383	0.01071	0.01406	0.01
2013	1	0.268				
2020	3	1.0	0.03	0.03		
2023	5	0.01	0.0	-73.13	0.00268	0.0
2028	5	0.07	0.00383	0.01071	0.01406	0.01
2033	1	0.268				
2040	3	1.0	0.03	0.03		
2223	2	1.0	1.0			
2233	2	0.0	10000.			
2243	2	1.0	1.0			

2253	2	0.0	10000.						
2263	2	1.0	1.0						
2273	2	0.0	10000.						
2463	1	9.80700D+00							
2464	5	2.13 D-10	2.13 D-10	2.13 D-10	2.13 D-10	2.13 D-10	2.13 D-10	2.13 D-10	2.13 D-10
2469	5	2.13 D-10	2.13 D-10	2.13 D-10	2.13 D-10	2.13 D-10	2.13 D-10	2.13 D-10	2.13 D-10
2474	5	2.13 D-10	2.13 D-10						
2578	5	282.2	282.2	282.2		431.2		431.2	
2583	5	423.2	423.2	431.2		50.0		557.5	
2588	2	1128.6	1128.6						
2616	5	67.56	374.0	617.6		23.086		23.086	
2621	5	251.1	251.1	6.7827		1.0		27.24	
2626	2	1.0	1.0						
2654	5	2.27493D+07	1.25940D+08	2.38580D+08		5.18247D+06		5.18247D+06	
2659	5	5.63681D+07	5.63681D+07	1.52262D+06		2.80606D+04		6.33777D+06	
2664	2	2.80606D+04	2.80606D+04						
2694	3	0.025	0.80	4.80					
2697	3	4.7	26.2	26.2					
2705	3	4.7	26.2	26.2					
2713	3	26.2	26.2	26.2					
2721	3	26.2	26.2	26.2					
2729	3	26.2	26.2	26.2					
2737	3	2.07845D-03	1.38584D-03	5.0 D-04					
2745	3	5.588 D-04	1.38584D-03	5.0 D-04					
2753	3	3.810 D-03	3.810 D-03	5.0 D-04					
2761	3	2.9 D+06	4.709 D+06	4.709 D+06					
2769	3	2.9 D+06	4.709 D+06	4.709 D+06					
2777	3	4.709 D+06	4.709 D+06	4.709 D+06					
2785	3	4.709 D+06	4.709 D+06	4.709 D+06					
2793	3	4.709 D+06	4.709 D+06	4.709 D+06					
2801	5	1.	1.	1.		1.		1.	
2806	5	1.	1.	1.		1.		1.	
2811	5	1.	1.	1.		1.		1.	
2816	5	1.	1.	1.		1.		1.	
2821	1	1.							
2857	5	3.51967D-04	0.	0.		0.		0.	
2862	4	0.	0.	0.		0.			
2881	5	0.	0.	0.		0.		0.	
2886	4	0.	0.	0.		0.			
2913	3	0.	0.	0.					
2921	3	4.98955	1632.37	1.00					
2929	3	0.49546	169.45	0.10					
2937	2	1.0	1.0						
2951	2	0.0	0.0						
3105	2	4.3942	4.3942						
3273	2	0.0	10000.						
3287	2	0.0	10000.						
3441	1	0.9999							
3445	3	7.0300 D+05	3.50000D+05	3.50000D+05					
3455	5	0.5	1.	1.		1.		1.	
3460	5	1.	1.	1.		1.		1.	
3465	5	1.	1.	1.		1.		1.	
3470	5	1.	1.	1.		1.		1.	
3475	5	1.	1.	1.		1.		1.	


```

3480      5 1.          1.          1.          1.          1.
3485      5 1.          1.          1.          1.          1.
3490      5 1.          1.          1.          1.          1.
3495      1 0.5
3516      3 14.73165    3.63736    3.63736
3524      3 502.7       18.31328    18.31328
3532      3 462.5       16.653     16.653
3540      3 0.0508      0.02285    0.02285
3548      3 1.14300D-03 8.63600D-04 8.63600D-04
3556      3 4.58       D+06 4.41900D+06 4.41900D+06
3564      3 4.58       D+06 4.41900D+06 4.41900D+06
3572      3 28.32      21.5       21.5
3580      3 28.32      21.5       21.5
3588      3 1.0        D+09 1.0        D+09 1.0        D+09
3596      3 1.0        D+09 1.0        D+09 1.0        D+09
3604      3 1.0        1.0         1.0
3612      5 0.0        1184.98    1035.63    0.0        0.0
3617      5 0.0        0.0         1.0         0.0        15.197
3622      2 10.0       10.0
3650      5 0.025      0.80         4.80         0.025      0.80
3655      1 4.80
3801      2 2.22250D-02 2.22250D-02
3805      2 2.41300D-02 2.41300D-02
3809      2 21.5       21.5
3813      2 3.73000D+01 3.73000D+01
3817      2 8.6        D+04 8.6        D+04
3821      2 3059.16    3059.16
3825      2 29.3878    29.3878
3829      2 0.84       0.84
3833      2 0.01       0.01
3837      2 0.0        0.0
3841      2 17.5958    17.5958
3845      2 29.2335    29.2335
3849      2 18.2709    18.2709
3853      2 143.6      143.6
3863      2 130.8      130.8
3893      2 0.0        10000.
3903      2 0.0        10000.
3933      1 300.
4174      4 1.25980D-04 1.52400D-05 1.52400D-05 2.03200D-06
4216      3 77443.6    16615.3    77443.6
4285      2 168.0      168.0
-1
ENDJOB    -1

```

II. DEFORM-4 Input Data

The models in DEFORM-4 are used to provide a much more detailed representation of the oxide fuel and cladding than is possible with the standard 'simple' models in SAS4A. The development of this input data is discussed in Appendix E. If the DEFORM-4 models are to be used, the following data is inserted in the input data deck in Section I just prior to the ENDJOB card:

XNOTES

DEFORM-4 USED FOR FUEL CHARACTERIZATION

```

-1
INPCOM      1      1      1
  26        1      3
  28        1      1
-1
PMATCHM     13      1      1
  171       5 1.09387D 04 1.08569D 04 1.07596D 04 1.06470D 04 1.05194D 04
  176       5 1.03774D 04 1.02211D 04 1.00510D 04 9.86748D 03 9.67080D 03
  181       3 9.63883D 03 8.86000D 03 8.74092D 03
  600       4 1.31000D+02 2.46000D-01 1.97000D+02 6.00000D-01
  604       2 1.20000D+00 2.00000D-02
  910       4 8.48000D+08 7.58000D+08 7.14000D+08 6.96000D+08
  914       4 6.76000D+08 6.36000D+08 4.62000D+08 2.62000D+08
  918       3 1.38000D+08 3.45000D+07 0.00000
  970       4 2.55350D+02 3.66450D+02 4.77550D+02 5.88750D+02
  974       4 6.99850D+02 8.10950D+02 9.22050D+02 1.03315D+03
  978       3 1.14425D+03 1.25535D+03 1.64815D+03
  1073      5 0.132      0.175      0.132      0.175      0.067
  1077      1 0.043
  1083      4 2.07040D+01 4.52810D+05 1.50000D+00 8.31434D+00
  1087      4 4.50000D-01 5.00000D+04 5.65065D+04 1.71700D+10
  1091      4 3.87000D+05 1.45556D-08 2.67000D+05 2.23000D-03
  1095      4 6.33750D+04 9.46900D+02 3.13000D+03 3.30000D-06
  1099      1 1.78000D-06
  1101      1 1.66667D+00
  1103      4 1.00000D+00 1.50000D-01 8.05000D-01 1.00000D+00
  1107      4 5.67000D-08 9.00000D-01 8.00000D-01 6.92000D+03
  1111      4 3.39500D+01 3.38000D-01 1.48000D+04 9.57500D+00
  1115      4 2.00000D-04 1.10000D+04 3.00000D-01 3.26500D-01
  1119      2 1.00000D+00 3.77000D+05
  1124      1 6.50000D+01
  1225      2 4.00000D+00 3.00000D-03
  1258      2 1.00000D+00 2.00000D-01
  1261      3 1.00000D-01 6.00000D-01 1.00000D+00
  1275      1 1.00000D-01
-1
INPCHN      51      1      1
   3        1      3
  15        2      1      3
  20        5      5      4      -2      -2      1
  31        2      4      1
  37        4      10     30     20     50
  51        4      99     99     99     30
  95        1      3
  122       3      1      0      1
  181       1      0
-1
INPCHN      51      2      2
   3        1      3
  15        2      1      3
  20        5      5      4      -2      -2      1
  31        2      4      1

