ANL-19/48

U.S. Effort Support to Examinations at Fukushima - November 2019 Meeting Notes with Updated Information Requests

Nuclear Engineering Division

About Argonne National Laboratory
Argonne is a U.S. Department of Energy laboratory managed by UChicago Argonne, LLC under contract DE-AC02-06CH11357. The Laboratory's main facility is outside Chicago, at 9700 South Cass Avenue, Argonne, Illinois 60439. For information about Argonne and its pioneering science and technology programs, see www.anl.gov.

DOCUMENT AVAILABILITY

Online Access: U.S. Department of Energy (DOE) reports produced after 1991 and a growing number of pre-1991 documents are available free at OSTI.GOV (http://www.osti.gov/), a service of the U.S. Dept. of Energy's Office of Scientific and Technical Information

Reports not in digital format may be purchased by the public from the National Technical Information Service (NTIS):

U.S. Department of Commerce
National Technical Information Service
5301 Shawnee Rd
Alexandria, VA 22312
www.ntis.gov
Phone: (800) 553-NTIS (6847) or (703)
605-6000 Fax: (703) 605-6900
Email: **orders@ntis.gov**

Reports not in digital format are available to DOE and DOE contractors from the Office of Scientific and Technical Information (OSTI):

U.S. Department of Energy
Office of Scientific and Technical Information
P.O. Box 62
Oak Ridge, TN 37831-0062
www.osti.gov
Phone: (865) 576-8401
Fax: (865) 576-5728
Email: **reports@osti.gov**

Disclaimer
This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor UChicago Argonne, LLC, nor any of their employees or officers, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of document authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof, Argonne National Laboratory, or UChicago Argonne, LLC.

U.S. Effort Support to Examinations at Fukushima - FY2020 Meeting Notes with Updated Information Requests

Prepared for:
U. S. Department of Energy - Office of Nuclear Energy

Report Contributors:
N. Andrews,[1] S. Basu,[2] R. Bunt,[3,4] M. Corradini,[5] P. Ellison,[4,6] M. Farmer,[7] J. Gabor,[8] R. Gauntt,[9] C. Henry,[10] T. Honda,[11] T. Kindred,[12] K. Klass,[4,13] T. Kobayashi,[11] S. Kraft,[14] W. Luangdilok,[10,15] R. Lutz,[16] D. Luxat,[1] S. Mizokami,[11] J. Nakano,[17] A. Nakayoshi,[18] J. Rempe,[19] M. Plys,[10] K. Robb,[20] K. Shearer,[21,22] H. Wabakabayashi,[17] P. Whiteman,[23] B. Williamson[4,24]

Contributor Organizations:
[1]Sandia National Laboratories, [2]McGill Engineering Associates, [3]Southern Nuclear, [4]BWR Owners Group, [5]University of Wisconsin - Madison, [6]GE-Hitachi Nuclear Energy Americas, [7]Argonne National Laboratory, [8]Jensen Hughes, [9]Gauntt Technical Associates, LLC, [10]Fauske and Associates LLC, [11]Tokyo Electric Power Company, Holdings, [12]Electric Power Research Institute, [13]Talen Energy, [14]Kraft-Contente, LLC, [15]H2 Technology LLC, [16]Lutz Nuclear Consulting, [17]Nuclear Damage Compensation and Decommissioning Facilitation Corporation, [18]Japan Atomic Energy Agency, [19]Rempe and Associates LLC, [20]Oak Ridge National Laboratory, [21]Westinghouse, [22]PWR Owners Group, [23]Framatome, [24]Tennessee Valley Authority,

Edited by: J.Rempe

January 31, 2020

ABSTRACT

Much is still not known about the end-state of core materials in each unit that was operating on March 11, 2011 at the Fukushima Daiichi Nuclear Power Station (Daiichi). Information obtained from Daiichi is required to inform Decontamination and Decommissioning (D&D) activities, improving the ability of the Tokyo Electric Power Company Holdings, Incorporated (TEPCO Holdings) to characterize potential hazards and to ensure the safety of workers involved with cleanup activities. This document summarizes results from the Fiscal Year 2020 (FY2020) U.S. effort to review Daiichi information and extract insights to enhance the safety of existing and future nuclear power plant designs. This U.S. effort, which was initiated in 2014 by the Department of Energy Office of Nuclear Energy (DOE-NE), is completed by a group of experts in reactor safety and plant operations that identify examination needs and evaluate recent Daiichi examination data to address these needs. Since its inception, annual reports were issued that document significant safety insights being obtained in areas of special emphasis: system and component performance, radionuclide surveys and sampling, debris end-state location, combustible gas effects, and plant operations and maintenance. In addition to reducing uncertainties related to severe accident modeling progression, these insights have and continue to be used to update guidance for severe accident prevention, mitigation, and emergency planning. Reduced uncertainties in modeling the events at Daiichi improve the realism of reactor safety evaluations that inform future D&D activities. For FY2020, it was decided that the program would gain more benefit from a more concise report that emphasizes new information and insights that affect prior findings and recommendations from the U.S. experts participating in this effort.

A key aspect of prior U.S. efforts, the updated list of information requests, is included in this letter report to ensure that they are transmitted to organizations within Japan. In addition, findings and associated recommendations are provided regarding information presented by TEPCO Holdings, Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF), and the Japan Atomic Energy Agency (JAEA). This letter report also continues to emphasize how information obtained from the affected reactors at Daiichi has been and will continued to be used to update severe accident management strategies, improve maintenance activities (especially in areas of radiation protection) and reduce uncertainties in systems analysis code models. In addition, recommendations are included that would expand the use of this information to provide insights regarding maintenance, radiation protection, design, and siting activities of existing and new reactors.

ACKNOWLEDGEMENTS

Successful preparation of this report required input and support from several individuals and organizations. Financial support for present and former national laboratory panel participants was provided by the U.S. Department of Energy Office of Nuclear Energy. In addition, there were substantial in-kind contributions made by various industry organizations that supported technical experts to participate in this multi-year effort; these organizations included the Boiling Water Reactor Owners Group, the Electric Power Research Institute, Exelon Corporation, GE-Hitachi, Fauske and Associates, LLC, Framatome, the Institute of Nuclear Power Operations, Jensen Hughes, the Nuclear Energy Institute, the Pressurized Water Reactors Owners Group, Southern Nuclear, the Tennessee Valley Authority, Westinghouse, and BWX Technologies. Finally, the U.S. Nuclear Regulatory Commission (U.S. NRC), the Tokyo Electric Power Company Holdings, Incorporated, and Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF), also provided technical experts to participate in the panel meetings. In particular, Mr. Takashi Hara, Mr. Tatsuro Kobayashi, Dr. Shinya Mizokami, and Mr. Daichi Yamada from TEPCO Holdings attended, as well as Mr. Donald Marksberry, Dr. Sudamay Basu (since retired), Dr. Richard Lee from the U.S. NRC Office of Nuclear Regulatory Research, Mr. Tom Kindred from the Electric Power Research Institute, Mr. Steven Kraft from the Nuclear Energy Institute (retired), and Mr. Damian Peko from the U.S. Department of Energy. These individuals facilitated the overall process by providing key clarifications in various areas as the meetings progressed. Their efforts are greatly appreciated.

CONTENTS

FIGURES

TABLES

ACRONYMS AND ABBREVIATIONS

AC	Alternating Current
ADS	Automatic Depressurization System
AFW	Auxiliary Feed Water
AM	Accident Management and Prevention
ANL	Argonne National Laboratory
ARC-F	Analysis of Information from Reactor Building and Containment Vessel and Water Sampling in Fukushima Daiichi Nuclear Power Station
ATF	Accident Tolerant Fuel
BAF	Bottom of Active Fuel
BIP	Behavior of Iodine Project
BSAF	Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant
BSAF2	Phase 2 of BSAF
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
BWXT	BWX Technologies
CBT	Computer Based Training
CCI	Core Concrete Interactions
CDF	Core Damage Frequency
CEA	French Alternative Energies and Atomic Energy Commission
CFD	Computational Fluid Dynamics
CFVS	Containment Filtered Venting System
CLADS	Collaborative Laboratories for Advanced Decommissioning Science
CNL	Canadian Nuclear Laboratories
CNWG	Civil Nuclear Energy Research and Development Working Group
CQ	CORQUENCH
CR	Control Rod
CRD	Control Rod Drive
CRIEPI	Central Research Institute of Electric Power Industry
CRGT	Control Rod Guide Tube
CS	Core Spray
CSARP	Cooperative Severe Accident Research Program
CSNI	Committee on the Safety of Nuclear Installations
CVS	Containment Venting System
Daiichi	Fukushima Daiichi Nuclear Power Station
DC	Direct Current or District of Columbia
D&D	Decontamination and Decommissioning
DF	Decontamination Factor
DOE	Department Of Energy

DOE-EM	Department of Energy Office of Environmental Management
DOE-NE	Department of Energy Office of Nuclear Energy
dP	Differential Pressure
DW or D/W	DryWell
EC	European Commission
EDF	Électricité de France
ELAP	Extended Loss of AC Power
EOP	Emergency Operating Procedure
EPC	Emergency Procedures Committee
EPG	Emergency Planning Guideline
EPRI	Electric Power Research Institute
ESTER	Experiments on Source Term for delayed Releases
EXOB	Expanded Operating Band
FAI	Fauske and Associates, LLC
FDW	FeeDWater
FE-SEM	Field Emission Scanning Electron Microscopy
FLEX	Diverse and Flexible Mitigation Capability (for accident mitigation)
FP	Fission Product
FY	Fiscal Year
GEH	GE-Hitachi Nuclear Energy, Limited
GEJE	Great East Japan Earthquake
HPCI	High Pressure Coolant Injection
HYMERES	Hydrogen Mitigation Experiments for REactor Safety
IAE	Institute of Applied Energy
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
ICP	Inductively Coupled Plasma
IEEE	Institute of Electrical and Electronics Engineers
IEEE PMBOK	IEEE Project Management Body of Knowledge
INL	Idaho National Laboratory
INPO	Institute of Nuclear Power Operations
IPRESCA	Integration of Pool Scrubbing Research to Enhance Source term Calculations
IRID	International Research Institute for Nuclear Decommissioning
IRSN	Institut de Radioprotection et de Sûreté Nucléaire
ISP NPP	Institute for Safety Problems of Nuclear Power Plants
IRM	Intermediate Range Monitor
IRSN	Institut de Radioprotection et de Sûreté Nucléaire
JAEA	Japan Atomic Energy Agency
JRC	Joint Research Centre

KAERI	Korea Atomic Energy Research Institute
KINS	Korea Institute of Nuclear Safety
LERF	Large Early Release Frequency
LIBS	Laser Induced Breakdown Spectroscopy
LNG	Liquefied natural gas
LT	Level Transmitter
LWR	Light Water Reactor
LWRS	Light Water Reactor Sustainability
MAAP	Modular Accident Analysis Program
MB	Management Board
MCCI	Molten Core Concrete Interactions
MELCOR	Methods for Estimation of Leakages and Consequences of Releases
METI	Ministry of Economy, Trade and Industry
MEXT	Ministry of Education, Culture, Sports, Science and Technology
MS	MELTSPREAD
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
NA	No information available
NANTeL	National Academy for Nuclear Training e-Learning
NC	Non-Condensible
NDF	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NPP	Nuclear Power Plant
NPS	Nuclear Power Station
NRA or NRAJ	Nuclear Regulatory Authority (Japan)
NRC	Nuclear Regulatory Commission
NTTF	Near Term Task Force
NUGENIA	NUclear GENeration II & III Association
OECD	Organization for Economic Cooperation and Development
ORNL	Oak Ridge National Laboratory
PARS	Passive Autocatalytic Recombiner System
PCV	Primary Containment Vessel
PSI	Paul Scherrer Institute
PLR	Primary Loop Recirculation
PM	Plant Maintenance
PreADES	Preparatory Studies for Fuel Debris Analysis
PRA	Probabilistic Risk Assessment
PRG	Program Review Group

PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group
R&D	Research and Development
RB	Reactor Building
RCIC	Reactor Core Isolation Cooling
RCW	Reactor Building Closed Cooling Water System
RES	U.S. NRC Office of Nuclear Regulatory Research
RHR	Residual Heat Removal
RN	RadioNuclide
ROSAU	Reduction Of Severe Accident Uncertainties
ROV	Remotely Operated Vehicle
RPV	Reactor Pressure Vessel
RST	Reactor Safety Technologies
SA	Severe Accident
SAG	Severe Accident Guideline (for BWRs)
SAMG	Severe Accident Management Guideline (for PWRs)
SAMPSON	Severe Accident Analysis Code with Mechanistic, Parallelized, Simulations Oriented towards Nuclear field
SAREF	SAfety REsearch opportunities post-Fukushima
SARNET	Severe Accident Research NETwork
SAWA	Severe Accident Water Addition
SAWM	Severe Accident Water Management
SBO	Station BlackOut
SC or S/C	Suppression Chamber
SDW	SAMG Direct Work
SGTS	Standby Gas Treatment System
SLC	Standby Liquid Cooling
SNF	Spent Nuclear Fuel
SNL	Sandia National Laboratories
SRM	Source Range Monitor
SRV	Safety Relief Valve
SSM	Strålsäkerhetsmyndigheten
STEM	Source Term Evaluation and Mitigation
TAF	Top of Active Fuel
TAG	Technical Advisory Group
TAMU	Texas A&M University
TBD	To Be Determined
TCOFF	Thermodynamic Characterization Of Fuel debris and Fission products based on Scenario Analysis for Severe Accident Progression at Fukushima-Daiichi NPS
TDAFW	Turbine Driven Auxiliary Feed Water

TDR	Time Domain Reflectrometry
TEM	Transmission Electron Microscopy
TEPCO Holdings	Tokyo Electric Power Company Holdings, Inc.
TIP	Traversing In-core Probe
TMI-2	Three Mile Island Unit 2
TSC	Technical Support Center
TSG	Technical Support Guidance
TTEXOB	Terry^{TM} Turbine Expanded Operating Band
TVA	Tennessee Valley Authority
U.S.	United States
V&V	Verification and Validation
VIP	Vessel and Internals Program
VTT	Technical Research Centre of Finland
W	Westinghouse
WW or W/W	Wetwell
XRD	X-Ray Diffraction
XRF	X-Ray Florescence
1F1	Fukushima Daiichi Unit 1
1F2	Fukushima Daiichi Unit 2
1F3	Fukushima Daiichi Unit 3
1F4	Fukushima Daiichi Unit 4

U.S. Efforts in Support of Examinations at Fukushima Daiichi - 2020 Evaluations

1. INTRODUCTION

The Great East Japan Earthquake (GEJE) of magnitude 9.0 and subsequent tsunami that occurred on March 11, 2011 led to a multi-unit severe accident at the Fukushima Daiichi Nuclear Power Station (Daiichi). Much is still not known about the end-state of core materials in each unit that was operating on that date. Some of this uncertainty can be attributed to a lack of information related to cooling system operation and cooling water injection during the events. There is also uncertainty in our understanding of phenomena affecting: a) in-vessel core damage progression during severe accidents in boiling water reactors (BWRs) [compared with Pressurized Water Reactors (PWRs)], and b) accident progression after vessel failure (ex-vessel progression) for BWRs and PWRs. These uncertainties arise due to limited full scale prototypic data. Similar to what occurred after the accident at Three Mile Island Unit 2 (TMI-2),[1] these Daiichi units offer the international community a means to obtain prototypic severe accident data from multiple full-scale BWR cores related to fuel heatup, cladding and other metallic structure oxidation and associated hydrogen production, fission product release and transport, and fuel/structure interactions from relocating fuel materials. In addition, these units may offer data related to the effects of salt water addition, vessel failure, containment failure, and ex-vessel core/concrete interactions (CCI).