```

37	4	10	30	20	50	
51	4	99	99	99	30	
95	1	3				
122	3	1	0	1		
181	1	0				
-1						
INPCHN	51	3	3			
3	1	3				
15	2	5	5			
20	5	5	4	-2	-2	1
31	2	4	1			
37	4	10	30	20	50	
51	4	99	99	99	30	
95	1	3				
122	3	1	0	1		
181	1	0				
-1						
GEOMIN	61	1	1			
128	1	2.98392D-03				
-1						
GEOMIN	61	2	2			
128	1	2.98392D-03				
-1						
GEOMIN	61	3	3			
128	1	4.36383D-03				
-1						
POWINC	62	1	1			
45	4	1.00000D+00	1.00000D+00	1.00000D+00	1.00000D+00	
53	4	8.64000D+05	3.45600D+06	8.64000D+06	5.04576D+07	
61	1	8.50624D+12				
-1						
POWINC	62	2	2			
45	4	1.00000D+00	1.00000D+00	1.00000D+00	1.00000D+00	
53	4	8.64000D+05	3.45600D+06	8.64000D+06	5.04576D+07	
61	1	8.50624D+12				
-1						
POWINC	62	3	3			
45	4	1.00000D+00	1.00000D+00	1.00000D+00	1.00000D+00	
53	4	8.64000D+05	3.45600D+06	8.64000D+06	5.04576D+07	
61	1	8.50624D+12				
-1						
PMATCH	63	1	1			
5	2	3.00000D+04	1.00000D+02			
26	2	1.00000D-05	1.01325D+05			
60	1	3.00000D-01				
69	2	2.70000D+03	1.00000D-08			
-1						
PMATCH	63	2	2			
5	2	3.00000D+04	1.00000D+02			
26	2	1.00000D-05	1.01325D+05			
60	1	3.00000D-01				
69	2	2.70000D+03	1.00000D-08			
-1						

```

PMATCH    63      3      3
          5      2 3.00000D+04 1.00000D+02
          26      2 1.00000D-05 1.01325D+05
          60      1 3.00000D-01
          69      2 2.70000D+03 1.00000D-08
          -1

```

III. Input Data for the Transients

The calculation of any particular transient requires very little additional input data. The data generally describes the behavior of the major reactor systems, such as a pump trip, steam generator trip, or control rod runout. The data for the three transients discussed in this report are given in the following three sections.

A. Unprotected Loss-of-Flow Accident

In addition to the data in Sections I and II, the loss-of-flow calculation requires a pump trip and a loss of the steam generator heat sink. To run this calculation, the following input data is included with the previous data, just prior to the ENDJOB card.

```

XNOTES      0      0      0
*****
*                                                    *
* Input required for LOF calculation.                *
*                                                    *
*****
-1
INPCOM      1      1      1
  11      3    400    20    20
-1
OPCIN      11      1      1
  7      1 6.00000D+01
-1
PMR4IN      18      1      1
  2223    2 0.0          0.0
  2233    2 0.          10000.0
  2243    2 0 0          0.0
  2253    2 0.0          10000.0
  2263    2 0.0          0.0
  2273    2 0.0          10000.0
  2937    3 1.0          0.0          0.0
  3105    3 4.3942       4.3942       4.3942
  3273    3 0.0          20.0          10000.0
-1

```

B. Unprotected Transient Overpower Accident

In addition to the input data in Sections I and II, the transient overpower accident requires a control rod withdrawal. This is simulated by a reactivity insertion table in the input data. All other conditions remain the same. To run this calculation, the following data is included with the data in Sections I and II, just prior to the ENDJOB card.

```

XNOTES      0      0      0
*****
*
*   Input required for TOP calculation.
*
*****
-1
INPCOM      1      1      1
  11      3  1200    50    50
  18      1      3
-1
OPCIN      11      1      1
   5      1 5.00000D-01
   7      1 3.00000D+02
-1
PMR4IN      18      1      1
 2223      2 1.0          1.0
 2233      2 0.          10000.0
 2243      2 1.0          1.0
 2253      2 0.0          10000.0
 2263      2 1.0          1.0
 2273      2 0.0          10000.0
 2937      3 1.0          1.0          1.0
 3105      3 4.3942        4.3942        4.3942
 3273      3 0.0          20.0          10000.0
-1
POWINA      12      1      1
  29      3 0.0          0.10          0.10
  49      3 0.0          20.0          10000.0
-1

```

C. Unprotected Loss-of-Heat-Sink Accident

In addition to the input data in Sections I, the loss-of-heat-sink accident requires a loss of heat rejection capability at the steam generator. All other conditions remain the same. DEFORM-4 is not used for this calculation due to the long-term nature of the accident and the computational overhead associated with DEFORM-4, and its lack of modelling of HT-9 cladding. To run this calculation, the following data is included with the data in Section I, just prior to the ENDJOB card.

```

XNOTES      0      0      0
*****
*
*   Input required for LOHS calculation.
*
*****
-1
INPCOM      1      1      1
  11      3  3000    300    300
-1
OPCIN      11      1      1
   5      1 5.00000D-01
   7      1 1.50000D+03
-1

```

PMR4IN	18	1	1	
2223	2	1.0		1.0
2233	2	0.		10000.0
2243	2	1.0		1.0
2253	2	0.0		10000.0
2263	2	1.0		1.0
2273	2	0.0		10000.0
2937	3	1.0	0.0	0.0
3105	3	4.3942	4.3942	4.3942
3273	3	0.0	20.0	10000.0
-1				

This case also requires a restart to continue the calculation to 3000 seconds of total problem time. This is accomplished with the following dataset, which is the only input data used for the restart. The remainder of the data is already available in the restart file written by the previous calculation.

3500 MWE OXIDE FUEL CORE, LSPB PLANT, ANL/EUROPEAN COMPARISON STUDY
3 CHANNEL DECK FOR ATWS TRANSIENTS, ARGONNE NATL LAB., 4/17/89

3	0	0	0	0
RESTART	100	0	0	
INPCOM	1	1	1	
11	1	6000		
-1				
OPCIN	11	1	1	
7	1	3000.0		
-1				
ENDJOB	-1			

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APPENDIX G. SAS4A INPUT DATA LISTING FOR THE METALLIC FUEL CORE

This appendix contains the listing of the SAS4A input data for the metallic fuel core. The first section covers the basic data required for calculating the steady-state conditions. The input data for running the calculations on the various transients is included in the second section.

I. Steady-State Input Data

The steady-state input data contains all the data required for initializing the core conditions prior to any particular transient. The data also include values for models which are not used in the steady-state calculation, but which are identical for all accidents. The generation of this input data is discussed in detail in Appendix B for the core modelling, Appendix C for the radial core expansion reactivity feedback model, and Appendix D for the modelling of the remainder of the reactor plant. The resulting input data listing is as follows:

3500 MWE METALLIC FUEL CORE, LSPB PLANT, ANL/EUROPEAN COMPARISON STUDY
3 CHANNEL DECK FOR ATWS TRANSIENTS, ARGONNE NATL LAB., 4/17/89

3 0 0 0

XNOTES

THREE CHANNEL METAL FUEL DECK, FIVE FUEL TYPES, ONE CLADDING
TYPE. SIX DELAYED NEUTRON PRECURSOR GROUPS, SIX DECAY HEAT
GROUPS. PRIMAR-4 REPRESENTATION OF THE REACTOR SYSTEM.
DETAILED RADIAL CORE EXPANSION REACTIVITY FEEDBACK MODEL.

-1

INPCOM	1	1	0								
1	9	3	0	5	1	0	0	0	0	0	
11	10	0	0	0	0	100	6	6	0	0	0
21	9	0	1	0	11	0	0	4	0	0	
31	1	2									
36	4	-4	0	0	-5						
45	2	1	1								
51	6	595	1	2	2	0	1				
59	7	991	1	0	1	2	2	1			

-1

OPCIN	11	1	0
1	1	1.00000D-01	
3	1	1.00000D-05	
5	2	2.50000D-01	1.00000D-02
9	3	1.00000D+01	5.00000D+01 3.00000D+01
26	2	20.	0.001

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XNOTES

NEUTRONICS FOR METAL CORE DESIGN AS GIVEN IN DATA PACKAGE
SAS CHANNELS REPRESENT DRIVERS AND INTERNAL BLANKETS
DECAY HEAT REPEATED FROM EARLIER DESIGN
CORE LOAD PAD DIMENSIONS RESULT IN GRID PLATE PITCH EQUALLING
THE ABOVE-CORE LOAD PAD PITCH AT NOMINAL OPERATING CONDITIONS
CORE RESTRAINT CLEARANCE FOR DEFINED CORE GEOMETRY FOR P/F GT .7

-1

POWINA	12	1	0				
	2	2	3.19500D-07	3.50000D+09			
	4	4	7.81930D-05	7.27030D-04	6.42330D-04	1.27560D-03	
	8	2	6.00940D-04	1.91170D-04			
	10	4	1.29920D-02	3.14110D-02	1.35590D-01	3.47520D-01	
	14	2	1.38070D+00	3.79190D+00			
	69	2	9.66367D-01	9.12206D-01			
	71	4	6.29920D+00	2.00000D-05	-1.16000D+01	0.00000D+00	
	75	3	5.87927D+04	2.41470D+03	2.62467D+01		
	260	5	2.20054D-02	1.90590D-02	9.93659D-03	5.25396D-03	1.93225D-03
	265	1	1.67409D-03				
	290	5	6.56251D-02	1.77911D-03	1.37667D-04	6.08268D-06	2.25624D-07
	295	1	1.89684D-08				
	320	1	6.22500D-02				
	325	1	1.0				
	365	1	2.52288E+07				
	408	4	1.00000D+00	1.54827D-01	1.54887D-01	1.54887D-01	
	412	2	-7.65000D+01	1.20000D-03			
	414	2	1.00000D-03	1.00000D-03			
	423	2	1.00000D-01	3.50000D-01			
	426	1	1.62400D-03				
	428	1	5.00000D+02				

-1
XNOTES

FUEL PROPERTY TABLES

FUEL DENSITY TABLES

TABLE 1: U-15PU-10ZR, THEORETICAL DENSITY
 TABLE 2: U-15PU-10ZR, FUEL PIN SMEAR DENSITY (75.0%)
 TABLE 3: U-10ZR, THEORETICAL DENSITY
 TABLE 4: U-10ZR, SMEAR DENSITY (85.0%)
 TABLE 5: NA, THEORETICAL DENSITY

THERMAL CONDUCTIVITY TABLES

TABLE 1: U-15PU-10ZR, THEORETICAL VALUE.
 TABLE 2: U-15PU-10ZR, THEORETICAL VALUE REDUCED TO 72.8% FOR 25% AVERAGE FUEL POROSITY AND 25% SODIUM LOGGING.
 TABLE 3: U-10ZR, THEORETICAL VALUE.
 TABLE 4: U-10ZR, THEORETICAL VALUE REDUCED TO 78.5% FOR 15% AVERAGE FUEL POROSITY AND 25% SODIUM LOGGING.
 TABLE 5: SODIUM, THEORETICAL VALUE.

HEAT CAPACITY TABLES

TABLES 1-2: HEAT CAPACITY FOR U-15PU-10ZR
 TABLES 3-4: HEAT CAPACITY FOR U-10ZR
 TABLE 5: HEAT CAPACITY FOR SODIUM

SOLIDUS AND LIQUIDUS TEMPERATURES, HEAT OF FUSION

VALUES 1-2: U-15PU-10ZR
 VALUES 3-4: U-10ZR
 VALUE 5: SODIUM

CLADDING TABLES

TABLE 1, HT-9

-1

PMATCM	13	1	0			
11	5	2.45000D+01	2.47000D+01	2.49000D+01	2.52000D+01	2.54000D+01
16	5	2.56000D+01	2.59000D+01	2.61000D+01	2.63000D+01	2.65000D+01
21	5	2.68000D+01	2.70000D+01	2.72000D+01	2.74000D+01	2.76000D+01
26	3	2.78000D+01	2.80000D+01	2.80000D+01		
71	5	2.94270D+02	3.23000D+02	3.73000D+02	4.23000D+02	4.73000D+02
76	5	5.23000D+02	5.73000D+02	6.23000D+02	6.73000D+02	7.23000D+02
81	5	7.73000D+02	8.23000D+02	8.73000D+02	9.23000D+02	9.73000D+02
86	3	1.02300D+03	1.07300D+03	1.08872D+03		
91	5	1.58000D+04	1.57200D+04	1.56300D+04	1.55500D+04	1.54700D+04
96	5	1.53900D+04	1.53300D+04	1.51700D+04	1.51100D+04	1.50200D+04
101	3	1.49300D+04	1.48500D+04	1.47800D+04		
111	5	1.18500D+04	1.17900D+04	1.17230D+04	1.16630D+04	1.16030D+04
116	5	1.15430D+04	1.14980D+04	1.13780D+04	1.13330D+04	1.12650D+04
121	3	1.11980D+04	1.11380D+04	1.10850D+04		
131	5	1.60200D+04	1.59500D+04	1.58900D+04	1.58100D+04	1.57300D+04
136	5	1.56400D+04	1.55400D+04	1.52200D+04	1.51200D+04	1.50200D+04
151	5	1.36170D+04	1.35580D+04	1.35070D+04	1.34390D+04	1.33710D+04
156	5	1.32940D+04	1.32090D+04	1.29370D+04	1.28520D+04	1.27670D+04
171	5	8.83243D+02	8.60528D+02	8.37592D+02	8.14434D+02	7.91056D+02
176	5	7.67457D+02	7.43637D+02	7.19596D+02	6.95334D+02	6.70851D+02
251	5	2.93000D+02	4.00000D+02	5.00000D+02	6.00000D+02	7.00000D+02
256	5	8.00000D+02	8.68000D+02	9.38000D+02	1.00000D+03	1.10000D+03
261	3	1.20000D+03	1.30000D+03	1.37800D+03		
271	5	2.93000D+02	4.00000D+02	5.00000D+02	6.00000D+02	7.00000D+02
276	5	8.00000D+02	8.68000D+02	9.38000D+02	1.00000D+03	1.10000D+03
281	3	1.20000D+03	1.30000D+03	1.37800D+03		
291	5	2.93000D+02	4.00000D+02	5.00000D+02	6.00000D+02	7.00000D+02
296	5	8.00000D+02	9.00000D+02	1.00000D+03	1.10000D+03	1.20000D+03
311	5	2.93000D+02	4.00000D+02	5.00000D+02	6.00000D+02	7.00000D+02
316	5	8.00000D+02	9.00000D+02	1.00000D+03	1.10000D+03	1.20000D+03
331	5	5.50000D+02	6.50000D+02	7.50000D+02	8.50000D+02	9.50000D+02
336	5	1.05000D+03	1.15000D+03	1.25000D+03	1.35000D+03	1.45000D+03
419	1	3.00000D 02				
420	5	9.80000D+00	1.14000D+01	1.34000D+01	1.55000D+01	1.77000D+01
425	5	2.01000D+01	2.26000D+01	2.53000D+01	2.78000D+01	3.06000D+01
440	5	7.13000D+00	8.30000D+00	9.76000D+00	1.12800D+01	1.28900D+01
445	5	1.46300D+01	1.64500D+01	1.84200D+01	2.02400D+01	2.22800D+01
460	5	1.64000D+01	1.80000D+01	2.01000D+01	2.24000D+01	2.49000D+01
465	5	2.76000D+01	3.05000D+01	3.36000D+01	3.68000D+01	4.03000D+01
480	5	1.28700D+01	1.41300D+01	1.57800D+01	1.75800D+01	1.95500D+01
485	5	2.16700D+01	2.39400D+01	2.63800D+01	2.88900D+01	3.16400D+01
500	5	9.26300D+01	8.81800D+01	8.28400D+01	7.77400D+01	7.28700D+01
505	5	6.82000D+01	6.37300D+01	5.94400D+01	5.53100D+01	5.13300D+01
580	5	2.93000D+02	3.73000D+02	4.73000D+02	5.73000D+02	6.73000D+02
585	5	7.73000D+02	8.73000D+02	9.73000D+02	1.07300D+03	1.17300D+03
606	5	1.33040D+02	1.46200D+02	1.58870D+02	1.72020D+02	1.85180D+02
611	5	1.97850D+02	2.07600D+02	1.91360D+02	1.59840D+02	1.69590D+02
616	5	1.82260D+02	1.94930D+02	2.08080D+02	2.17830D+02	2.17680D+02
621	5	2.17080D+02	2.16370D+02	2.16370D+02	2.16370D+02	2.16370D+02
626	5	1.33040D+02	1.46200D+02	1.58870D+02	1.72020D+02	1.85180D+02
631	5	1.97850D+02	2.07600D+02	1.91360D+02	1.59840D+02	1.69590D+02
636	5	1.82260D+02	1.94930D+02	2.08080D+02	2.17830D+02	2.17680D+02
641	5	2.17080D+02	2.16370D+02	2.16370D+02	2.16370D+02	2.16370D+02