Recognizing the importance of the information obtained from these units, the Department of Energy Office of Nuclear Energy (DOE-NE) initiated an effort in 2014 for U.S. experts in plant safety and operations to learn from this information. Since its inception, this effort has documented its findings and recommendations in annual reports and other publications [2 through 8]. For FY2020, it was decided that the program would gain more benefit from a more concise report that emphasizes new information and insights that affect prior findings and recommendations from the U.S. experts participating in this effort.

1.1. Objectives and Limitations

The DOE-NE sponsored effort for U.S. experts to participate in the Daiichi Forensics Evaluations has the following objectives:

- ***Objective 1:*** Develop consensus U.S. input for high priority time-sequenced examination tasks and supporting research activities that can be completed with minimal disruption of Tokyo Electric Power Company Holdings, Incorporated, (TEPCO Holdings) Decontamination and Decommissioning (D&D) plans for Daiichi.
- ***Objective 2:*** Evaluate obtained information to:
    - Gain a better understanding related to events that occurred in each unit at Daiichi
    - Gain insights to reduce uncertainties in predicting phenomena and equipment performance during severe accidents
    - Provide insights beneficial to TEPCO Holdings D&D activities

- Confirm and, if needed, improve guidance for severe accident prevention, mitigation, and emergency planning
- Update and/or refine Objective 1 information requests.

- ***Objective 3:*** Facilitate implementation of Japan-led international research efforts to support D&D.

There are several potential safety benefits from this U.S. effort. As documented here and in [2 through 6], the U.S. has already gained significant safety benefit from the information obtained from the affected units at Daiichi to reduce uncertainties in BWR severe accident progression and implement safety enhancements for BWRs, PWRs, and future nuclear power plant designs. As uncertainties in modeling the events at Daiichi are reduced, it not only improves guidance for accident mitigation but it informs future D&D activities by improving the capability to characterize potential hazards to workers involved with cleanup activities. Although there are many potential benefits from this U.S. effort, it is also important to recognize its limitations. As discussed below, other organizations have activities underway to address these limitations.

First, other organizations within the U.S. have the role of implementing institutional measures to ensure prevention of severe accidents. For example, the U.S. Nuclear Regulatory Commission (U.S. NRC) established the Fukushima Near Term Task Force (NTTF) and Japan Lessons Learned activities to ensure that appropriate near-term regulatory actions were taken in areas where additional efforts were required.

Second, within the U.S., the industry leads the implementation of safety measures in response to insights from Fukushima. For example, industry has implemented the diverse and flexible coping strategies or Diverse and Flexible Mitigation Capability (FLEX) program to address concerns related to events associated with extended loss of AC power (ELAP) conditions. In addition, both the PWR Owners Group (PWROG) and BWR Owners Group (BWROG) have implemented updates to their severe accident guidance (SAG) to address insights from the forensic effort. These guidelines will continue to be enhanced as further insights are gained from the ongoing work related to the Fukushima Accident.

Third, it is beyond the scope of the U.S. DOE Forensics Effort to develop an international program. However, it is recognized that information gained from Daiichi is of benefit to global nuclear reactor safety. A long-term international framework, led by Japanese organizations, may be appropriate for supporting post-accident examinations at Daiichi. As discussed in [2], organizations within Japan are initiating such international efforts. The U.S. Forensics Effort provides a means for U.S. experts to contribute to and benefit from such international efforts.

1.2. Motivation

Data, models, and insights from post-accident inspections at Daiichi inform many aspects of reactor safety, including severe accident modeling and simulation tools, severe accident management guidelines, plant staff training, and new or revised safety requirements in response to Fukushima. To increase the benefit from post-accident examinations that support D&D endeavors, this effort is needed to: (a) identify data needs to ensure that key information is not lost; (b) identify examination techniques, sample types, and evaluations to address each information need; and (c) when necessary, help finance acquisition of the required data and conduct of the analyses. Results from this effort are beneficial to the U.S. and to Japan.

For the U.S., this effort provides access to prototypic data from three units with distinctively different accident signatures. U.S. experts are interested in examination information with respect to:

- *Component Performance and System Survivability Assessments* - Examinations provide key information related to the performance of structures, systems, and components at each unit. For example, many improvements were made to plant instrumentation after the TMI-2 accident.[9] Similarly, the events at Daiichi provide information to better ensure that operators are able to assess the status of the plant and the effects of mitigating actions that may be taken.
- *Enhancements to Accident Progression and Source Term Models* – Similar to the processes that occurred with TMI-2 examinations,[10,11] knowledge gained from examinations at Daiichi is being used to reduce uncertainties in systems analysis codes, such as the Modular Accident Analysis Program (MAAP) code[12] and the Methods for Estimation of Leakages and Consequences of Releases (MELCOR) code.[13] These codes are used both domestically and internationally to evaluate the safety of operating plants, as well as new nuclear reactor designs.
- *Accident Management Strategies and Plant Staff Training* – As uncertainties in predicting BWR and PWR accident progression and associated source terms are reduced, strategies for mitigating severe accidents can be improved. Knowledge gained from Daiichi has and will continue to be factored into accident management guidance and staff training to prevent or reduce the consequences of future accidents in the operating fleet as well as in new reactor designs.
- *Preserving Severe Accident Capabilities* - Examinations provide important research opportunities that can serve as a springboard for rekindling much needed expertise within the younger generation of U.S. nuclear engineers regarding Light Water Reactor (LWR) severe accident behavior.

For Japan, U.S. involvement provides an independent evaluation of inputs to D&D activities. Such evaluations are useful because of U.S. experience with respect to:

- *Plant Operations* – The U.S. has over 20 operating BWRs and personnel with considerable experience with respect to BWR operations.
- *Reactor Safety* - Experts involved in this U.S. effort have a long history in developing severe accident systems analysis codes and large-scale experimental programs.
- *TMI-2 Post-Accident Examinations and Defueling* – Several U.S. experts participating in this program were also involved in TMI-2 post-accident evaluations. To facilitate exchange of this information, the U.S. DOE collaborated with the U.S. NRC to host a U.S.-Japan TMI-2 Knowledge Transfer and Relevance to Fukushima meeting in FY2017 to promote exchange of relevant information to cognizant organizations within Japan.

Unique U.S. expertise provides TEPCO Holdings an independent assessment of their progress reports, the adequacy of severe accident analysis code models for evaluations to support their D&D plans, and the adequacy of available examination information and proposed plans for additional examinations. In the latter case, U.S. input focuses on the desired amount of information, the resolution of data required from sampling, and the cost versus the benefit of obtaining such information. As discussed in [2], the U.S. devoted significant funding for extraction of radioactive samples of core debris from the TMI-2 vessel and evaluating these samples in hot cells. These efforts provided insights about the chemical composition and porosity of core debris. Although such evaluations from the core region improved our understanding of melt progression, it is less clear that results from relocated core debris samples obtain from the lower head were as beneficial. Conversely, additional samples to characterize the interface between relocated debris and the vessel head could have helped reduce uncertainties in characterizing heat transfer from relocated debris and the potential for vessel failure. Such insights are useful to Japan.

Because of the benefit to global nuclear reactor safety, it is recognized that an international framework is ultimately needed to support post-accident examinations. Japanese organizations should lead this international framework. Nevertheless, the U.S. has a vested interest in these examinations. The U.S. has the largest number of operating nuclear power plants in the world; there are also a significant number of reactors operating around the world based on U.S. plant designs. Hence, U.S. organizations – both industry and government—are major beneficiaries from any improvements in LWR severe accident knowledge just as Japan was a major beneficiary from their participation in prior international TMI-2 programs. Collaborative work with the international community to establish a Japan-led international framework is not only beneficial to the U.S. and Japan, but also offers the potential to advance reactor safety across the global nuclear energy community.

1.3. Approach

The approach developed to ensure that objectives outlined in Section 1.1 are achieved relies primarily on expert panel meetings. Over 30 experts from industry, universities, and national laboratories participate in this process. Experts from the U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research (U.S. NRC-RES), the U.S. Department of Energy Office of Nuclear Energy (U.S. DOE-NE), the Department of Energy Office of Environmental Management (U.S. DOE-EM), TEPCO Holdings, Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF), and the Japan Atomic Energy Agency (JAEA) also attend and inform participants during these meetings.

1.3.1. Objective 1 Activities

To complete Objective 1, expert panel meetings initially focused on developing a report during FY2015 with a prioritized initial list of information of interest to U.S. stakeholders.[6] In this report, special attention was devoted to identifying why such information is important and how it will be used to benefit the U.S. nuclear enterprise.

During initial meetings to complete Objective 1, U.S. experts agreed upon several significant findings:

- Information obtained from the affected reactors at Daiichi offers a unique means to obtain full-scale, prototypic data for enhancing reactor safety (e.g., improved severe accident guidance, possible plant modifications, improved simulation codes for staff training, etc.).
- Insights gained from collecting and comparing similar observations and data from each of the three units are valuable because the accident progression at each unit was unique in many respects.
- This information is important for BWRs and PWRs; i.e., many insights gained from this information are not only applicable to BWRs, but also could have significant safety impacts on PWRs.
- Some information is required for all identified items to obtain a complete picture of the events. It is only meaningful to prioritize data needs with respect to the 'cost' and 'logical sequence' for obtaining such information.
- Information from other units at Daiichi and other plants, such as Fukushima Daini, also provide valuable insights for forensics, repair, maintenance, and field applications. Critical information

from these plants can be more easily obtained at lower cost and with less radiation exposure to personnel.
- D&D plans (or activities already completed) address much of the information identified by the U.S. expert panel.
- Maximum benefits from this information requires: reviews by cognizant experts, posting for easy-to-use access, interactions with TEPCO Holdings for added requests and understanding of information available, and interactions with code assessments.
- Ultimately, an international framework should be established to benefit from information obtained during D&D efforts at Daiichi.
- Important information and data are already available, and more is being gathered at the current time. U.S. forensics evaluation tasks should be initiated as soon as possible.

Most of the information needs identified by the expert panel are related to Daiichi Units 1 through 4 (1F1, 1F2, 1F3, and 1F4).* Although details varied, U.S. experts generally identified needs required to answer fundamental questions related to how the accident progressed in each unit, to understand equipment and component survivability, and to benchmark severe accident progression and dose assessment codes. These needs are organized in tables per location [e.g., the reactor building (RB), the primary containment vessel (PCV), and the reactor pressure vessel (RPV)]. These tables also identify applicable units for each need and other relevant factors (e.g., how information should be obtained, why it is needed, its expected use or benefits, when it should be obtained, and the estimated level of effort).

Table 1-1 summarizes, at a high level, the activities identified by the expert panel for addressing information needs from the affected units at Daiichi. As indicated above, the expert panel concluded that some information is needed from all locations to obtain a complete picture of the entire accident progression in each unit. Therefore, experts concluded that information needs were best prioritized with respect to cost and the logical sequence for obtaining such information. For each location, Table 1-1 groups the desired examination information by method and specifies the priority of the information need by the number of asterisks in each box. Results indicate that the expert panel typically placed the most emphasis upon information obtained from visual examinations, such as videos and photographs, and near-term proximity exams, such as dose surveys. Experts agreed that such information was the easiest to obtain and could provide critical information related to whether additional examinations were required.

Another important conclusion is that much information is already available. As discussed in Section 1.3.2, Objective 2 activities evaluate available information and make revisions as appropriate.

* Only Units 1, 2, and 3 (1F1, 1F2, and 1F3) were operating on March 11, 2011. Because of the hydrogen explosion damage observed at Unit 4 (1F4), this unit is also of interest.

Table 1-1. Prioritization of possible examination activities [6]

Region	Examination Information Classification[a,b]			
	Visual	Near-Proximity	Destructive	Analytical
Reactor Building (RB)				
Reactor Core Isolation Cooling (RCIC) or Isolation Condenser (IC)	****	***	**	
High Pressure Coolant Injection (HPCI)	****		***	
Building	****	***	**	*
Primary Containment Vessel (PCV)				
Main Steam Isolation Valves (MSIVs) and Safety Relief Valves (SRVs)	****		***	
Drywell (DW) Area	****	***	**	*
Suppression Chamber (SC)	****	***		
Pedestal / RPV-lower head	****		***	**
Instrumentation		****	***	
Reactor Pressure Vessel (RPV)				
Upper Vessel Penetrations	****		***	**
Upper Internals	****	***	**	*
Core Regions & Shroud	****		***	**
Lower Plenum	****		***	**

a. Examination Classification Examples:
Visual– Videos, Photographs, etc.
Near-Proximity– Radionuclide Surveys, Seismic Integrity Inspections, Bolt Tension Inspections, and Instrumentation Calibration Evaluations
Destructive– System or Component Disassembly, Sampling, etc.
Analytical– Chemical Analysis, Metallurgical Analysis, Gamma Scanning, etc.

b. Prioritization based on number of asterisks, e.g., more asterisks designate a higher priority on this information.

1.3.2. Objective 2 Activities

Activities used to complete the second objective are shown in Figure 1-1. As shown in this figure, activities and products completed by U.S. organizations focus on Phase 2 Activities associated with the Mid-and-Long-Term Roadmap for D&D (the blue box). As indicated by the gray box, severe accident and plant operations experts from U.S. industry, universities, and national laboratories evaluate plant examination information obtained from Daiichi. Since its origin, the forensics effort has striven to include a broad spectrum of U.S. stakeholder input. Objective 2 activities are also informed by experts from the U.S. NRC, U.S. DOE, and TEPCO Holdings that participate in expert panel meetings.

Activities and products completed by U.S. organizations are shown in green. Severe accident and plant operations experts evaluated information from five higher priority topic areas identified by the panel. These areas are:

- Component/System Performance
- Radiological Sampling and Surveys
- Core Debris End-state
- Combustible Gas Effect†
- Operations and Maintenance‡

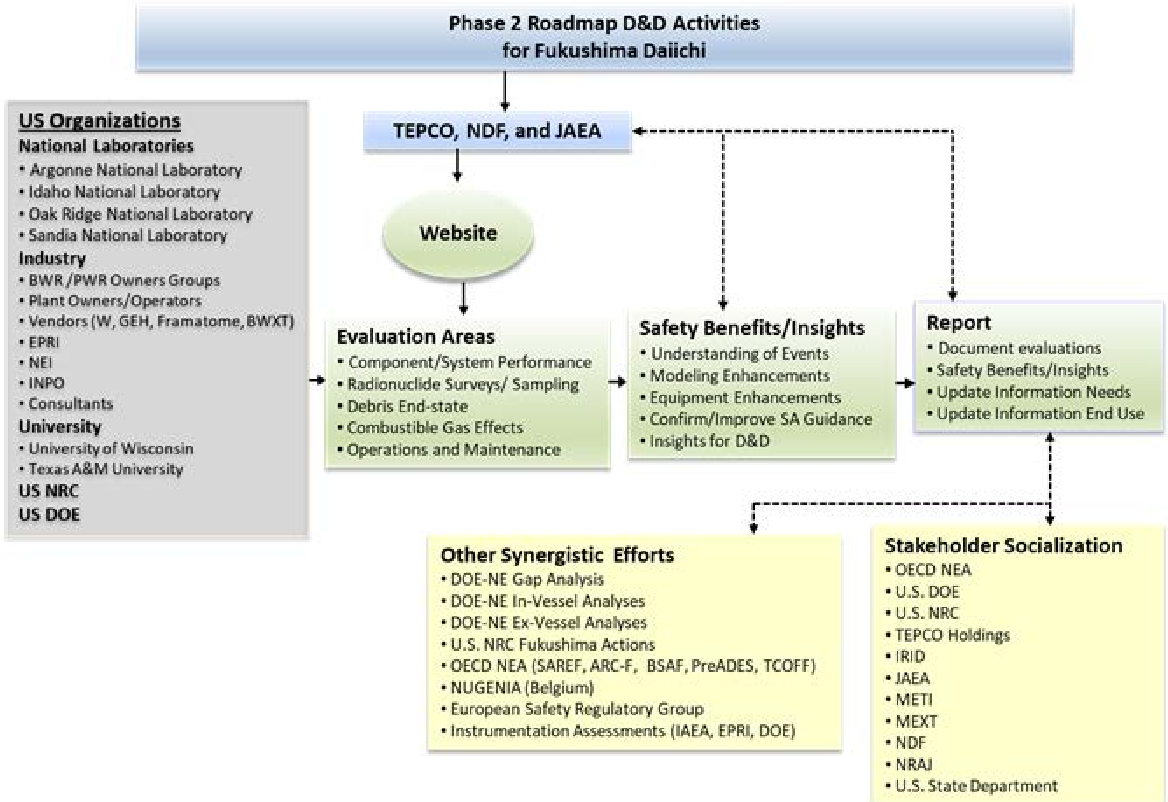


Figure 1-1. Objective 2 activities.[a]

a. See Acronyms for definitions of organizations and programs.

The fifth area, “Plant Operations and Maintenance,” covers a range of topics of interest to industry, such as instrumentation survivability information obtained from Daiichi examinations and practical insights from D&D that can be used to enhance radiation safety for the existing fleet.

The primary source of information used in U.S. Forensics Effort evaluations is information provided on websites from TEPCO Holdings[14] and other Japanese organizations, including NDF, the Government of Japan, the Ministry of Health, Labor, and Welfare, the Ministry of Economy, Trade and Industry (METI), and the Nuclear Regulatory Authority (NRA or NRAJ). Each year at Forensics Effort meetings, presentations based on this information are provided by representatives from TEPCO Holdings, NDF, JAEA, U.S. industry, and topic area leads. TEPCO Holdings reports documenting unconfirmed and unresolved issues also receive special attention in the forensics effort.[15 through 20] The website created by the Institute of Applied Energy (IAE)[21] is also an important reference for this effort. In addition, as discussed in [2], a website has been developed by this program to archive key references used by U.S. experts to complete these evaluations.

As previously discussed, these evaluations lead to several types of safety benefits and insights:

- Increased understanding of the events that occurred at each of the affected units at Daiichi

† This fourth area was added in FY2016.

‡ This fifth area was added in FY2018.

- Enhanced severe accident analysis models (reduced severe accident modeling uncertainties)
- Increased understanding of equipment performance during severe accidents
- Confirmed / improved guidance and training for severe accident prevention, mitigation, and emergency planning
- Additional insights beneficial to future D&D activities

As shown in Figure 1-1, U.S. experts prepare a report documenting results from these evaluations and updates related to the U.S. information requests for additional examinations. For the first five years of this effort, these reports were substantive in order to capture insights associated with information coming from the affective units. For each area, prioritized questions of interest were identified; available information was reviewed; and insights gained from evaluating this information was provided. Where appropriate, information requests were updated, and a complete list of information requests that includes these updates was developed. Additional details, such as the benefits, use, and suggested methods for obtaining higher priority, near-term examination activities were provided. For FY2020, it was decided that the program would gain more benefit from a more concise letter report that emphasizes new information and insights that affect changes to findings and recommendations from the U.S. experts participating in this effort. A key aspect of prior U.S. efforts, the updated list of information requests, is still included in this concise letter report.

1.3.3. Other Considerations

In completing Objective 2 activities, there are other considerations (shown in yellow boxes in Figure 1-1). These other considerations are important aspects of this forensics effort. The first consideration relates to other synergistic efforts, including those funded by DOE, those completed by NRC, and those organized by other agencies and other organizations. In addition, results from this U.S. effort support several aspects of these synergistic efforts. These other considerations are described in [2].

1.4. Report Objectives and Organization

As noted above, this FY 2020 letter report focuses upon new information and insights that affect changes to findings and recommendations from the U.S. experts participating in this effort. Section 2 provides an overview of presentations and discussions occurring during the FY2020 meeting, held in Washington, DC, November 18-19, 2019. Section 3 highlights findings and recommendations from these meetings and changes to key insights and recommendations documented in the FY2019 report. References for this letter report are listed in Section 4. Appendices to this document provide more detailed information. Specifically, Appendix A provides lists of attendees and agendas from the U.S. Forensics Effort expert meetings held during FY2020. Appendix B provides updated tables with detailed information requests developed by U.S. experts and additional details for high priority, nearer term examination activities. Appendix C includes presentations from participants wishing to include them in this publication.

2. FY2020 EXPERT PANEL MEETING HIGHLIGHTS

This section highlights presentations and discussions that occurred during the FY2020 Expert Panel meeting for the US DOE sponsored Forensics Effort. Appendix A includes an agenda and list of participants attending this meeting, which was held on November 18 and 19, 2019 at the Argonne National Laboratory offices in Washington, DC. Appendix C includes presentations from participants wishing to include them in this publication. Highlights from the meeting are organized into three categories: presentations from Japanese organizations (Section 2.1), presentations for topic areas identified by US organizations (Section 2.2), and other topics of interest (Section 2.3). As part of the presentations for each topic area, updates to key items found in prior reports, such as key questions of interest and tables summarizing recent examination insights, and recommendations for additional information requests, are provided as appropriate.

2.1. Presentations from Japan

2.1.1. Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF)

There were two presentations by representatives from NDF (see Appendix C.1.1.1 and C.1.1.2): Hiroji Wakabayashi provided an overview of the 2019 strategic plan for decommissioning of Fukushima,[22] and Junichi Nakano provided an overview of the philosophy adopted by Japan for debris examinations.

2.1.1.1. 2019 Strategic Plan for Decommissioning

In his overview, Mr. Wakabayashi emphasized Japan's policy to continuously and quickly reduce radioactive risks (See Slide #5 in Appendix C.1.1.1) in the areas of fuel debris retrieval, waste management, water management, and Spent Nuclear Fuel (SNF) removal. Using a 'step-by-step' approach, activities continue to characterize and minimize risks. It is currently recommended that the first unit in which fuel will be removed is 1F2. Mr. Wakabayashi emphasized several aspects about the planned debris removal strategy:

- 1F2 was selected to be the first unit for debris retrieval because its relatively lower radiation levels and its higher 'airtightness' and because it offers the potential to optimize decommissioning work. As side entry is planned using a partial submersion method. It is planned that this work will start in 2021.
- Fuel debris retrieval will start on a small-scale basis by methods such as gripping and suction.
- Next steps for fuel debris retrieval will be based on insights gained from information and experience accumulated through initial activities.
- Retrieved fuel debris will be transferred to the on-site temporary dry storage facilities at Daiichi.
- The method for expanded scale of fuel debris retrieval will be determined by engineering evaluations that includes safety assessments based on progress of research and development, PCV internal investigations, improvement of conditions at the site and information and experience accumulated through previous operations.

At this time, it is unclear whether decommissioning activities in other units will proceed in parallel or in series. There are concerns about long-term removal equipment survivability and human resource issues for parallel activities.

Mr. Wakabayashi also provided an overview of plans for removing SNF from the pools in each unit (see Slide #13 in Appendix C.1.1.1). In his presentation, he emphasized the following activities:

- 1F1- Plans emphasize careful removal of rubble and continued implementation of measures to prevent radioactive dust dispersion during removal
- 1F2- In addition to the conventional method of completely dismantling the upper part of the operating floor, a method of accessing from the south side of the reactor building without dismantling is being considered
- 1F3-SNF removal began this April and is expected to be completed by the end of FY 2020

During his presentation, Mr. Wakabayashi emphasized safety concerns due to evacuation orders being lifted and residents returning to municipalities near the region.[22]

2.1.1.2. Fundamental Concept for Fuel Debris Analysis in Japan

Mr. Nakano reviewed the principles for prioritizing evaluations of fuel debris that have been adopted by Japan (See Slide #2 in Appendix C.1.1.2). While safety and efficient decommissioning of Daiichi is the highest priority, the principles recognize that it is also important to understand the cause of the accident and that obtained data offer the potential to improve global safety of nuclear power (e.g., changes to accident management strategies and improvements in models for predicting severe accident progression). He reviewed some areas where information from Daiichi could provide important insights, such as BWR accident progression uncertainties, addition of salt water, etc. (see Slides #7 through #11 in Appendix C.1.1.2), but also observed that the material will be non-homogeneous. Hence, it will be important to consider the location from which the debris sample was obtained in evaluating debris analysis results. It may not be possible to obtain the required number of samples to reduce uncertainties if one considers the cost and resources needed (see Slide #12 in Appendix C.1.1.2).

2.1.2. Tokyo Electric Power Company Holdings, Inc. (TEPCO)

The presentations by Shinya Mizokami covered several topics, including the most recent entry into 1F2, cooling water suspension tests in 1F1, and uncertainties in modeling selected aspects of the events at Daiichi.

- The presentation, “Current status and recent investigation result of Fukushima Daiichi,” [Appendix C.1.2.1] provides an overview of recent insights from completed investigations within each unit and plans for future examinations. It describes the devices used to complete the 1F2 entry and a video taken within 1F2 was provided. TEPCO estimates that the debris depth in the cavity is between 40 to 60 cm. It seems that the debris depth is deeper where water is raining down. The rain comes from about ¼ of the perimeter. Prior 1F2 investigations suggest that much of the upper surface of relocated material is composed of pebble-like particles (less than 8 cm in diameter) that is loosely aggregated and fairly easy to pick up and retrieve.[See Slide #10 in Appendix C.1.2.1]

This presentation also describes on-going activities for another investigation into the 1F1 PCV. Six different robot concepts will be used for visual examinations and sample retrieval. In 1F1, they are cutting a hole using a high-pressure water jet for initial access in the X-2 penetration. In attempting to cut the hole, contaminated airborne dust (radioactivity) in the PCV exceeded approved limits so some delays have been incurred. The explosion in the 1F1 reactor building suggests that leakage from the PCV occurred during this event. Hence, there is the potential that radiation may be released with this dust. A temporary dust monitor was installed at this location because there is no filter. The presentation also included several photos of the reactor well interior surface and PCV head flange. No obvious PCV flange deformations were observed, which is consistent with the hypothesis that elastic stretching of the head bolts occurred.