646	5	1.50700D+02	1.53140D+02	1.69720D+02	1.91180D+02	2.16050D+02
651	5	2.41900D+02	2.61290D+02	2.65800D+02	2.39760D+02	1.78990D+02
656	5	1.78990D+02	1.78990D+02	1.78990D+02	1.78990D+02	1.78990D+02
661	5	1.78990D+02	2.03620D+02	2.06540D+02	2.20440D+02	2.20440D+02
666	5	1.50700D+02	1.53140D+02	1.69720D+02	1.91180D+02	2.16050D+02
671	5	2.41900D+02	2.61290D+02	2.65800D+02	2.39760D+02	1.78990D+02
676	5	1.78990D+02	1.78990D+02	1.78990D+02	1.78990D+02	1.78990D+02
681	5	1.78990D+02	2.03620D+02	2.06540D+02	2.20440D+02	2.20440D+02
686	5	1.28241D+03	1.28241D+03	1.28241D+03	1.28241D+03	1.27242D+03
691	5	1.26575D+03	1.26309D+03	1.26247D+03	1.26240D+03	1.26322D+03
696	5	1.26795D+03	1.27721D+03	1.29159D+03	1.30707D+03	1.31186D+03
701	5	1.33514D+03	1.37016D+03	1.37499D+03	1.40013D+03	1.42181D+03
766	5	2.98000D+02	4.00000D+02	5.00000D+02	6.00000D+02	7.00000D+02
771	5	8.00000D+02	8.73000D+02	8.90000D+02	9.23000D+02	1.00000D+03
776	5	1.10000D+03	1.20000D+03	1.30000D+03	1.37900D+03	1.40000D+03
781	5	1.48700D+03	1.58800D+03	1.60000D+03	1.65700D+03	1.70000D+03
786	5	1.37900D+03	1.37900D+03	1.48700D+03	1.48700D+03	3.70000D+02
794	5	1.58800D+03	1.58800D+03	1.65700D+03	1.65700D+03	3.80000D+02
802	5	9.16000D+04	9.16000D+04	9.02000D+04	9.02000D+04	1.13000D+05
810	1	1.69015D+03				
813	1	1.71015D+03				
816	1	2.68300D+05				
819	5	4.82500D+02	5.21480D+02	5.63370D+02	6.15380D+02	6.84680D+02
824	5	7.84960D+02	8.66080D+02	1.02150D+03	1.21000D+03	8.68340D+02
829	2	7.47970D+02	7.62140D+02			
879	5	3.73000D+02	4.73000D+02	5.73000D+02	6.73000D+02	7.73000D+02
884	5	8.73000D+02	9.23000D+02	9.73000D+02	9.98000D+02	1.02300D+03
889	2	1.07300D+03	1.08856D+03			
990	5	3.73566D+06	4.02379D+06	4.33085D+06	4.71178D+06	5.22021D+06
995	5	5.95863D+06	6.55969D+06	7.71948D+06	9.13369D+06	6.54728D+06
1000	2	5.62698D+06	5.72962D+06			
1050	5	3.73000D+02	4.73000D+02	5.73000D+02	6.73000D+02	7.73000D+02
1055	5	8.73000D+02	9.23000D+02	9.73000D+02	9.98000D+02	1.02300D+03
1060	2	1.07300D+03	1.08856D+03			
1086	1	8.31434D+00				

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XNOTES

COVER GAS ASSUMED TO BE AT ATMOSPHERIC PRESSURE AT OPERATION

-1

PRIMIN	14	1	0
1	1	1.52599D 05	
45	1	6.30150D 02	
65	1	0.	
85	1	7.83000D 02	
87	2	-1.4986	3.20041
90	2	0.1	0.25

-1

XNOTES

CHANNEL 1 - PEAK DRIVER ASSEMBLY, MAX. POWER, NEAR MAX. POWER-TO-FLOW
 CHANNEL 2 - AVERAGE OF THE REMAINING DRIVER ASSEMBLIES
 CHANNEL 3 - AVERAGE OF THE INTERNAL BLANKET ASSEMBLIES

-1

INPCHN	51	1	0				
4	3	6	3	1			
7	5	2	2	3	30	4	
14	4	11	2	5	1		
25	4	271	6	2	2		
90	5	21	28	28	28	21	
118	1	1					
181	1	1					
203	1	1					
-1							
INPCHN	51	2	1				
25	2	271	390				
-1							
INPCHN	51	3	2				
15	2	4	5				
25	2	169	163				
-1							
GEOMIN	61	1	0				
1	5	2.29993D-05	2.29993D-05	2.29993D-05	2.29993D-05	2.29993D-05	
8	4	5.08000D-02	5.08000D-02	4.57200D-02	4.57200D-02		
12	4	4.57200D-02	4.57200D-02	4.57200D-02	4.57200D-02		
16	4	4.57200D-02	4.57200D-02	4.57200D-02	4.57200D-02		
20	4	4.57200D-02	4.57200D-02	4.57200D-02	4.57200D-02		
24	4	4.57200D-02	4.57200D-02	4.57200D-02	4.57200D-02		
28	4	4.57200D-02	4.57200D-02	5.08000D-02	5.08000D-02		
32	5	3.23022D-03	3.23022D-03	3.23022D-03	3.23022D-03	3.23022D-03	
39	5	2.20261D-03	2.20261D-03	2.20261D-03	2.20261D-03	2.20261D-03	
46	5	2.20261D-03	2.20261D-03	2.20261D-03	2.20261D-03	2.20261D-03	
53	1	1.21920D+00					
54	1	3.06070D-03					
78	1	3.61950D-03					
102	2	3.06070D-03	3.61950D-03				
104	1	1.00000D-06					
128	1	3.06070D-03					
152	5	4.44500D-01	4.57200D-01	5.96900D-01	2.33680D+00	8.63600D-01	
159	5	2.16085D-03	2.16085D-03	2.16085D-03	2.16085D-03	2.16085D-03	
166	1	6.41566D-05					
169	5	9.04875D-04	2.79400D-04	9.04875D-04	5.58800D-04	9.04875D-04	
180	2	3.06070D-03	3.61950D-03				
182	5	2.27420D-02	2.27420D-02	2.27420D-02	2.27420D-02	2.27420D-02	
189	5	9.04875D-04	2.79400D-04	9.04875D-04	0.00000D+00	9.04875D-04	
-1							
GEOMIN	61	2	1				
-1							
GEOMIN	61	3	2				
1	5	2.63808D-05	2.63808D-05	2.63808D-05	2.63808D-05	2.63808D-05	
32	5	2.85779D-03	2.85779D-03	2.85779D-03	2.85779D-03	2.85779D-03	
54	1	4.41960D-03					
78	1	4.97840D-03					
102	2	4.41960D-03	4.97840D-03				
128	1	4.41960D-03					
159	5	3.15494D-03	3.15494D-03	3.15494D-03	3.15494D-03	3.15494D-03	
166	1	1.04244D-04					
169	5	1.24460D-03	2.79400D-04	1.24460D-03	5.58800D-04	1.24460D-03	

180	2	4.41960D-03	4.97840D-03			
182	5	3.12802D-02	3.12802D-02	3.12802D-02	3.12802D-02	3.12802D-02
189	5	1.24460D-03	2.79400D-04	1.24460D-03	0.00000D+00	1.24460D-03
-1						
POWINC	62	1	0			
2	1	7.80000D-03				
4	2	4.80000D-03	1.14000D-02			
6	5	7.52221D-03	6.86521D-03	6.60043D-01	7.14610D-01	7.81665D-01
11	5	8.47031D-01	8.72372D-01	9.20434D-01	9.59184D-01	9.87454D-01
16	5	9.94152D-01	1.00000D+00	9.94698D-01	9.78333D-01	9.64462D-01
21	5	9.25834D-01	8.76984D-01	8.19028D-01	7.83962D-01	7.06867D-01
26	4	6.27461D-01	5.58589D-01	2.23048D-03	2.29822D-03	
30	5	1.0	1.0	1.0	1.0	1.0
35	5	1.0	1.0	1.0	1.0	1.0
40	1	1.0				
45	1	1.0				
62	1	4.77626D-05				
63	1	2.95938D-05				
64	5	0.0	0.0	6.53705D-02	5.11790D-02	4.71601D-02
69	5	4.74705D-02	4.87359D-02	5.14942D-02	5.42450D-02	5.64256D-02
74	5	5.73831D-02	5.77311D-02	5.70859D-02	5.54695D-02	5.34117D-02
79	5	4.99994D-02	4.60121D-02	4.19347D-02	3.90310D-02	3.66708D-02
84	4	3.76275D-02	4.55639D-02	0.0	0.0	
112	5	7.68718D-07	1.34947D-06	5.13797D-07	3.34114D-06	7.36556D-06
117	5	1.13848D-05	1.51386D-05	1.83871D-05	2.11185D-05	2.31976D-05
122	5	2.45201D-05	2.49942D-05	2.46198D-05	2.34199D-05	2.14234D-05
127	5	1.87725D-05	1.56091D-05	1.20689D-05	8.06958D-06	3.87347D-06
132	4	1.11838D-07	3.92056D-06	4.80780D-06	3.59017D-06	
160	5	4.60136D-07	7.64374D-07	2.16637D-07	1.26528D-06	2.80550D-06
165	5	4.31357D-06	5.67833D-06	6.86098D-06	7.85596D-06	8.60946D-06
170	5	9.07766D-06	9.24420D-06	9.10656D-06	8.67361D-06	7.95911D-06
175	5	6.99292D-06	5.83519D-06	4.54082D-06	3.07715D-06	1.49636D-06
180	4	2.41444D-08	1.45903D-06	1.80113D-06	1.34482D-06	
208	5	6.29119D-06	7.50829D-06	9.39567D-06	1.14188D-05	1.35669D-05
213	5	1.56950D-05	1.67334D-05	1.84367D-05	1.98554D-05	2.09211D-05
218	5	2.12357D-05	2.14685D-05	2.12707D-05	2.06518D-05	2.00693D-05
223	5	1.86791D-05	1.69938D-05	1.50962D-05	1.38978D-05	1.17380D-05
228	4	9.65240D-06	7.83908D-06	6.48397D-06	5.24731D-06	
256	1	1.61197D+00				
-1						
POWINC	62	2	1			
6	5	5.86759D-03	5.33443D-03	5.20808D-01	5.63896D-01	6.17881D-01
11	5	6.70723D-01	6.94061D-01	7.33000D-01	7.64269D-01	7.86999D-01
16	5	7.93042D-01	7.97534D-01	7.92893D-01	7.79149D-01	7.66228D-01
21	5	7.34015D-01	6.92923D-01	6.43946D-01	6.10760D-01	5.47504D-01
26	4	4.83681D-01	4.32813D-01	1.75730D-03	1.84146D-03	
62	1	1.88176D-03				
63	1	1.12517D-03				
64	5	0.0	0.0	6.79383D-02	5.14379D-02	4.66691D-02
69	5	4.69138D-02	4.84767D-02	5.14359D-02	5.43435D-02	5.66340D-02
74	5	5.77017D-02	5.80457D-02	5.73265D-02	5.55620D-02	5.32120D-02
79	5	4.94899D-02	4.50177D-02	4.05210D-02	3.72960D-02	3.51529D-02
84	4	3.74078D-02	4.94191D-02	0.0	0.0	
112	5	7.06831D-07	1.21273D-06	6.99901D-07	1.73069D-06	4.28040D-06