- The presentation, "Reactor Cooling Water Temporary Suspension Test at Unit 1 - Rapid Communication," [See Appendix C.1.2.2] reported findings from tests involving cessation of water injection to the1F1 RPV. These test results suggest two leak paths from the drywell; i.e., one smaller PCV liner failure to sand cushion and one leakage path through the vacuum breaker piping at the top of the torus (Slide #8 in Appendix C.1.2.2). The latter leak increases PCV pressure when the water level rises above the vacuum breaker and likely seals the gas leakage path. TEPCO has used this to estimate the hole size. TEPCO also believes that these findings suggest that there is currently no leakage in the 1F1 PCV head.
- The presentation, "Findings on Fukushima Daiichi NPP Severe Accident and Implication to SA Code Validation,"[C.1.2.3] was a good exposition of the uncertainties in severe accident modeling with examples, such as loss of information regarding equipment functionality (ADS valve actuation), equipment performance outside its design range (RCIC performance at low void conditions), lack of knowledge on a transition of system physical state (RPV lower plenum hole by CRD hole- Slide #46 in Appendix C.1.2.3 identifies potential failure locations corresponding to partial number on fuel assembly tie handle observed in 1F2 investigation), and uncertainty in ablation by Molten Core Concrete Interactions (MCCI) at 1F3. During his presentation, Dr. Mizokami emphasized the importance of on-going RCIC testing (see Section 2.2.5) and the Reduction Of Severe Accident Uncertainties (ROSAU) tests (see Section 2.2.3) to reduce uncertainties.

As in prior years, Dr. Mizokami's presentation is very detailed and rich with new information.

2.1.3. Japan Atomic Energy Agency (JAEA)

Akira Nakayoshi, JAEA, provided slides about the Preparatory Studies for Fuel Debris Analysis (PreADES) project, a JAEA-led OECD project to prepare Japan for upcoming analyses of fuel containing debris samples. These slides, which were presented by Joy Rempe, review the objective, motivation, approach, and schedule for completing the three project tasks. At this time, 15 organizations from 7 countries participate in PreADES. In addition, representatives from the Institute for Safety Problems of Nuclear Power Plants (ISP NPP), who are involved with stabilization of the Chernobyl Unit 4 reactor, attend project meetings. Task 1, which is nearly complete, focuses on the debris endstate for 1F1, 1F2, and 1F3 and identifies relevant data for characterizing debris based on prior evaluations of debris from TMI-2, Chernobyl Unit 4, and larger scale experiments using prototypic materials. Task 2 uses results from Task 1 to identify where additional data are required and then prioritizes such data gaps based on their importance to safety issues during defueling, transportation, examinations, and storage. Task 3 will focus on planning future international research and development to address prioritized data needs. During this presentation, Dr. Rempe emphasized areas where contributions from the U.S. such as the examination requests from the Forensics Effort and a report listing U.S. hot cell capabilities, are being used by the PreADES project.

2.2. Topic Areas

2.2.1. Topic Area 1 - Component/System Performance

Leads, Jeff Gabor, Jensen Hughes, and Kevin Robb, Oak Ridge National Laboratory (ORNL) provided an update of recent examination information that addressed key questions of interest to Topic Area 1:

- What visual damage has been observed in component and structures within the RPV, PCV, RB?
- What plant data support damage assessment?
- What insights are gained from damage assessment (e.g. peak temperatures, pressures, and radiation levels)?
- Can insights be used to enhance reactor safety and SA guidance?
- Are analysis improvements needed?

During their presentation, area leads noted that the third presentation by Shinya Mizokami (see Section 2.1.2) was of particular interest to Topic Area 1. In particular, leads and participants expressed interest in insights that could reduce uncertainties in safety margins that affect long-term cooling and water addition strategies. In their presentation, area leads highlighted the following information from recent examinations:

- **1F1 Shield Plug Examinations.** Dr. Robb (see Slides #9-14 of C.2.1) reviewed recent information posted by TEPCO regarding the measured deformation or sagging in the shield plug. It was not clear that this deformation was due to duress on the concrete as the accident evolved. The current configuration of the 3-layered shield plug could also be due to pressure differences that occurred during venting of the PCV. The dose rate in the shield plug region is consistent with PCV venting.
- **1F1 Water Injection Termination** - Dr. Gabor reviewed the water injection experiment that was also described by Dr. Mizokami (see Section 2.1.2). He noted the increases in RPV lower head and PCV temperature response were minimal as water injection was suspended and that similar tests are planned for 1F3 in March 2020.
- **1F2 RCIC Operation** - Dr. Gabor also reviewed information regarding 1F2 RCIC performance presented by Dr. Mizokami. It was noted that the 1F2 RCIC was restarted only 2 minutes before the tsunami led to a loss of DC power in this unit. Had the RCIC not been operating at the time that the DC power was lost, it could not have been started and would have put the 1F2 reactor on a shorter timeframe to core damage. Recall in 1F1, the isolation condenser valves were closed at the time of loss of DC power to prevent over-cooling of the vessel. If the valves had been left open, the 1F1 accident progression might have been less severe as cooling could have been maintained for several more hours.
- **1F2 Fuel Bundle Handle** - There was a lot of discussion about the 1F2 fuel bundle bail handle that is partly submerged in cavity debris, but still quite recognizable. It was concluded that this component must have fallen during the core meltdown since it is intact but partly submerged in re-solidified melt on the floor of the cavity. It was observed that many code models are based on TMI-2 experience; some of which may not be applicable to what occurred at 1F1, 1F2, and 1F3.

During the Topic 1 discussion, participants agreed that the systems analysis codes have demonstrated a good ability to capture the main initial trends of the accident progressions, including nuanced differences between the 1F1, 1F2, and 1F3 damage sequences. However, there are still areas, such as RPV lower head failure mechanism and RCIC operation in two-phase flow, where larger uncertainties remain.

Area leads did not propose any changes to Topic Area 1 recommendations (see Section 3 of Reference [2] or the U.S. list of examination requests related to Topic Area 1 in Appendix C of Reference [2]). Recent examination results, however, led topic area leads to update the Reference [2] summary table of examination information pertaining to component and system performance (see Table 2-1).

Table 2-1. Results from component and system examinations[a]

Area	1F1	1F2	1F3
X-100B PCV penetration[b]	Possible melted shielding material [23] No damage observed on outside [24]	NA	NA
X-51 PCV penetration[c]	NA	No damage observed; pressurized water could not penetrate blockage in standby liquid cooling system line [25, 26]	NA
X-53 HPCI steam supply penetration (1F2/1F3)[d]	High dose rate measured [27]	No damage observed [28]	No damage observed [29]
X-6 PCV penetration (CRD hatch)	NA	Melted material [30, 31]	No damage observed from inside [32]
Equipment hatch	NA	NA	Water puddle [33, 34] unknown source
Personnel hatch and nearby penetrations	No damage observed [35]	NA	NA
HPCI pipe penetration[e]	No damage observed, but high dose rates measured; traces of flow and white sediment observed [27, 35,36]	NA	NA
TIP room	No leakage observed from PCV through TIP guide penetrations. Relatively high dose rates measured near other primary system instrumentation penetrations (X-31, X-32, X-33) [27,37]	Dose surveys do not indicate leakage from PCV through TIP guides. High dose levels in samples of materials from TIP indexer [38]	NA
WW vacuum breaker line	Leakage on expansion joint of one line (X-5E) [39]	NA	NA
DW/WW vent bellows	Water leakage attributed to vacuum line above [39]	No leakage observed [40]	
DW sand cushion drain pipe	Leakage [41]	No leakage observed [40]	NA
SC water level	Almost full [20]	Middle [20]	Full [20]
DW Water Level	~2 m[20]	~0.2 m[20]	~6 m[20]

Table 2-1. Results from component and system examinations[a]

Area	1F1	1F2	1F3
Torus room	Partially flooded [42, 43]	Partially flooded [44]	Partially flooded [44]
	Rusted handrails/equipment [23]	Non-rusted handrails/ equipment [23,45]	Non-rusted handrails/ equipment [23,46]
	NA	Some room penetrations tested, no leakage observed [47]	NA
MSIV room	Limited view obtained [48]	Water leakage cannot be observed [49]	Leakage in Line D near bellows [50]
DW shield plugs	Reactor well shield plug displaced [51]	Possible leakage [52]	Leakage likely due to radiation measurements at head and presence of H_2 burn [20,53]
DW head/flange	No obvious PCV flange deformations observed; but elastic stretching of bolts during event possible [Appendix C.1.2.2]	NA	NA
RCIC or other low SC piping	NA	Suspected leak location, not confirmed [23]	NA
RPV upper head	NA	NA	NA
RPV lower head	Ex-vessel debris images, dose surveys, and sample examinations indicate failure [20,54,55]	Ex-vessel debris and images confirm failure [53]	Ex-vessel debris images confirm failure [53]

a. Nomenclature: [Clear]: NA; no information available; [Red]: available information indicates damage or leakage; [Orange]: available information suggests possible damage; [Green]: available information indicates no damage. See Acronyms for other abbreviations.
b. X-100B is vacant for 1F1, allowing this penetration to be used for DW investigations.
c. X-51 is an instrument pipe penetration for measuring differential pressure in 1F2/1F3. The penetration is joined to the Standby Liquid Cooling (SLC) pump injection line in the DW. This penetration is designated as X-27 in 1F1.
d. X-53 is vacant for 1F2 and 1F3, allowing these penetrations to be used for DW investigations.
e. X-53 is the HPCI steam supply penetration and X-54 is the HPCI steam instrument pipe penetration for 1F1. X-11 is the HPCI steam supply penetration for 1F2 and 1F3.

2.2.2. Topic Area 2 - Radionuclide Surveys and Sampling

Topic Area Leads, David Luxat and Nathan Andrews, Sandia National Laboratories (SNL) reviewed recent information from examinations at Daiichi and concluded that there was no need to provide any presentations at this meeting. Area leads did not propose any changes to Topic Area 2 recommendations (see Section 4 of Reference [2] or the U.S. list of examination requests related to Topic Area 2 in Appendix C of Reference [2]).

2.2.3. Topic Area 3 - Debris Endstate

The lead for Topic Area 3, Mitch Farmer, Argonne National Laboratory (ANL) provided a presentation on recent insights from Fukushima related to MCCI and debris coolability. The presentation began with a summary of insights from the Severe Accident Water Addition (SAWA)/Severe Accident Water

Management (SAWM) study that was completed last year as part of the DOE-NE Light Water Reactor Sustainability program and carried out with the MELTSPREAD3 and CORQUENCH4 codes using results from the MELCOR and MAAP codes. The study was based on a Peach Bottom BWR, which is housed in a Mark I containment similar to the 1F1, 1F2, and 1F3 units at Fukushima Daiichi. Many of the findings from this SAWA/SAWM study were relevant to interpreting the sequence of events during the accidents.

Regarding 1F1, Dr. Farmer first reviewed examination information regarding its debris endstate:

- Based on limited robotics examinations inside the PCV as well as muon tomography, most of the core inventory is believed to have exited the RPV and reside in the pedestal/drywell regions.
- The presence of significant accumulations of material in the drywell outside the pedestal doorway (~0.8-1.0 m) has been identified as evidenced by persistent water leakage from the sand cushion area.
- At the X-100B location, ~130 degrees from the pedestal doorway, material ~30 cm deep has been found. Covered by loose sediment, it is not currently known how far down the loose sediment extends and whether the sediment covers other material (e.g., fuel containing debris from the core).
- There is visual evidence suggesting that the PCV liner has failed (e.g., images showing that the sand cushion drain line is leaking).
- The presence of core debris in pedestal region is consistent with MELCOR/MAAP/MELTSPREAD/CORQUENCH studies.

Dr. Farmer observed that recent TEPCO information indicates that effectively, there may not have been any water injection into the RPV for the first ~12 days of the accident due to possible valve misalignment and uncertainty in plumbing configurations. Dr. Farmer, however, noted the extent of damage observed to date inside the PCV does not appear to be consistent with dry MCCI occurring over this time interval. This conclusion was drawn based on observations in tests conducted at Argonne, as well as the extent of damage to the reactor building observed at Chernobyl Unit 4.

Regarding 1F2, Dr. Farmer reviewed selected muon tomography and robotic exams information:

- Initial access through the X-6 penetration revealed the presence of significant core debris retention on the CRD access platform.
- Robot entries revealed debris accumulation in the pedestal region that ranges 40 to 70 cm thick, which is well above the water height of 30 cm in the drywell.

Despite the extent of this relocated material, Dr. Farmer observed that there did not appear to be significant damage to structures within the pedestal region, at least at locations near the upper surface of the debris. He attributed this to:

- Significant water present on the pedestal floor when the vessel failed, and/or
- Debris in the pedestal region being predominately lower temperature metallics with lower fission product (decay heat) content.

The latter observation raised questions from participants regarding the potential for uranium metal or cesium to be in the metallic fuel and how much debris coolability was affected by the metal/oxide mixture in relocated debris. However, Dr. Farmer emphasized that the major observation relative to reactor safety is that the relatively deep accumulation of core material in the reactor pedestal (ranging from 40 to 70 cm) has provided evidence that the core debris was coolable by water ingression and that water injection through the vessel is the preferred route. Injection through the vessel ensures that water will flow over and cool underlying core debris, even if the height of the material is greater than the downcomer inlet to the torus.