117	5	6.82532D-06	9.19978D-06	1.12647D-05	1.29962D-05	1.43095D-05
122	5	1.51379D-05	1.54290D-05	1.51772D-05	1.43955D-05	1.31043D-05
127	5	1.13725D-05	9.27257D-06	6.91879D-06	4.37995D-06	1.80143D-06
132	4	-6.03636D-07	-2.72385D-06	-3.18077D-06	-2.19934D-06	
160	5	3.61970D-07	6.02637D-07	2.93087D-07	-6.33355D-07	-1.60551D-06
165	5	-2.55822D-06	-3.42208D-06	-4.17360D-06	-4.80362D-06	-5.27911D-06
170	5	-5.57259D-06	-5.67401D-06	-5.57960D-06	-5.29400D-06	-4.82637D-06
175	5	-4.18982D-06	-3.41501D-06	-2.54939D-06	-1.62128D-06	-6.60695D-07
180	4	2.42103D-07	1.01320D-06	1.14178D-06	7.70551D-07	
208	5	4.24943D-06	5.03460D-06	6.26250D-06	7.63857D-06	9.09742D-06
213	5	1.05447D-05	1.13216D-05	1.24844D-05	1.34521D-05	1.41778D-05
218	5	1.44127D-05	1.45653D-05	1.44204D-05	1.39842D-05	1.35398D-05
223	5	1.25716D-05	1.13976D-05	1.00745D-05	9.15256D-06	7.66715D-06
228	4	6.24262D-06	4.97042D-06	4.07537D-06	3.25050D-06	
256	1	1.27509D+00				
-1						
POWINC	62	3	2			
6	5	1.06219D-02	9.53833D-03	4.41783D-01	4.78640D-01	5.26215D-01
11	5	5.73515D-01	6.64910D-01	7.04056D-01	7.35208D-01	7.58132D-01
16	5	7.95415D-01	8.00555D-01	7.95894D-01	7.81378D-01	7.24130D-01
21	5	6.92717D-01	6.53373D-01	6.06467D-01	4.98539D-01	4.47669D-01
26	4	3.98848D-01	3.64095D-01	3.01641D-03	3.20640D-03	
62	1	-1.72639D-03				
63	1	-1.17952D-03				
64	5	0.0	0.0	5.73903D-02	4.68623D-02	4.42674D-02
69	5	4.56973D-02	4.84666D-02	5.21043D-02	5.55065D-02	5.82174D-02
74	5	5.97452D-02	6.02908D-02	5.95928D-02	5.76656D-02	5.48720D-02
79	5	5.08990D-02	4.63283D-02	4.16845D-02	3.80583D-02	3.60123D-02
84	4	3.81542D-02	4.81859D-02	0.0	0.0	
112	5	1.53795D-07	7.83238D-07	2.32342D-06	5.12263D-06	7.98484D-06
117	5	1.07948D-05	1.34725D-05	1.59054D-05	1.79169D-05	1.94286D-05
122	5	2.03757D-05	2.07202D-05	2.04276D-05	1.95072D-05	1.80318D-05
127	5	1.60548D-05	1.36507D-05	1.09229D-05	8.10984D-06	5.34322D-06
132	4	2.70143D-06	3.04901D-07	-1.21017D-06	-1.63467D-06	
160	5	1.47084D-07	2.32594D-08	-5.67388D-07	-1.48830D-06	-2.45083D-06
165	5	-3.39400D-06	-4.30644D-06	-5.12421D-06	-5.79412D-06	-6.29780D-06
170	5	-6.62264D-06	-6.73877D-06	-6.63818D-06	-6.32272D-06	-5.81143D-06
175	5	-5.14713D-06	-4.34204D-06	-3.42375D-06	-2.47062D-06	-1.56156D-06
180	4	-7.06677D-07	2.83882D-08	4.68197D-07	5.55331D-07	
208	5	7.60241D-07	4.92571D-07	3.26012D-07	9.46623D-08	-1.71690D-07
213	5	-4.46680D-07	-1.43007D-07	-3.31006D-07	-4.84264D-07	-5.98656D-07
218	5	-3.49359D-07	-3.72382D-07	-3.52954D-07	-2.90759D-07	-6.37828D-07
223	5	-4.72948D-07	-2.68930D-07	-3.24153D-08	-3.55941D-07	-3.49901D-08
228	4	2.54229D-07	4.77699D-07	6.41543D-07	6.80340D-07	
256	1	7.37602D-01				
-1						
PMATCH	63	1	0			
2	3	1.80000D	03	6.10000D-05	1.32000D-06	
5	3	1.00000D	06	1.00000D	06	2.86000D-02
11	5	2.72000D	01	2.72000D	01	2.72000D 01 2.72000D+01
18	5	2.72000D	01	2.72000D	01	2.72000D 01 2.72000D+01
25	3	4.818	D-08	0.0	1.01325D	05
28	5	2.72000D	01	2.72000D	01	2.72000D 01 2.72000D+01
35	2	7.76000D	03	8.45000D	02	

406	3	15	21	34								
418	3	2	2	2								
470	1	3										
473	3	12	29	32								
481	3	37	48	55								
497	1	1										
512	1	40										
513	10	4	7	4	4	7	4	4	7	4	4	
523	10	41	4	4	4	4	4	4	4	4	4	
533	10	4	41	4	4	41	4	20	20	41	10	
543	10	4	4	4	2	10	10	4	2	10	10	
613	10	2	3	4	5	6	7	8	9	10	11	
623	10	12	13	14	16	18	20	22	24	26	27	
633	10	28	29	30	31	32	33	34	38	40	41	
643	10	42	43	44	45	46	49	51	52	53	56	
713	10	2	3	4	5	6	7	8	9	10	11	
723	10	12	13	15	17	19	21	23	25	26	27	
733	10	28	29	30	31	32	33	36	39	40	41	
743	10	42	43	44	45	47	50	51	52	54	57	
813	1	3										
814	3	7	7	7								
822	3	1	1	1								
830	3	3	6	9								
839	2	1	40									
852	1	1										
864	1	3										
890	1	50										
973	2	1	33									
982	1	2										
991	3	2	45	52								
1008	5	4	-2	-6	-7	-3						
1052	3	-2	-3	-6								
1082	3	-6	-7	-7								
1153	2	2	2									
1156	2	1	1									
1171	1	0										
1288	2	200	20									
-1												
PMR4IN	18	1	0									
2	5	1.63916D+04	1.08961D+01	3.45370D+02	2.51560D+02	4475.85						
7	5	4492.3	2246.15	2246.15	4492.3	2246.15						
12	5	2246.15	0.0	0.0	63.0	65.8						
17	5	4460.4	4460.4	0.0	65.8	63.0						
42	5	-1.4986	-1.4986	-1.4986	-1.4986	8.3566						
47	5	-2.8448	0.3048	0.3048	-2.8448	0.3048						
52	5	0.3048	3.2004	1.6764	7.5946	9.49962						
57	5	19.2532	19.2532	19.2532	33.2740	33.2740						
82	5	1.	1.	90.	90.	1.						
87	5	1.	1.	1.	4.	4.						
92	5	2.	1.	1.	2.	1.						
97	5	2.	2.	1.	1.	2.						
102	5	1.	2.	1.	1.	1.						
107	5	1.	1.	1.	1.	1.						
112	5	1.	1.	1.	1.	1.						

117	5 1.	1.	1.	1.	1.
162	5 3.2004	0.0000	2.3368	3.2004	0.0000
167	5 2.3368	3.2004	0.0000	2.3368	3.2004
172	5 8.25502	0.3302	-2.5400	3.7084	7.1120
177	5-3.3020	-3.3020	-3.3020	-3.3020	+3.7084
182	5 7.1120	-3.3020	-3.3020	-3.3020	-3.3020
187	5 3.1004	1.7764	7.5946	3.8862	3.8862
192	5 7.5946	3.8862	3.7862	19.2532	19.2532
197	5 0.3302	8.25502	15.9004	15.9004	-6.7056
202	5 19.2532	19.2532	24.0792	33.2740	32.0548
207	5 32.0548	3.8862	7.5946	33.2740	33.2740
212	5 33.2740	32.0548	32.0548	3.8862	7.5946
217	2 33.2740	33.2740			
302	5 4.699	1.4986	2.3368	0.8636	1.4986
307	5 2.3368	0.8636	1.4986	2.3368	0.8636
312	5 1.885	7.92482	2.87020	6.5532	3.4036
317	5 3.6068	4.03861	3.6068	4.03861	6.5532
322	5 3.4036	3.6068	4.03861	3.6068	4.03861
327	5 0.1	0.1	0.2159	3.7084	0.2159
332	5 1.90502	3.7084	0.1	1.0	17.5970
337	5 18.9230	7.92482	7.64538	22.7076	22.6060
342	5 25.9588	6.47701	4.826	3.6069	11.2167
347	5 4.3282	28.1686	3.7084	25.6794	1.0
352	5 3.6069	11.2167	4.3282	28.1686	3.7084
357	2 25.6794	1.0			
442	5 1.0	0.00445836	0.00445836	0.00445836	1.14510
447	5 1.14510	1.14510	0.860023	0.860023	0.860023
452	5 2.7545	3.0397	2.3776	0.50278	0.8234
457	5 0.29186	0.29186	0.29186	0.29186	0.50278
462	5 0.8234	0.29186	0.29186	0.29186	0.29186
467	5 0.1000	0.1000	0.62264	0.24520	0.62264
472	5 0.62264	0.072119	0.62264	0.8234	0.62072
477	5 0.62072	4.58089	0.62072	0.62072	0.882
482	5 0.62072	0.62072	0.15518	0.11224	0.097763
487	5 0.11224	0.11224	0.072119	0.11224	0.11224
492	5 0.11224	0.097763	0.11224	0.11224	0.2452
497	2 0.11224	0.11224			
582	5 1.0	3.06025D-03	3.06025D-03	3.06025D-03	0.0025
587	5 0.0025	0.0025	0.5	0.5	0.5
592	5 0.4572	0.026289	1.7399	0.8001	0.3302
597	5 0.6096	0.6096	0.6096	0.6096	0.8001
602	5 0.3302	0.6096	0.6096	0.6096	0.6096
607	5 0.35682	0.35682	0.53045	0.04458	0.53045
612	5 0.53045	0.01732	0.53045	0.3302	0.889
617	5 0.889	0.03502	0.889	0.889	0.03502
622	5 0.889	0.889	0.4445	0.37803	0.04445
627	5 0.37803	0.37803	0.017323	0.37803	0.37803
632	5 0.37803	0.04445	0.37803	0.37803	0.04458
637	2 0.37803	0.37803			
722	5 0.00001	0.00001	0.00001	0.00001	0.00001
727	5 0.00001	0.00001	0.00001	0.00001	0.00001
732	5 0.00001	0.00001	0.00001	0.00001	0.00001
737	5 0.00001	0.00001	0.00001	0.00001	0.00001
742	5 0.00001	0.00001	0.00001	0.00001	0.00001

747	5	0.00001	0.00001	0.00001	0.00001	0.00001
752	5	0.00001	0.00001	0.00001	0.00001	0.00001
757	5	0.00001	0.00001	0.00001	0.00001	0.00001
762	5	0.00001	0.00001	0.00001	0.00001	0.00001
767	5	0.00001	0.00001	0.00001	0.00001	0.00001
772	5	0.00001	0.00001	0.00001	0.00001	0.00001
777	2	0.00001	0.00001			
862	5	0.0	0.0	0.0	0.0	0.0
867	5	0.0	0.0	0.0	0.0	0.0
872	5	0.0	0.0	0.0	0.0	0.0
877	5	1.0	1.0	1.0	1.0	0.0
882	5	0.0	1.0	1.0	1.0	1.0
887	5	0.0	0.0	0.0	0.0	0.0
892	5	0.0	0.0	0.0	0.0	3.0
897	5	0.0	0.0	0.0	3.0	0.0
902	5	3.0	0.0	1.0	0.0	6.0
907	5	1.0	0.0	0.0	1.0	0.0
912	5	0.0	6.0	1.0	0.0	0.0
917	2	1.0	0.0			
1013	1	7.25840D+00				
1017	1	1.63623D 00				
1019	1	1.63623D 00				
1023	1	1.63623D 00				
1025	1	1.63623D 00				
1038	1	4.36618D+01				
1142	1	12.0				
1143	5	1.	7.71770D+04	7.71770D+04	7.71770D+04	2.71657D+07
1148	5	2.71657D+07	2.71657D+07	1.00000D+05	1.00000D+05	1.00000D+05
1153	5	7.63334D+06	2.52430D+06	4.95536D+06	2.91088D+05	3.60030D+06
1158	5	5.31795D+04	5.31795D+04	5.31795D+04	5.31795D+04	2.91088D+05
1163	5	3.60030D+06	5.31795D+04	5.31795D+04	5.31795D+04	5.31795D+04
1168	5	1.0	1.0	2.50694D+05	2.50694D+05	2.50694D+05
1173	5	2.50694D+05	6.67180D+04	2.50694D+05	3.60030D+06	1.58978D+05
1178	5	1.58978D+05	2.50783D+06	1.58978D+05	1.58978D+05	2.11374D+06
1183	5	1.58978D+05	1.58978D+05	7.83699D+04	8.05690D+04	7.72335D+04
1188	5	8.05690D+04	8.05690D+04	6.67181D+04	8.05690D+04	8.05690D+04
1193	5	8.05690D+04	7.72335D+04	8.05690D+04	8.05690D+04	2.50694D+05
1198	2	8.05690D+04	8.05690D+04			
1283	5	1.	5.68800D+03	5.68800D+03	5.68800D+03	5.0 D+03
1288	5	5.0 D+03	5.0 D+03	5.0 D+03	5.0 D+03	5.0 D+03
1293	5	1.11450D+03	2.47769D+04	1.48670D+03	1.11450D+03	1.34750D+02
1298	5	4.45980D+03	4.45980D+03	4.45980D+03	4.45980D+03	1.11450D+03
1303	5	1.34750D+02	4.45980D+03	4.45980D+03	4.45980D+03	4.45980D+03
1308	5	1.0 D+05	1.0 D+05	1.23880D+03	1.23880D+03	1.23880D+03
1313	5	1.23880D+03	3.27930D+04	1.23880D+03	1.34750D+02	2.22990D+03
1318	5	2.22990D+03	5.57480D+02	2.22990D+03	2.22990D+03	5.57480D+02
1323	5	2.22990D+03	2.22990D+03	2.22990D+03	1.99635D+03	7.43310D+03
1328	5	1.99635D+03	1.99635D+03	3.27930D+04	1.99635D+03	1.99635D+03
1333	5	1.99635D+03	7.43310D+03	1.99635D+03	1.99635D+03	1.23880D+03
1338	2	1.99635D+03	1.99635D+03			
1423	5	61.126	2497.5	2664.6	6.0	6.0
1428	5	140.5	140.5	3.67	1.0	30.395
1433	2	20.0	20.0			
1461	5	0.0	1.01325D+05	1.01325D+05	0.0	0.0