Regarding 1F3, Dr. Farmer reviewed muon tomography and robotics examination results:

- The CRD platform is dislodged from the rails and a portion of it is buried under core debris.
- The depth of the deposits is largest in the center of the pedestal and falls off as the pedestal wall is approached. This trend is consistent with lower head failure near the centerline, as opposed to 1F2 for which data suggest that the lower head failed near the periphery
- Recent TEPCO renditions of debris endstates suggest that the debris is quite deep; i.e., in the range of 2-3 meters. Dr. Farmer completed 'back-of-the-envelop' calculations (see Slides # 18-22 in Appendix C.2.3) that the mass of debris to be as much as 270 MT if it is in a relatively dense condition (lower if there is significant porosity). Dr. Farmer observed that the material depths were beyond that which would likely be coolable based on the existing water ingression correlation assuming that the core debris was quenched using top flooding from a once molten condition. Further discussion explored the possibility that the material may have formed as a result of corium jet fragmentation (breakup) in water and subsequent accumulation on the pedestal floor (if that condition existed when the reactor vessel failed). In effect, this would resemble the Swedish severe accident mitigation strategy of flooding the reactor pedestal with a deep water pool prior to vessel failure to enhance debris fragmentation during relocation and coolability (without generating fine fragments that could result in energetic fuel coolant interactions). Dr. Farmer emphasized the need to characterize the debris in the pedestal to the greatest extent possible, as the observation that this deep accumulation of material could be cooled by top flooding was significant for reactor safety evaluations.

Dr. Farmer closed his presentation by providing an overview of the OECD Reduction Of Severe Accident Uncertainties (ROSAU) program. The objective of ROSAU is to address two knowledge gaps in LWR severe accident progression identified following the events at Fukushima Daiichi:

- Coolability of high metal content (BWR-type) core debris, and
- The effect of water on core debris spreading following vessel failure.

At present, sixteen organizations, including NRC and EPRI, from eight OECD countries have joined the ROSAU project. Additional OECD countries and organizations will likely join the project in the future. The project was formally launched with a project kick-off meeting held in September 2019.

In summary, Dr. Farmer did not propose any changes to Topic Area 3 recommendations (see Section 5 of Reference [2] or the U.S. list of examination requests related to Topic Area 3 in Appendix C of Reference [2]). Recent examination results regarding the height of debris observed in 1F3, however, led Dr. Farmer to emphasize the importance of characterizing the debris in the pedestal region to greatest extent possible to discern characteristics affecting debris coolability.

2.2.4. Topic Area 4 - Combustible Gas Effects

In his presentation, Wison Luangdilok, Fauske and Associates, LLC (FAI) and H2 Technology, LLC, presented recent research that he had performed using available information related to the hydrogen explosions at 1F1, 1F3, and 1F4. Dr. Luangdilok summarized his research and literature collection and explored reasons for difference in apparent kinetic energy resulting from the 1F1 and 1F3 explosions. He provided extensive hydrocarbon-based fireball explosions data and used that as a basis for estimating the 1F3 explosion energy. He also discussed the significance of the 1F3 fireball and explored the various amounts of hydrogen predicted by codes in the Benchmark Study of Accident at the Fukushima Daiichi Nuclear Power Plant 1 and 2 (BSAF and BSAF2) project efforts. He concluded that (1) about 1450 kg of hydrogen was needed to explain the observed 1F3 fireball, and (2) about 2680 kg of hydrogen equivalent (from oxidation of core components and MCCI) must be generated in 1F3 in order to explain the migration of hydrogen during venting from 1F3 to 1F4 and the subsequent 1F4 explosion. He emphasized that models in system analysis codes were not predicting sufficient combustible gas generation to result in the observed explosions. He closed his presentation discussing possible areas (e.g., significantly higher oxidation kinetics associated with eutectic melts that uniquely form in BWR systems, formation of these eutectics, formation of protective oxide layers on the channel box to prevent attack by B_4C/SS melts, degradation of such channel box protective layers, etc.) where systems analysis code models might be revised. In the discussions following his presentation, some attendees questioned whether it was possible to explain the observed 1F3 explosion if smaller masses of hydrogen equivalent accumulated in a higher concentration within a localized region. However, several attendees observed that models may be under-predicting combustible gas generation.

Dr. Luangdilok did not propose any changes to Topic Area 4 recommendations (see Section 6 of Reference [2] or the U.S. list of examination requests related to Topic Area 4 in Appendix C of Reference [2]). However, he recommended that models in system analysis codes be reviewed and revised, if needed, to consider phenomena, that could not only increase combustible gas production to levels that lead to the combustion events observed at Daiichi but could also improve predictions of other severe accident progression phenomena (core heatup, relocation, vessel failure, ex-vessel relocation and interactions).

2.2.5. Topic Area 5 - Operations and Maintenance

As emphasized in Tables 2-4 and 2-5 of the 2019 Forensics Effort report [2], owners groups have used insights from forensics examinations to update guidance and support procedures for severe accident prevention and mitigation. Representatives from the BWR Owners Group (BWROG) and PWR Owners Group (PWROG) provided three presentations related to this topic: Bill Williamson provided an update on BWROG Emergency Procedures Guidelines; Kyle Shearer provided an update on PWROG Procedures; and Randy Bunt provided an update on a project, led by the BWROG, to investigate Terry™ Turbine performance. In addition, Nathan Andrews, SNL, provided an update on recent testing completed in support of the Terry Turbine project.

In his presentation, Mr. Williamson emphasized the following topics:

- Status report on implementing the Emergency Planning Guideline (EPG)/Severe Accident Guideline (SAG) procedure changes to the BWR fleet based on lessons learned from investigations at Daiichi;
- Implication for operations, maintenance, severe accident mitigation, and accident analysis
- Computer Based Training (CBT) using the Institute of Nuclear Power Operation (INPO) training system (the National Academy for Nuclear Training e-Learning or NANTeL system), and
- Instrumentation practical insights based on information from Daiichi.

One of the higher priority goals of the BWR SAG education efforts is to raise the overall level of comprehension of severe accidents and their key signatures as well as the implications associated with plant damage states. Discussion regarding the status of the Technical Support Guidance (TSG) tool-set emphasized difficulties that the BWROG had experienced in their efforts to secure funding from the DOE Light Water Reactor Sustainability (LWRS) program to complete this effort as originally planned.

In his presentation, Mr. Shearer emphasized the following topics:

- Severe Accident Management Guideline (SAMG) maintenance program
- Risk beneficial procedure changes program
- Long term containment venting strategy.

The maintenance program for the SAMGs allows issues to be identified, tracked, prioritized, and resolved. Although there is a backlog, most are editorial. Mr. Shearer also discussed that the PWROG expects that changes in the maintenance rule will result in less stringent quality assurance requirements on FLEX equipment (because this equipment is primarily to provide additional defense-in-depth). With respect to the new strategy for long-term venting, Mr. Shearer noted that this strategy provides for long-term control of combustible gas generation (after Passive Autocatalytic Recombiner System or PARs installed in international PWRs are no longer effective). In summary, Mr. Shearer observed that the PWROG is continuing to study and enhance, as needed, severe accident strategies, emphasizing the continued importance of information from the affected reactors at Daiichi.

In the BWROG presentation on Terry™ Turbine Expanded Operating Band (TTEXOB) project, which is a collaborative effort between the BWROG, Institute for Applied Energy (IAE), DOE [with participation by Idaho National Laboratory (INL), SNL, and Texas A&M University (TAMU)], Mr. Bunt provided an overview of the project milestones, schedule, and completion status:

- Milestones 1 and 2: Principles & Phenomenology: Scoping and limited modeling efforts is complete;
- Milestone 3: Full-Scale Separate Effects Component Tests: Experiments at Texas A&M University (TAMU) - started in 2019 and is underway;
- Milestone 4: Terry Turbo-pump Basic Science Experiments: Testing at TAMU - started in 2019 and is underway;
- Milestone 5: Integral Full-Scale Experiments for Long-Term Low Pressure Operations: Test facility evaluation is in progress, but is on hold at this time due to funding delays from IAE (Japan) funds (BWROG and DOE funds are available);
- Milestone 6: Integrated Full-Scale or Small-Scale Experiments Replicating 1F2 Self-Regulating Feedback: Scoping and Cost Estimate - is to be performed;
- Milestone 7: Collection of Milestone Information for Code Updates and Project Closeout: Integral with milestone work is to be completed after other milestones are completed.

During his presentation, participants discussed anticipated use of information obtained from this program, observing that test data may show that there is additional time for operators to implement severe accident strategies and that some strategies may change (e.g., there may not be a need to trip the Terry™ Turbine at low [< 150 psia] pressure). Furthermore, it is possible that test data could be used to reduce Terry™ Turbine maintenance testing requirements.

In the SNL presentation on the Terry™ Turbine project, Dr. Andrews provided an update on RCIC experiments and associated turbo-pump modeling effort. Results from oil-bearing tests indicate that this oil is adequate for long term RCIC operation at temperatures that exceed current oil technical specifications. Valve testing has been completed to characterize flow characteristics, such as the loss coefficient (i.e., "C_v") as a function of valve open position. This testing includes both the governor valve and the trip throttle valve and measurements were obtained for water and air conditions. Turbine efficiency characterization tests have been completed for the ZS-1 small Terry^{TM} turbine in air and water conditions, which is believed to be directly applicable to the larger version of this Terry^{TM} turbine. It was fortunate that this small scale replica of the full sized turbine was available, because it allows this work to be done at much lower cost. Modeling work is also progressing. Participants suggested that direct validation of the MELCOR RCIC turbo-pump model with plant periodic RCIC test data would be a useful addition to this project.

In summary, Leads did not propose any changes to Topic Area 5 recommendations (see Section 7 of Reference [2] or the U.S. list of examination requests related to Topic Area 5 in Appendix C of Reference [2]).

2.3. Other Topics of Interest

2.3.1. US DOE Activities

At the start of the meeting, Damian Peko, the DOE manager of the U.S. Forensics Effort, welcomed attendees. He emphasized that the focus of this DOE-sponsored effort is to provide as much information as possible from the Fukushima forensics activities to improve the safety of our operating fleet while not adversely affecting NDF activities to proceed expeditiously with the deconstruction and decommissioning of the Fukushima reactors and site cleanup. Mr. Peko also provided an overview of other relevant DOE activities to this topic. Most notably, he discussed the on-going Civil Nuclear Energy Research and Development Working Group (CNWG) efforts in which the U.S. Forensics Effort is a key activity.

Joy Rempe, the Technical Lead for the U.S. Forensics Effort, also welcomed participants, noting that this year there were new organizations participating in this effort, the NDF and BWX Technologies (BWXT). Dr. Rempe then reviewed the objectives, motivation, and approach for this effort. She proposed an approach and a schedule that would allow the FY2020 letter report to be completed and to identify other topic information briefs that may be provided during FY2020 (as funding allows). Finally, she provided an overview of the meeting agenda and link from which participants could access presentation material.

2.3.2. US NRC Activities

Richard Lee provided an overview of relevant NRC-sponsored computer codes that benefit from forensics information from Daiichi and other international programs in which the US NRC participates. Currently, the NRC participates in several OECD/NEA international projects, such as:

- Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF) project
- Preparatory Studies for Fuel Debris Analysis (PreADES)
- Thermodynamic Characterization of Fuel Debris and Fission Products based on Scenario Analysis for Severe Accident Progression at Fukushima-Daiichi NPS (TCOFF)
- Analysis of Information from Reactor Building and Containment Vessel and Water Sampling in Fukushima Daiichi Nuclear Power Station (ARC-F), and
- Reduction of Severe Accident Uncertainties (ROSAU).

See [2] for additional details regarding these OECD/NEA projects. Additional details about ROSAU are provided in Section 2.2.3 and about PreADES are provided in Section 2.1.3. Dr. Lee indicated that the NRC will also participate in the new CEA program, ESTER, that was motivated because a prior CEA program was not capturing the source term releases between ventings observed in 1F3 or capturing measured trends for iodine release. Dr. Lee's presentation emphasized the importance of data from Daiichi and these international experimental programs for models in MELCOR as well as MAAP.

2.3.3. Systems Analysis Code Model Improvements and Evaluations

2.3.3.1. MELCOR

Nathan Andrews provided an overview of new models being incorporated into MELCOR for non-LWRs and LWRs and a new uncertainty analysis being completed in support of OECD/NEA Fukushima project. The revised eutectic formation model for accident tolerant fuel, may be applicable for LWR severe accident applications. His presentation also described a new approach for conducting an uncertainty analysis to support severe accident forensics evaluations. It still requires engineering judgment, but its formal structure will reduce the apparent arbitrary nature of past forensic investigations. In a subsequent presentation, Dr. Andrews provided additional information regarding materials interactions that could be included in systems analysis codes that could affect predictions for relocation and combustible gas generation.

2.3.3.2. MAAP Evaluations

Chris Henry, FAI, presented material from three recent evaluations he completed using the MAAP code: 1F2 Data Interpretations and Associated Implications; RCIC Model Benchmark against 1F2 RCIC Performance; and 1F3 PCV Pressure and RPV Plenum Wall Failure Implications.

During his presentation of the 1F2 benchmark evaluations, Dr. Henry indicated that MAAP models rely heavily on TMI-2 benchmarks. In other discussions (e.g., see Section 2.2.1), participants queried whether TMI-2 data may be applicable to BWR melt progression in which there are different materials and geometries. Although he concludes his presentation observing that MAAP models can be used to predict accident progression phenomena consistent with available data, there are several phenomena, such as early-stage vessel breach, late-stage vessel failure, debris quenching, and potential plugging of vessel breach for which 'modeling fundamentals' are lacking (see Slide #31 in Appendix C.3.3.1).

In his presentations about the 1F2 RCIC performance and the 1F3 vessel failure, Dr. Henry presented results from MAAP that are again consistent with available data. However, as emphasized by meeting participants in other sessions, there are still many uncertainties about the phenomena that led to the observed RCIC performance, the actual mechanism that led to RPV failure, and the observed relocation of large core components.

2.3.3.3. Panel Discussion regarding Examination Needs for Systems Analysis Codes

A panel, that included representatives from the US NRC that sponsors the MELCOR code and the Electric Power Research Institute (EPRI) that sponsors the MAAP code, provided thoughts regarding examination information important to code modeling. This section summarizes key points raised by participants:

- Tom Kindred, EPRI - It is important to recognize that there are uncertainties in predicting severe accident phenomena, and information from examinations at Daiichi are important to reducing these uncertainties. Uncertainties highlighted during this meeting include RCIC performance, MCCI combustible gas generation, and debris coolability. Efforts to reduce uncertainties be prioritized based on how they affect risk measures, such as Core Damage Frequency (CDF) and Large Early Release Frequency (LERF). Systems analysis codes, such as MAAP, should not be used to push any agendas or should not be applied beyond their intended use.
- Hossein Esmaili, US NRC - Resources are limited at the US NRC. The funding required for modeling changes must be justified based on risk impact and noted that uncertainties in boundary conditions may have more impact than any modeling changes. Although there is much to be learned, future applications will likely involve different initial and boundary conditions.
- Randy Gauntt, Gauntt Technical Safety Associates, LLC, - MELCOR predictions for hydrogen generation have been decreasing in recent years owing to incremental model improvements. These changes have led to a degradation in other predicted quantities in the Fukushima accident sequences that are improved by higher hydrogen generation. Such changes also suggest potential limitations in current modeling approximations, namely the oxidation of molten materials that are relocating. Additional effort to account for hydrogen modeling limitations could improve other predicted accident pressure signatures.
- Chris Henry, FAI, - Systems analysis codes have come a long way. Further improvements are academic unless they affect severe accident guidance.
- Jeff Gabor, Jensen Hughes - There are some important insights from the affected reactors at Daiichi that could affect severe accident guidance. For example, the location of relocated core materials, the location where blockages form, and the morphology of relocated debris could affect water addition strategies.
- David Luxat, SNL - Examination information regarding the amount of oxidation and relocation of peripheral assemblies and morphology of relocated debris is of interest.
- Mitch Farmer, ANL, and Kevin Robb, ORNL - There have been several efforts, by LWRS program and the Severe Accident Research NETwork (SARNET), to identify gaps in our knowledge for modeling severe accident progression. Findings from these efforts are summarized in Slides #2 and 3 in Appendix C.3.4. Many of these gaps still exist that could impact severe accident guidance. In addition, several new areas were identified for consideration. Specific phenomena identified by these participants include (Slides #5 and 6 of C.3.4):
  - Simulating the flashing of reference legs in differential pressure (dP) cells to indicate water level in the RPV and to aid in operator training regarding expected instrumentation performance during a severe accident
  - Considering the amount of debris holdup on ex-vessel structures, modeling breakout, spreading, and cooling of relocated debris
  - Simulating the composition of relocated materials,
  - More detailed MCCI modeling
  - Simulating upper internals heatup and relocation, and
  - Simulating suppression chamber heatup and stratification.

In summary, several participants emphasized that updates to models in systems analysis codes could provide important insights that could be used to further enhance severe accident guidance and emergency operating procedures.

2.3.4. Updates to U.S. Information Requests

As described in Section 1.1, primary objectives of the U.S. forensics effort are to develop and update consensus U.S. input for high priority time-sequenced examination tasks and supporting research activities that can be completed with minimal disruption of D&D plans for Daiichi. Initial information requests were developed in 2014. Every year, these information requests are reviewed and as appropriate, updated. Appendix B presents the current version of these information requests. Since 2014, several new information requests were added and the status of several U.S. information requests was modified. Since these requests were first documented, an emphasis has also been placed upon identifying the motivation for the request and how the obtained information would be used. Experts participating in the U.S. forensics effort factored in experience from TMI-2 examinations, prioritizing information that would be beneficial for defueling efforts and for operations and safety. In addition, representatives from TEPCO Holdings have participated in each expert panel meeting, discussing data obtained from 1F examinations and planned future investigations.

During the FY20 meeting, experts did not identify any new information requests. However, in their review, participants from the US and Japan noted progress made on several requests, refined existing requests, provided additional documentation on how information had been used, and how information could benefit D&D as well as operation of the existing fleet and advanced reactors. Notable changes identified during the review of information requests in Appendix B include:

- Additional information that is now, or will soon be, available was denoted on Reactor Building Requests RB-3, RB-4, RB-5, RB-8, and RB-11; PCV Requests PC-1, PC-3(b),
- Additional information should be requested from NRA (Japan) on Requests RB-4, RB-5, RB-8, RB-11
- A decision to distinguish completed information requests by shading them in light gray.
- PC-8 was updated to request specific images of seals around PCV pressure sensors.
- Benefit/Use descriptions were enhanced, providing additional justification on how information could assist by reducing maintenance costs, reduce FLEX equipment requirements, enhance operator training and severe accident guidance, affect risk metrics, assist efforts to request life extensions to beyond 80 years, and reduce seismic requirements and shielding requirements in codes and standards for existing and new reactors. Cases where examination information has been used have been updated to note benefit (e.g., PC-19).
- During the discussions, several information requests (RB-1, RB-2, RB-5, PC-6, PC-9, PC-12) were identified as having the potential to affect plant maintenance activities, in particular lessons learned in the area of radiation protection were of interest. It was requested that an information bulletin be prepared on this topic if FY2020 funds are restored to FY2019 levels.

2.4. Summary

The DOE has established the U.S. Forensics Effort to work with TEPCO Holdings to learn what information is being obtained and to communicate this information to cognizant U.S. experts that could use this information to enhance safety of the U.S. commercial fleet. Presentations and discussions at the FY2020 meeting again emphasize the importance of this effort and the benefit being obtained by the nuclear enterprise. Important findings and associated recommendations from this meeting are highlighted in Section 3.

3. FINDINGS AND RECOMMENDATIONS

Information obtained from Daiichi is required to inform D&D activities. In 2014, the DOE-NE has funded U.S. experts in LWR safety and plant operations to meet each year to evaluate information obtained from Daiichi. Representatives from TEPCO Holdings and other Japan organizations (NDF and JAEA) have participated in expert panel meetings, discussing data obtained from 1F examinations and planned future investigations. Since its inception, this effort has documented its findings and recommendations in annual reports and other publications. For FY2020, it was decided that the program would gain more benefit from a more concise report that emphasizes new information and insights that affect changes to findings and recommendations from the U.S. experts participating in this effort. This section highlights findings and associated recommendations from these meetings and changes to key insights and recommendations documented in the FY2019 report.[2]

Finding 1:

The complexity of D&D activities at Daiichi is unprecedented.

NDF presentations emphasized the ‘step-by-step’ approach that allows for learning as activities are completed. Concerns regarding radiation release have increased because residents are returning to previously evacuated regions near the Daiichi site.[22] While safety and efficient decommissioning of Daiichi is the highest priority, principles established by NDF recognize that it is also important to understand the cause of the accident and that obtained data offer the potential to improve global safety of nuclear power.

Finding 2 and Associated Recommendation:

Information obtained from the affected reactors continues to be implemented in severe accident management strategies and systems analysis code evaluations.

As emphasized in Tables 2-4 and 2-5 of the 2019 Forensics Effort report [2], owners groups have used insights from forensics examinations to update guidance and support procedures for severe accident prevention and mitigation. Presentations by the BWROG, PWROG, ANL, and FAI illustrate the impact of information already gleaned from the affected reactors at Daiichi and the need to continue monitoring new insights obtained from forensics examinations.

Recommendation: U.S. organizations should continue to monitor information obtained from the affected reactors at Daiichi. Important insights continue to come from examinations at Daiichi that affect accident management strategies and could reduce uncertainties in systems analysis codes.

Finding 3 and Associated Recommendation:

Information from the affected reactors could also provide many other important insights in the areas of maintenance requirements, radiation protection methods, reactor design, and siting requirements. These insights apply to the existing fleet, new reactor design and siting, and radiation cleanup activities.

Recommendation: To illustrate the broader impact of information from Daiichi, an information bulletin should be prepared regarding radiation protection ‘best practices’ learned from Daiichi D&D activities.

Finding 4:

No new information requests were identified by U.S. experts.

Topic area leads and participants did not identify any new information requests. However, justification information (e.g., benefits/use) for several information was revised. In some cases, additional clarification was provided regarding desired information.

Finding 5 and Associated Recommendation:

Careful evaluation and re-evaluation of information obtained from on-going D&D activities at Daiichi continue to provide important insights about the accident progression in the affected units.

Presentations by TEPCO provided new information about the endstate of the components and systems and debris from recent activities, such as robotic examinations, water suspension tests, and visual examinations. Subsequent presentations by U.S. experts illustrate how such information can be used to provide insights regarding accident mitigation strategies.

Recommendation: Participants from U.S. nuclear power industry and NRC office of research, as well as the national laboratories, all agreed that DOE-NE should continue to fund this effort at FY2019 levels. Important insights continue to come from examinations at Daiichi that affect accident management strategies and maintenance requirements for the existing fleet and design, maintenance, and siting requirements for new reactors. In addition, Japan organizations rely on U.S. input to identify important information needs to support D&D efforts at Daiichi, as well as reducing modeling uncertainties for advancing nuclear safety.

Finding 6 and Associated Recommendation:

Participants agreed that systems analysis codes have demonstrated a good ability to capture the main trends of the accident progressions up through core degradation, including nuanced differences between the 1F1, 1F2, and 1F3 sequences, but uncertainties remain in simulating later aspects of the accidents.

Uncertainties emphasized during the FY2020 discussions include the mode of RPV lower head failure, holdup on ex-vessel structures, combustible gas generation, and ex-vessel debris coolability. Participants emphasized that risk important insights are still coming from the affected reactors at Daiichi that could affect severe accident guidance, such as the location of relocated core materials, the location where blockages form, and the location and morphology of relocated debris that could affect water addition strategies. However, representatives from funding organizations cautioned that resources are limited. Hence, updates need to provide value commensurate with the risk importance. Because modeling improvements associated with these insights must be prioritized based on risk significance, it is important that insights be documented in reports rather than just slides. Reports are less costly than code development.

Recommendation: To the extent possible, funding agencies should continue to document insights from the affected reactors at Daiichi in reports and update systems analysis code models to reflect risk-important insights. Unless implemented in these codes, their ability to predict accident progression in future accidents will be limited and models of physical phenomena that provided such insights will be lost.

Finding 7 and Associated Recommendation:

The ex-vessel debris observed at 1F1, 1F2, and 1F3 provides important insights regarding conditions at the time of failure. The ROSAU project also offers the potential to reduce uncertainties regarding ex-vessel debris quenching.

U.S. expert evaluations, based on evidence from prior experiments and Chernobyl Unit 4, conclude that there was water present in the cavity at the time of vessel failure in 1F1, 1F2, and 1F3. However, the height of material observed in 1F2 and 1F3 suggest quench phenomena not previously evaluated in prototypic testing. In his presentation, Dr. Farmer emphasized the importance of this information because of its potential to impact water additional strategies.

Recommendation: Examinations should provide information to reduce uncertainties about debris quenching. The desired information is documented in Information Request PC-3, PC-17, PC-18, and PC-22 of Appendix B.

Finding 8 and Associated Recommendation:

The Terry™ Turbine project offers the potential for important reactor safety insights and reduce operating costs.

This project, which was initiated to investigate the long duration RCIC performance observed at 1F2 and 1F3, is a collaborative effort between the BWROG, IAE, DOE (INL, SNL, and Texas A&M). Results offer data that may be applicable to BWR and PWR Terry™ Turbine performance and lead to changes to accident management strategies and reduced maintenance requirements. The project is making excellent progress, but funding delays may adversely impact its success.

Recommendation: Terry™ Turbine project collaborators should find avenues to overcome current funding difficulties. It is also recommended that plant periodic RCIC test data be used to validate on-going modeling efforts. Such benchmarking of the MELCOR and MAAP models for RCIC performance on plant periodic RCIC test data is crucial, if industry intends to take any credit for this known system behavior.

4. REFERENCES

1. J. Rempe, M. Farmer, M. Corradini, L. Ott, R. Gauntt, and D. Powers, “Revisiting Insights from Three Mile Island Unit 2 Post-Accident Examinations and Evaluations in View of the Fukushima Daiichi Accident,” *Nuclear Science and Engineering*, **172**, November 2012, pp 223-248.

2. J. L. Rempe (editor) P. Amway, N. Andrews, S. Basu, R. Bunt, J. Butler, J., P. Ellison, M. Farmer, J. Gabor, R. Gauntt, C. Henry, T. Honda, P. Humrickhouse, K. Klass, T. Kobayashi, S. Kraft, R. Linthicum, W. Luangdilok, Lutz; R; D. Luxat, S. Mizokami, A. Nakayoshi, D. Osborn, C. Paik, M. Plys, J. Rempe. K. Robb, R. Sanders, K. Shearer, R. Wachowiak, T. Washiya, P. Whiteman, and B. Williamson, *U.S. Efforts in Support of Examinations at Fukushima Daiichi - 2019 Evaluations*, ANL-19/08, August 2019. doi:10.2172/1561221. https://www.osti.gov/biblio/1561221.

3. J. Rempe (editor) P. Amway, N. Andrews, S. Basu, R. Bunt, M. Corradini, P. Ellison, M. Farmer, J. Gabor, R. Gauntt, T. Hara, P. Humrickhouse, S. Kraft, R. Linthicum, W. Luangdilok, R. Lutz, D. Luxat, S. Mizokami, D. Osborn, C. Paik, M. Plys, J. Rempe, K. Robb, R. Wachowiak, and B. Williamson, *US Efforts in Support of Examinations at Fukushima Daiichi – 2018 Evaluations*, ANL-18/24, August 2018.