1466	5	0.0	0.0	2.73646D+05	0.0	2.73646D+05
1471	2	1.01325D+05	1.01325D+05			
1499	5	2.0 D-08	2.0 D-08	2.0 D-08	2.0 D-08	2.0 D-08
1504	5	2.0 D-08	2.0 D-08	2.0 D-08	2.0 D-08	2.0 D-08
1509	2	2.0 D-08	2.0 D-08			
1537	5	6.00000D-05	6.00000D-05	6.00000D-05	6.00000D-05	6.00000D-05
1542	5	6.00000D-05	6.00000D-05	6.00000D-05	6.00000D-05	6.00000D-05
1547	2	6.00000D-05	6.00000D-05			
1575	5	-1.4986	3.2004	-2.8448	0.3048	0.3048
1580	5	2.8194	2.0574	19.2532	19.2532	24.0792
1585	2	33.2740	33.2740			
1613	5	1.0	202.3	16.7	1.0	1.0
1618	5	1.0	1.0	0.5	1.0	12.697
1623	2	4.0	4.0			
1651	5	0.0	758.3	758.3	0.0	0.0
1689	4	1.6	2078.	1.94 D-05	273.	
1693	1	1.828				
1721	1	1.78000D-03				
1749	2	1.	1.			
1805	1	4.76250D-02				
1833	1	1.00000D-05				
1861	5	0.0	1.00000D+08	1.00000D+08	0.0	0.0
1866	5	0.0	0.0	1.00000D+08	0.0	1.00000D+08
1871	2	1.00000D+08	1.00000D+08			
1899	3	18650.0	18650.0	400.0		
1911	3	7.58423D+05	7.58423D+05	2.42338D+05		
1923	3	51.84	51.84	116.76		
1935	3	4492.3	4492.3	4460.4		
1947	3	0.689	0.689	0.83		
1983	5	0.01	0.0	-73.13	0.00268	0.0
1988	5	0.07	0.00383	0.01071	0.01406	0.01
1993	1	0.268				
2000	3	1.0	0.03	0.03		
2003	5	0.01	0.0	-73.13	0.00268	0.0
2008	5	0.07	0.00383	0.01071	0.01406	0.01
2013	1	0.268				
2020	3	1.0	0.03	0.03		
2023	5	0.01	0.0	-73.13	0.00268	0.0
2028	5	0.07	0.00383	0.01071	0.01406	0.01
2033	1	0.268				
2040	3	1.0	0.03	0.03		
2223	2	1.0	1.0			
2233	2	0.0	10000.			
2243	2	1.0	1.0			
2253	2	0.0	10000.			
2263	2	1.0	1.0			
2273	2	0.0	10000.			
2463	1	9.80700D+00				
2464	5	2.13 D-10	2.13 D-10	2.13 D-10	2.13 D-10	2.13 D-10
2469	5	2.13 D-10	2.13 D-10	2.13 D-10	2.13 D-10	2.13 D-10
2474	5	2.13 D-10	2.13 D-10			
2578	5	282.2	282.2	282.2	431.2	431.2
2583	5	423.2	423.2	431.2	50.0	557.5
2588	2	1128.6	1128.6			

2616	5	67.56	374.0	617.6	23.086	23.086
2621	5	251.1	251.1	6.7827	1.0	27.24
2626	2	1.0	1.0			
2654	5	2.27493D+07	1.25940D+08	2.38580D+08	5.18247D+06	5.18247D+06
2659	5	5.63681D+07	5.63681D+07	1.52262D+06	2.80606D+04	6.33777D+06
2664	2	2.80606D+04	2.80606D+04			
2694	3	0.025	0.80	4.80		
2697	3	26.2	26.2	26.2		
2705	3	26.2	26.2	26.2		
2713	3	26.2	26.2	26.2		
2721	3	26.2	26.2	26.2		
2729	3	26.2	26.2	26.2		
2737	3	2.07845D-03	1.38584D-03	5.0	D-04	
2745	3	5.588 D-04	1.38584D-03	5.0	D-04	
2753	3	3.810 D-03	3.810 D-03	5.0	D-04	
2761	3	4.709 D+06	4.709 D+06	4.709	D+06	
2769	3	4.709 D+06	4.709 D+06	4.709	D+06	
2777	3	4.709 D+06	4.709 D+06	4.709	D+06	
2785	3	4.709 D+06	4.709 D+06	4.709	D+06	
2793	3	4.709 D+06	4.709 D+06	4.709	D+06	
2801	5	1.	1.	1.	1.	1.
2806	5	1.	1.	1.	1.	1.
2811	5	1.	1.	1.	1.	1.
2816	5	1.	1.	1.	1.	1.
2821	1	1.				
2857	5	3.51967D-04	0.	0.	0.	0.
2862	4	0.	0.	0.	0.	
2881	5	0.	0.	0.	0.	0.
2886	4	0.	0.	0.	0.	
2913	3	0.	0.	0.		
2921	3	4.98955	1632.37	1.00		
2929	3	0.49546	169.45	0.10		
2937	2	1.0	1.0			
2951	2	0.0	0.0			
3105	2	4.5974	4.5974			
3273	2	0.0	10000.			
3287	2	0.0	10000.			
3441	1	0.9999				
3445	3	7.0300 D+05	3.50000D+05	3.50000D+05		
3455	5	0.5	1.	1.	1.	1.
3460	5	1.	1.	1.	1.	1.
3465	5	1.	1.	1.	1.	1.
3470	5	1.	1.	1.	1.	1.
3475	5	1.	1.	1.	1.	1.
3480	5	1.	1.	1.	1.	1.
3485	5	1.	1.	1.	1.	1.
3490	5	1.	1.	1.	1.	1.
3495	1	0.5				
3516	3	14.73165	3.63736	3.63736		
3524	3	502.7	18.31328	18.31328		
3532	3	462.5	16.653	16.653		
3540	3	0.0508	0.02285	0.02285		
3548	3	1.14300D-03	8.63600D-04	8.63600D-04		
3556	3	4.58 D+06	4.41900D+06	4.41900D+06		

```

3564      3 4.58      D+06 4.41900D+06 4.41900D+06
3572      3 28.32      21.5      21.5
3580      3 28.32      21.5      21.5
3588      3 1.0      D+09 1.0      D+09 1.0      D+09
3596      3 1.0      D+09 1.0      D+09 1.0      D+09
3604      3 1.0      1.0      1.0
3612      5 0.0      1184.98      1035.63      0.0      0.0
3617      5 0.0      0.0      1.0      0.0      15.197
3622      2 10.0      10.0
3650      5 0.025      0.80      4.80      0.025      0.80
3655      1 4.80
3801      2 2.22250D-02 2.22250D-02
3805      2 2.41300D-02 2.41300D-02
3809      2 21.5      21.5
3813      2 3.73000D+01 3.73000D+01
3817      2 8.6      D+04 8.6      D+04
3821      2 3059.16      3059.16
3825      2 29.3878      29.3878
3829      2 0.84      0.84
3833      2 0.01      0.01
3837      2 0.0      0.0
3841      2 17.5958      17.5958
3845      2 29.2335      29.2335
3849      2 18.2709      18.2709
3853      2 143.6      143.6
3863      2 130.8      130.8
3893      2 0.0      10000.
3903      2 0.0      10000.
3933      1 300.
4174      4 1.25980D-04 1.52400D-05 1.52400D-05 1.32080D-05
4216      3 77443.6      16615.3      77443.6
4285      2 168.0      168.0
-1
ENDJOB      -1

```

II. Input Data for the Transients

The calculation of any particular transient requires very little additional input data. The data generally describe the behavior of the major reactor systems, such as a pump trip, steam generator trip, or control rod runout. The data for the three transients discussed in this report are given in the following three sections.

A. Unprotected Loss-of-Flow Accident

In addition to the data in Section I, the loss-of-flow calculation requires a pump trip of all primary and intermediate loop pumps, and a loss of the steam generator heat sink. To run this calculation, the following input data is included with the data in Section I, just prior to the ENDJOB card.

```
XNOTES      0      0      0
```

```

*****
*
*   Input required for LOF calculation.
*
*****

```

```

-1
INPCOM      1      1      1
11          3 1200      40      40
-1
OPCIN       11      1      1
7           1 3.00000D+02
-1
PMR4IN      18      1      1
2223        2 0.0          0.0
2233        2 0.0          10000.0

2243        2 0.0          0.0
2253        2 0.0          10000.0
2263        2 0.0          0.0
2273        2 0.0          10000.0
2937        3 1.0          0.0          0.0
3105        3 4.5974        4.5974        4.5974
3273        3 0.0          20.0          10000.0
-1

```

B. Unprotected Transient Overpower Accident

In addition to the input data in Section I, the transient overpower accident requires a control rod withdrawal. This is simulated by a reactivity insertion table in the input data. All other conditions remain the same. To run this calculation, the following data is included with the data in Section I, just prior to the ENDJOB card:

```

XNOTES      0      0      0
*****
*
* Input required for TOP calculation.
*
*****
-1
INPCOM      1      1      1
11          3 800      50      50
18          1      3
-1
OPCIN       11      1      1
5           1 5.00000D-01
7           1 2.00000D+02
-1
POWINA      12      1      1
29          3 0.0          0.11          0.11
49          3 0.0          22.0          1000.0
-1
PMR4IN      18      1      1
2223        2 1.0          1.0
2233        2 0.          10000.0
2243        2 1.0          1.0
2253        2 0.0          10000.0
2263        2 1.0          1.0
2273        2 0.0          10000.0

```

2937	3	1.0	1.0	1.0
3105	3	4.5974	4.5974	4.5974
3273	3	0.0	20.0	10000.0
-1				

C. Unprotected Loss-of-Heat Sink Accident

In addition to the input data in Section I, the loss-of-heat-sink accident requires a loss of heat rejection capability at the steam generator. All other conditions remain the same. To run this calculation, the following data is included with the data in Section I, just prior to the ENDJOB card.

```

XNOTES      0      0      0
*****
*
*   Input required for LOHS calculation.
*
*****
-1
INPCOM      1      1      1
11          3 3000 300 300
-1
OPCIN       11      1      1
5           1 5.00000D-01
7           1 1.50000D+03
-1
PMR4IN      18      1      1
2223        2 1.0      1.0
2233        2 0.       10000.0
2243        2 1.0      1.0
2253        2 0.0      10000.0
2263        2 1.0      1.0
2273        2 0.0      10000.0
2937        3 1.0      0.0      0.0
3105        3 4.5974   4.5974   4.5974
3273        3 0.0      20.0     10000.0
-1

```

This case also requires a restart to continue the calculation to 3000 seconds of total problem time. This is accomplished with the following dataset, which is the only input data used for the restart. The remainder of the data is already available in the restart file written by the previous calculation.