4. J. Rempe (editor) P. Amway, N. Andrews, S. Basu, R. Bunt, M. Corradini, P. Ellison, M. Farmer, T. Farthing, J. Gabor, R. Gauntt, C. Gerardi, T. Hara, P. Humrickhouse, S. Kraft, R. Linthicum, S. Mizokami, W. Luangdilok, R. Lutz, D. Luxat, C. Negin, C. Paik, M. Plys, J. Rempe, K. Robb, R. Sanders, R. Wachowiak, B. Williamson, and D. Yamada, *US Efforts in Support of Examinations at Fukushima Daiichi – 2017 Evaluations*, ANL/LWRS-17/02, August 2017.

5. J. L. Rempe (editor), Amway, N. Andrews, W. Bixby, R. Bunt, M. Corradini, P. Ellison, M. Farmer, T. Farthing, M. Francis, J. Gabor, R. Gauntt, C. Henry, P. Humrickhouse, S. Kraft, R. Linthicum, W. Luangdilok, R. Lutz, D. Luxat, J. Maddox, C. Negin, C. Paik, M. Plys, J. Rempe, K. Robb, R. Sanders, R. Wachowiak, B. Williamson, *US Efforts in Support of Examinations at Fukushima Daiichi – 2016 Evaluations*, August 2016, ANL/LWRS-16/02.

6. J. L. Rempe (editor), P. Amway, R. Bunt, M. Corradini, P. Ellison, M. Farmer, M. Francis, J. Gabor, R. Gauntt, C. Henry, D. Kalinich, S. Kraft, R. Linthicum, W. Luangdilok, R. Lutz, D. Luxat, C. Paik, M. Plys, C. Rabiti, J. Rempe, K. Robb, R. Wachowiak, and B. Williamson, *US Efforts in Support of Examinations at Fukushima Daiichi,* ANL/LWRS-15/02*,* August 2015.

7. J. Rempe, M. Corradini, M. Farmer, J. Gabor, R. Gauntt, T. Hara, W. Luangdilok, R. Lutz, D. Luxat, S. Mizokami, K. Robb, M. Plys, K. Tateiwa, Y. Yamanaka, “Safety Insights from Forensics Evaluations at Daiichi,” *invited paper, Fukushima Daiichi Special Edition, Journal of Nuclear Materials and Energy, December 2016.* http://dx.doi.org/10.1016/j.nme.2016.08.010.

8. D. Peko, S. Basu, S. Kraft, S. Mizokami, and J. Rempe, “Working Together to Enhance Reactor Safety,” *Nuclear News Featured Cover Article*, April 2018.

9. J. L. Rempe and D. L. Knudson, “Instrumentation Performance during the TMI-2 Accident”, *IEEE Transactions on Nuclear Science*, **61**, Issue 4, January 2014, pp 1963-1970.

10. U.S. NRC, "Three Mile Island Accident of 1979 Knowledge Management Digest," NUREG/KM-0001, Rev 1, Volume 1: Overview, June 2016, and Volume 2: Recovery and Cleanup, January 2019. Documents also available at: https://tmi2kml.inl.gov/HTML/Page1.html and reports from EPRI can be downloaded from their website.

11. J. R. Wolf, J. L. Rempe, L. A. Stickler, G. E. Korth, D.R. Diercks, L.A. Neimark, D.W. Akers, B.K. Schuetz, T. L. Shearer, S. A. Chàvez, G. L. Thinnes, R. J. Witt, M. L. Corradini, and J. A. Kos, *Integration Report, OECD-NEA-TMI-2 Vessel Investigation Project*, TMI V(93) EG10, October 1993. (Also issued as NUREG/CR-6197, EGG-2734, March 1994).

12. EPRI, *Modular Accident Analysis Program (MAAP),* Version 5.02, EPRI Product No. 3002001978/3002001979LLC, 2013.

13. Sandia National Laboratories, *MELCOR Computer Code Manuals, Vol. 1: Primer and User's Guide, and Vol. 2: Reference Manual, Version 2.2.9541, SAND 2017-0455 O,* Sandia National Laboratories, January 2017 (ADAMS Accession No. ML17040A429).

14. TEPCO Holdings, Website, http://www.tepco.co.jp/en/index-e.html and http://photo.tepco.co.jp/en/index-e.html, last accessed July 2019.

15. TEPCO Holdings, *Evaluation of the situation of cores and containment vessels of Fukushima Daiichi Nuclear Power Station Units-1 to 3 and Examination into Unsolved Issues in the Accident Progression — Progress Report No. 1*, December 13, 2013.

16. TEPCO Holdings, *Report on the Investigation and Study of Unconfirmed/Unclear Matters in the Fukushima Nuclear Accident Progress Report No. 2*, August 6, 2014.

17. TEPCO Holdings, *Report on the Investigation and Study of Unconfirmed/Unclear Matters in the Fukushima Nuclear Accident - Progress Report No. 3*, May 2015. http://www.tepco.co.jp/en/press/corp-com/release/betu15_e/images/150520e0101.pdf, last accessed May 2018.

18. TEPCO Holdings, *Evaluation of the Situation of Cores and Containment Vessels of Fukushima Daiichi Nuclear Power Station Units-1 to 3 and Examination into Unsolved Issues in the Accident Progression — Progress Report No. 4*, December 17, 2016 (with attachments).

19. TEPCO Holdings, *The 4th Progress Report on the Investigation and Examination of Unconfirmed and Unresolved Issues on the Development Mechanism of the Fukushima Daiichi Nuclear Accident*, December 17, 2015, last accessed May 2019, https://www4.tepco.co.jp/en/press/corp-com/release/2015/1264602_6844.html, Includes 151217e0101.pdf [Slides]; 151217e0102.pdf [Main Body]; and 151217e0111.pdf - [Attachment I-4] "Examination into water injection by fire engines".

20. TEPCO Holdings, *The 5th Progress Report on the Investigation and Examination of Unconfirmed and Unresolved Issues on the Development Mechanism of the Fukushima Daiichi Nuclear Accident*, December 25, 2017, last accessed May 2019, https://www7.tepco.co.jp/newsroom/press/archives/2017/1485273_10469.html, Includes: 171225e0201.pdf [Slides], 171225e0201.pdf [Main Body], "Evaluation of the situation of cores and containment vessels of Fukushima Daiichi Nuclear Power Station Units 1to 3and examination into unsolved issues in the accident progression - Progress Report No. 5"; 171225e0217.pdf - [Attachment 1-10] "Analysis of the hydrogen explosion at Unit-1 reactor building"; 171225e0101.pdf; 171225e0218.pdf - [Attachment 1-11] "Estimation of accident

progression at Unit-1 based on the air dose rate monitoring data"; [Attachment 2-14] "Estimation of reactor water levels at the time when core damage and core melt progressed at Unit-2";171225e0241.pdf [Attachment 3-9] "Estimation of reactor water levels at the time when core damage and core melt progressed at Unit-3"; and 171225e0242.pdf - [Attachment 3-10] Evaluation of the fraction of Unit-3 vent gas that flowed into Unit-4 reactor building".

21. Institute for Applied Energy, "Information Portal for the Fukushima Daiichi Accident Analysis and Decommissioning Activities," https://fdada.info/en/home2/, last accessed March 2017.

22. NDF, "Technical Strategic Plan 2019 for the Decommissioning of the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company Holdings, Inc.," September 9, 2019, http://www.dd.ndf.go.jp/eindex.html, last accessed December 2, 2019.

23. Y. Yamanaka, "The Facts obtained from the Investigation Activities at Fukushima Daiichi NPS," *presentation at the first DOE Meeting on Reactor Safety Gap Evaluation and Fukushima Forensics at Argonne National Laboratory*, January 7, 2015.

24. TEPCO Holdings, http://www.tepco.co.jp/en/nu/fukushima-np/images/handouts_120927_02-e.pdf, last accessed Feb. 2016.

25. TEPCO Holdings, http://www.tepco.co.jp/en/nu/fukushima-np/images/handouts_121003_01-e.pdf, last accessed Feb. 2016.

26. TEPCO Holdings, http://www.tepco.co.jp/nu/fukushima-np/roadmap/images/m120730_05-j.pdf, (in Japanese) last accessed March 2016.

27. D. Yamada, T. Honda, and S. Mizokami, "Fukushima Update (2) Hot spots found around penetration pipes in Unit-1 Reactor buildings," Tokyo Electric Power Company, *presented at DOE Fukushima Forensic Meeting,* Washington D.C., April 28-29, 2016.

28. TEPCO Holdings, http://www.tepco.co.jp/nu/fukushima-np/roadmap/images/m111226_08-j.pdf, (in Japanese), last accessed March 2016.

29. TEPCO Holdings, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2015/images/handouts_151020_01-e.pdf, last accessed February 2016.

30. TEPCO Holdings, http://www.tepco.co.jp/nu/fukushima-np/roadmap/images/d151001_08-j.pdf, last accessed Feb. 2016.

31. TEPCO Holdings, http://www.tepco.co.jp/nu/fukushima-np/roadmap/images/d150827_08-j.pdf, last accessed Feb. 2016.

32. TEPCO Holdings, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2015/images/handouts_151022_01-e.pdf, last accessed Feb. 2016.

33. TEPCO Holdings, http://www.tepco.co.jp/en/nu/fukushima-np/images/handouts_120419_03-e.pdf, last accessed February 2016.

34. TEPCO Holdings, http://www.tepco.co.jp/nu/fukushima-np/handouts/2015/images/handouts_151127_08-j.pdf, last accessed February 2016.

35. TEPCO Holdings, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2013/images/handouts_130409_10-e.pdf, last accessed Feb. 2016.

36. TEPCO Holdings, http://www.tepco.co.jp/nu/fukushima-np/roadmap/images/d151224_08-j.pdf, (in Japanese), last accessed March 2016.

37. TEPCO Holdings, http://www.tepco.co.jp/nu/fukushima-np/roadmap/images/d151029_08-j.pdf, (in Japanese), last accessed March 2016.

38. TEPCO Holdings, http://photo.tepco.co.jp/en/date/2013/201307-e/130708-02e.html, last accessed March 2016.

39. TEPCO Holdings, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2014/images/handouts_140527_07-e.pdf, last accessed Feb. 2016.

40. TEPCO Holdings, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2013/images/handouts_130315_01-e.pdf, last accessed Feb. 2016.

41. TEPCO Holdings, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2013/images/handouts_131114_05-e.pdf, last accessed Feb. 2016.

42. TEPCO Holdings, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2013/images/handouts_130221_02-e.pdf, last accessed Feb. 2016.

43. TEPCO Holdings, http://www.tepco.co.jp/en/nu/fukushima-np/images/handouts_120627_02-e.pdf, last accessed Feb. 2016.

44. TEPCO Holdings, http://www.tepco.co.jp/en/nu/fukushima-np/images/handouts_120607_02-e.pdf, last accessed Feb. 2016.

45. TEPCO Holdings, http://www.tepco.co.jp/en/nu/fukushima-np/images/handouts_120418_04-e.pdf, last accessed Feb. 2016.

46. TEPCO Holdings, http://www.tepco.co.jp/en/nu/fukushima-np/images/handouts_120712_03-e.pdf, last accessed Feb. 2016.

47. TEPCO Holdings, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2014/images/handouts_140728_05-e.pdf, last accessed Feb. 2016.

48. D. Yamada, T. Futatsugi, S. Mizokami, (1) Dose rate distribution and gamma spectrum analysis on refueling floor at Unit 3, *presented at DOE Fukushima Forensic Meeting*, Washington DC, April 28-29, 2016.

49. TEPCO Holdings, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2013/images/handouts_130416_06-e.pdf, last accessed Feb. 2016.

50. TEPCO Holdings, http://photo.tepco.co.jp/en/date/2014/201405-e/140515-01e.html, last accessed Feb. 2016.

51. TEPCO Holdings, http://www.tepco.co.jp/nu/fukushima-np/roadmap/images1/images1/d160128_07-j.pdf, last accessed Feb. 2017.

52. TEPCO Holdings, "Air dose rates in the buildings," http://www.tepco.co.jp/en/nu/fukushima-np/f1/surveymap/index-e.html, last accessed March 2016.

53. TEPCO Holdings, http://www.tepco.co.jp/nu/fukushima-np/roadmap/2018/images1/d180426_08-j.pdf, last accessed May 2018.

54. IRID and IAE, "Estimation on Fuel Debris Location," *presentation given by K. Nozaki, TEPCO Holdings, at DOE Fukushima Forensic Meeting*, Washington D.C., November 15, 2016.

55. TEPCO Holdings, "On the Internal Survey of Unit 1 Reactor Containment Vessel - Analysis Result of Sediment," (in Japanese), presentation by Tokyo Electric Power Company Holdings, Inc., http://www.tepco.co.jp/nu/fukushima-np/roadmap/2017/images1/d170525_08-j.pdf, May 25, 2017.

56. C. Igarashi and N. Shikazono, "Sound-Producing Sand in Japan: Major Element Composition and Its Minerals Determined by X-ray Diffraction and X-ray Fluorescence," *Analytical Sciences*, **Vol. 19**, pp. 1371-1374 (2003).

57. "Concrete Reference Reactor Building Core Boring Sample Radioactivity Analysis," August 29, Heisei 20, Japan Atomic Energy Agency (in Japanese).