3500 MWE METALLIC FUEL CORE, LSPB PLANT, ANL/EUROPEAN COMPARISON STUDY
3 CHANNEL DECK FOR ATWS TRANSIENTS, ARGONNE NATL LAB., RESTART, 4/17/89

```

3 0 0 0 0
RESTART 100      0      0

INPCOM      1      1      1
11          1 6000
-1
OPCIN       11      1      1
7           1 3000.0
-1
ENDJOB      -1

```

APPENDIX H. SAS4A CODE MODIFICATIONS

The analyses of the various transients described in this report required a number of minor changes to the SAS4A Version 2.0 computer code. The following is a listing of all of the changes used in the calculations.

Subroutine REXDET

```

*DELE REXD..27 REXD..28
*DELE REXD..30
*INSE REXD..55
      ATTN1S=0.000
      ATTN2S=0.000
*INSE REXD..75
      IF(IPLUP.EQ.1) JSTRDX=JCPLBT+MZ+NPLN
*SUBS REXD..85
      W2REX1=W0
      W2REX2=W2(1)
      ATTN1S=ATTN1S+W2REX1*WTMP
      ATTN2S=ATTN2S+W2REX2*WTMP
      WTCSS=WTCSS+W2REX2*WTMP
*INSE REXD.107
      ATTN1=ATTN1S
      ATTN2=ATTN2S
*INSE REXD.222
      IF(IPLUP.EQ.1) ZLBRFI=ZCOOL(JCPNBT+NPLN)-ZCOOL(1)
*DELE REXD.562
      DISP2=R3
      ATTN1T=0.000
      ATTN2T=0.000
      NFLRVR=0
      NSBBLG=0
*INSE REXD.636
      IF(IPLUP.EQ.1) JSTRDX=JCPLBT+MZ+NPLN
*SUBS REXD.646
      W2REX1=W2REX2
      W2REX2=W2(1)
      IF(ICALC(ICH).LT.0.000) GO TO 5350
      IF(W2(1).LT.0) GO TO 5355
5340  ATTN1T=ATTN1T+W2REX1*WTMP
      ATTN2T=ATTN2T+W2REX2*WTMP
      GO TO 5360
5350  W2REX2=W2REX1*ATTN2/ATTN1
      NSBBLG=NSBBLG+1
      GO TO 5360
5355  W2REX2=-W2(1)
      NFLRVR=NFLRVR+1
5360  WTCTR=WTCTR+W2REX2*WTMP
*INSE REXD.664
      IF(ATTN1T.GT.0.000) ATTN1=ATTN1T
      IF(ATTN2T.GT.0.000) ATTN2=ATTN2T
*DELE REXD1066

```



```

DISP2=R3
*   WRITE(IWRITE,12201) ATTN1,ATTN1T,ATTN2,ATTN2T
    WRITE(IWRITE,12202) NSBBLG,NFLRVR
*INSE REXD1155
12201 FORMAT('  ATTN1=',1P,D13.6,' ATTN1T=',D13.6,'  ATTN2=',D13.6,
& '  ATTN2T=',D13.6)
12202 FORMAT('  NSBBLG=',I5,8X,'NFLRVR=',I5)

```

Subroutine FEEDSB

```

*INSE FDSB.456
    DPTIP=DPTI
    DPTI=PTI-PTIST

```

Subroutine SSPMLP

```

*INSE SS1L.133
    IF(JCVL(1,ISGL).EQ.ICV.AND.JCVL(2,ISGL).EQ.ICV) FLOOP=0.000

```

Subroutine PRMPRT

```

*INSE PRPR.104
    CALL LINES(2,IFL1)
    WRITE (IWRITE,7800) QDHX
*INSE PRPR.386
    7800 FORMAT ('OTOTAL HEAT REMOVAL =',E13.5)

```

Subroutine LQGSFN(ICV)

```

*SUBS LGFN..17
    VOLGC4(ICV)=VOLLGC(ICV)*(1.000+ALPHAT(ICV)*(TLQCV4(ICV)-TLCVSS(
1 ICV)))-VL4

```

Subroutine SSPRM4

```

*DELE SSP4.136 SSP4.137
*SUBS SSP4.141
    1900 DO 2000 ICV=1,NCVT
*INSE SSP4.198
C      DO THE PRIMARY LOOP
    CALL SSPMLP
*SUBS SSP4.213
    IF (FLOSSL(ISGL).LE.0.000) FLOSSL(ISGL)=FLOSSL(1)
*INSE SSP4.214
C      SAVE SS CV TEMPERATURES IN TLCVSS
    DO 3310 ICV=1,NCVT
    3310 TLCVSS(ICV)=TLQCV2(ICV)

```

Subroutine SSP4TH

```

*INSE SP4T..17
    QDHX=0.000

```

Subroutine SSADHX

```

*SUBS SSDX..41
  WA=1.100*FLOSL2(ISGL)*(CNA/CAIR)
*INSE SSDX..44
  IF (DABS(DENOM).GT.1.0D-12) GO TO 410
  WA=.99*WA
  GO TO 400
410 CONTINUE
*INSE SSDX.111
  ICVI=JCVL(1,ISGL)
  QDHX=QDHX+CNA*FLOSL2(ISGL)*(THOT-TCOLD)*CVMULT(ICVI)

```

Subroutine STEPTM

```

*INSE STTM..34
  QDHX=0.0D0
*INSE STTM.186
  TT=TCRIT-TIN4
  TTIN=1.0D0/TT
  CNA=((A28*TTIN+A29)*TTIN+A30+TT*(A31+A32*TT))
  ICVI=JCVL(1,ISGL)
  INODO=ITNDST(ITGP)+1
  IF (IDIR.LT.0) INODO=ITNDST(ITGP)
  QDHX=QDHX+CNA*FLOSL4(ISGL)*(TIN4-TLNOD4(INODO))*CVMULT(ICVI)
*INSE STTM.320
  IF (JSSCPC.EQ.0) GO TO 5920
  DO 5910 ICV=1,NCVT
5910 XLQMS4(ICV)=XLQMS3(ICV)
  GO TO 6010
5920 CONTINUE
*INSE STTM.323
6010 CONTINUE

```

Subroutine SSP4PR

```

*INSE SP4P..92
  CALL LINES(2,IFL1)
  WRITE (IWRITE,1900) QRVACS,QDHX
*INSE SP4P.121
1900 FORMAT ('ORVACS HEAT REMOVAL =',E13.5,
1 'AIR DUMP HEAT EXCHANGER HEAT REMOVAL =',E13.5)

```

Subroutine SSCPNL

```

*SUBS SSNL..60
  FRSTGN=(POWTOT-QRVACS-QDHX)/POWTOT
*INSE SSNL..83
CC    SET REFERENCE TEMPERATURES FOR CVS WITHOUT COVER GAS
  NCVT=NCVP+NCVS+NCVD
  DO 1110 ICV=1,NCVT
  IF (ITYPCV(ICV).GE.6) GO TO 1110
  TREFCV(ICV)=TLQCV2(ICV)
  TT=TLQCV2(ICV)

```

```
      XLQMS2(ICV)=VOLLGC(ICV)*(A12+TT*(A13+TT*A14))  
1110 CONTINUE
```

Subroutine REINIT

```
*DELE REIN..26
```

C

Subroutine DTMFND

```
*DELE DTMF..93  DTMF..96
```

```
500 RENEW=PREA(TIME1+DTIME)
```

```
*INSE DTMF.109
```

```
IF (DTIME.LT.2.D-5) DTIME=2.D-5
```

Subroutine PRMADJ

```
*DELE PRAJ..63
```

```
      VOLGNW=VOLLGC(ICV)*(1.0D0+ALPHAT(ICV)*(TLQCV2(ICV)-  
& TLCVSS(ICV)))-VL2
```