APPENDIX A. FY2020 Meeting Agenda and Attendee List

A.1. November 17-18, 2019 Meeting Agenda

Reactor Safety Technology Expert Panel Forensics Meeting

Meeting Agenda
November 18-19, 2019

Argonne National Laboratory (ANL) Offices
955 L'Enfant Plaza, North, SW, Suite 6000
Washington, DC 20024-2168

Monday, November 18, 2019

8:30 AM	Welcome, Administrative Matters, and Safety Minute	M.Farmer, ANL
8:35 AM	Welcome and Overview – DOE Activities, Plans, and Constraints	A. Duncan/D. Peko, DOE-NE
8:45 AM	NRC International Activities	R. Lee, NRC
9:00 AM	Objective and Planned Agenda	J. Rempe, Rempe and Associates, LLC
9:10 AM	Strategic plan 2019 for fuel debris retrieval from the Fukushima Daiichi NPS	H. Wakabayashi NDF
9:35 AM	Fundamental concept for fuel debris analysis of the Fukushima Daiichi NPS	J. Nakano NDF
10:00AM	*Break*	All
10:15 AM	TEPCO Update and Discussion - Plans for Unit 1 Investigation - Unit 2 Investigations - Insights with respect to systems analysis codes (RCIC performance, MCCI, vessel failure) - Unit 1 water suspension tests - Unit 1 missile shield investigation (if time permits)	S. Mizokami TEPCO
11:30 PM	*Working Lunch* Update on PreADES Project	All A. Nakayoshi, JAEA (presented by J. Rempe)
12:30 PM	Topic 5 – Operations & Maintenance BWROG EPC Update – Implementation of SAMG Rev 4 – Development of computer-based training (CBT) for SAGs – Updates on Information Requests (Implications for Operations, Maintenance, Severe Accident Mitigation, and Accident Analysis)	Bill Williamson, TVA Phil Ellison, GEH Kenneth Klass, Talen Energy

Reactor Safety Technology Expert Panel Forensics Meeting

Meeting Agenda
November 18-19, 2019

Argonne National Laboratory (ANL) Offices
955 L'Enfant Plaza, North, SW, Suite 6000
Washington, DC 20024-2168

Monday, November 18, 2019 (Continued)

1:15 PM	Topic 5 – Operations & Maintenance (cont'd) BWROG RCIC ExOB Update RCIC Expanded Operating Band (RCIC ExOB) Committee / Terry Turbine Expanded Operating Band (TTEXOB) Project Overview	Randy Bunt BWROG Committee Chairman (Southern Nuclear)
1:45 PM	Topic 5 – Operations & Maintenance (cont'd) PWROG Update – Implementation of SAMGs Updates on Information Requests (Implications for Operations, Maintenance, Severe Accident Mitigation, and Accident Analysis)	K. Shearer, PWROG
2:30	MAAP Code Updates and Analysis Insights - 1F 2 accident progression - 1 F2 RCIC performance - 1F3 vessel failure	C.Henry, FAI
3:30PM	*Break*	All
3:45 PM	MELCOR Code Analysis Insights and Updates	N. Andrews/D. Luxat, SNL
4:15 PM	Path Forward on Modeling Issues and Related Examination Needs	Panel
5:00 PM	*Adjourn*	

Reactor Safety Technology Expert Panel Forensics Meeting

Meeting Agenda
November 18-19, 2019

Argonne National Laboratory (ANL) Offices
955 L'Enfant Plaza, North, SW, Suite 6000
Washington, DC 20024-2168

Tuesday, November 19, 2019

8:30 AM	Topic 1 – Component and System Performance - 1F1 Shield plug - Efforts to reduce injection flow - RCIC testing insights - Mizokami presentation (RCIC performance, Vessel Failure, etc.) Insights/Comments on Information Consistency and Adequacy for Reactor Safety Insights Revisions to information requests (as needed)	K. Robb, ORNL/ J. Gabor Jensen Hughes/ N. Andrews/SNL
9:45 AM	Topic 2 – Radiation Surveys and Sampling Updates Related to New Material Available Insights/Comments on Information Consistency and Adequacy for Reactor Safety Insights Revisions to information requests (as needed)	N. Andrews, SNL / D. Luxat, SNL
10:00 AM	*Break*	All
10:15 AM	Topic 3 - Core Debris Location Evaluations - MCCI Insights - Update on ROSAU - Recent 1F2 Exam Results Insights/Comments on Information Consistency and Adequacy for Reactor Safety Insights Revisions to information requests (as needed)	M. Farmer, ANL
11:00 AM	Topic 4 – Combustible Gas Effects -1F3 Explosion and Implications based on BSAF Phase 2 Results Insights/Comments on Information Consistency and Adequacy for Reactor Safety Insights Additional information requests (if needed)	W. Luangdilok, H2Technology LLC

Reactor Safety Technology Expert Panel Forensics Meeting

Meeting Agenda
November 18-19, 2019

Argonne National Laboratory (ANL) Offices
955 L'Enfant Plaza, North, SW, Suite 6000
Washington, DC 20024-2168

11:50 AM	Next Steps ▪ Proposed letter report(s) ▪ Action items and schedule	J. Rempe, Rempe and Associates, LLC
Noon	*Adjourn*	All

<u>Tuesday, November 19, 2019 [Follow-on Meeting to Update Information Requests]</u>

Noon	*Working Lunch*	All
	Update to Consensus Information Requests	All
2:00 PM	*Adjourn*	All

A.2. November 18-19, 2019 Attendees

Name	Organization
Don Algama	U.S. Nuclear Regulatory Commission
Nathan Andrews	Sandia National Laboratories
Sud Basu	McGill Engineering Associates
Randy Bunt	Southern Nuclear Company, BWR Owners Group
Michael L. Corradini	University of Wisconsin-Madison
Aleshia Duncan	U.S. Department of Energy
Phillip G. Ellison	GE-Hitachi Nuclear Energy Americas, BWR Owners Group
Hossein Esmaili	U.S. Nuclear Regulatory Commission
Mitchell T. Farmer	Argonne National Laboratory
Terri V. Farthing	GE Hitachi
Jeff Gabor	Jensen Hughes
Randy Gauntt	Gauntt Technical Safety Associates, LLC
Chris Henry	Fauske and Associates, LLC
Tom Kindred	Electric Power Research Institute
Ken Klass	Talen Energy
Tatsuro Kobayashi	TEPCO Holdings
Jun Kondo	Embassy of Japan
Kenneth Klass	Talen Energy, BWR Owners Group
Steven Kraft	Kraft-Contente, LLC
Richard Lee	U.S. Nuclear Regulatory Commission
Wison Luangdilok	Fauske and Associates, LLC; H2 Technology, LLC
David Luxat	Sandia National Laboratories
Donald Marksberry	U.S. Nuclear Regulatory Commission
Robert Martin	BWX Technologies
Shinya Mizokami	TEPCO Holdings JAEA (CLADS)
Junichi Nakano	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Damian Peko	U.S. Department of Energy
Marty Plys	Fauske and Associates, LLC
Joy Rempe	Rempe and Associates, LLC
Kevin Robb	Oak Ridge National Laboratory
Kyle Shearer	PWR Owners Group, Westinghouse
Hiroji Wakabayashi	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Paul Whiteman	Framatome
Bill T. Williamson	TVA, BWR Owners Group

APPENDIX B. Information Requests

As described in Section 1.1, primary objectives of the U.S. forensics effort are to develop and update consensus U.S. input for high priority time-sequenced examination tasks and supporting research activities that can be completed with minimal disruption of D&D plans for Daiichi. Initial information requests were developed in 2014. Every year, these information requests are reviewed and as appropriate, updated. Appendix B.1 presents the current version of these information requests. As described in Section 1.3, these information requests are organized into tables for each location (e.g., the reactor building, the PCV, and the RPV). Since 2014, several new information requests were added and the status of several U.S. information requests was modified. Requests that have been completed are shaded in light gray. Since these requests were first documented, an emphasis has also been placed upon identifying the motivation for the request and how the obtained information would be used. Experts participating in the U.S. forensics effort factored in experience from TMI-2 examinations. Hence, this appendix only lists information requests that are judged to be beneficial for defueling efforts and for operations and safety. In addition, representatives from TEPCO Holdings have participated in each expert panel meeting, discussing data obtained from 1F examinations and planned future investigations.

Selected items in Section B.1 are shaded in light purple. This designates that more detailed requests have been developed for nearer-term information requests:

- **RB-9b:** Photos/ videos of damaged walls and structures (1F3).
- **RB-10:** Photos/ videos and dose surveys of 1F1 (vacuum breaker), 1F1, 1F2, and 1F3 PCV leakage points (bellows, penetrations).
- **RB-15:** Examinations (water level and additional dose measurement) of 1F1 Reactor Building Closed Cooling Water System (RCW) surge tank
- **PC-1:** Photos/ videos of drywell head, head seals, and sealing surfaces (1F1, 1F2, and 1F3). Procedures used to tension and torque the bolts used to close the drywell head bolts.
- **PC-3a:** Photos/ videos of relocated debris and crust, debris and crust extraction, hot cell exams, and possible subsequent testing (1F1 - 1F3)
- **PC-3b:** PCV liner examinations of debris (photos/videos and metallurgical exams; 1F1-1F3)
- **PC-3c:** Photos/ video, RN surveys, and sampling of debris and water samples near the pedestal wall and floor (1F1-1F3).
- **PC-3d:** Concrete erosion profile; photos/videos and sample removal and examination (1F1-1F3)
- **PC-3e:** Photos / videos of RPV lower head and of structures and penetrations beneath the vessel to determine damage and corium hang-up (1F1-1F3).
- **PC-5:** Photos/videos of 1F1, 1F2, and 1F3 main steam lines and Automatic Depressurization System (ADS) lines to end of SRV tailpipes, including instrument lines.
- **PC-6:** Visual inspections of 1F1, 1F2, and 1F3 SRVs and Main Steam Lines (MSLs) including standpipes (interior valve mechanisms).
- **PC-17:**[*] Chemical and isotopic analysis of the upper layer of sediment on drywell floor at the X-100B penetration location in 1F1. Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes.

---

* The detailed request for PC-17, PC-18, PC-19, PC-20, and PC-22 are combined (see Table B-14).

- **PC-18:*** Evaluate nature of material below the sediment at the 1F1 X-100B penetration location to determine if fuel debris is present. Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes
- **PC-19:***Chemical analysis (using X-Ray Florescence or XRF) of black material discovered on CRD exchange rail in 1F2 at X-6 penetration location. This item has been completed, so it is now shaded gray rather than purple.
- **PC-20:*** Chemical analysis of black material on 'existing structure' in 1F1 images at location 'D3'.
- **PC-21:** Images from examinations in 1F3 X-53 penetration
- **PC-22:*** Chemical analysis of debris from locations at different axial and radial positions (bores, if possible). Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes (1F1-1F3)
- **RPV-1b:** Photos/videos, probe inspections of 1F1, 1F2, and 1F3 MSLs; interior examinations of MSLs at external locations. If significant distortion observed, then metallurgical exams of samples would be of interest for D&D
- **RPV-4:*** Remote mapping of 1F1, 1F2, and 1F3 core through shroud wall from annular gap region (muon tomography and other methods, as needed).
- **RPV-5:*** Mapping of end state of core and structural material (visual, sampling, hot cell exams, etc.).

The current version of these more detailed requests, which are also updated each year, are found in Section B.2. These detailed requests provide additional information regarding the benefits of obtaining this information, how obtained data would be used, the methods and/or tools required to obtain this data, the expected schedule for when this data would be available, and any follow-on research that may be required to use this data.

* The detailed request for RB-4 and RB-5 are combined (see Table B-18).

B.1. Summary Information Requests

Table B-1. Information requests for the reactor building

Item	What/How Obtained	Why	Benefit /Use	When	Status
RB-1	Photos/ videos[a] of condition of RCIC valve and pump before drain down and after disassembly (1F2 and 1F3)	• Determine turbine condition. • Gain insights about status of valve and pump at time of failure [PWRs have almost identical pumps for AFW].	Impacts BWR AM strategies (cause of RCIC room flooding). Use to support RCIC testing project (for confirmation of testing results). Potential PWR impacts (e.g., modeling, AM strategies, etc.). May also be beneficial in engineering of systems and interactions with the plant; may reduce maintenance costs; may reduce FLEX requirements; may increase operator knowledge.	Currently flooded (requires underwater investigations unless drained). Inspections could be completed more easily at Daini.	Not currently considered by TEPCO Holdings. If torus not drained, requires underwater technology available. If photos or data are obtained as part of D&D activities, please provide (but the U.S. recognizes that additional information may not be obtained).
RB-2	Photos/ videos of HPCI System after disassembly (1F1, 1F2, and 1F3)	• Gain insights about degradation due to seismic events (1F1, 1F2, and 1F3) and due to operation (1F3). • Compare endstate of 1F3 (look for flaws) with the endstate of 1F1 and 1F2. If similar flaws are observed in all three units, it would be useful for assessing impact of the seismic event and of longer term operation.	Impacts AM strategies (equipment utilization). May also be beneficial in engineering of systems and interactions with the plant; may reduce maintenance costs; may reduce FLEX requirements; may increase operator knowledge.	Currently flooded (requires other alternatives for underwater investigations unless drained).	Not currently considered by TEPCO Holdings; If torus not drained, requires underwater technology. If photos are obtained as part of D&D activities, please provide (but the U.S. recognizes that additional information may not be obtained and that system degradation may be due to long term exposure to water since the accident).

Table B-1. Information requests for the reactor building

Item	What/How Obtained	Why	Benefit /Use	When	Status
RB-3a	Photos/videos of damaged walls and structures (1F1)	• Determine mode of explosion in 1F1 compared to 1F3.	Understanding what happened; assist D&D efforts. Potential BWR improvements; Impacts BWR AM strategies and code models (venting and interconnection between units); Potential PWR impacts (e.g., modeling, AM strategies, etc.)	Some have been performed, but additional information may be obtained after debris removal.	TEPCO Holdings has obtained information (Dose rate distribution measurement around SGTS filter was performed for 1F4 and 1F3. Visual inspection inside RB was performed from view of integrity of structures for 1F4). If additional images are obtained as part of D&D activities, please include reference length scales (or information about component dimensions). In particular, if D&D strategy allows additional photos of the shield plugs for all units, include a reference length of damaged components, if possible. 1F1 shield plug surveys have been completed. When shield plugs are removed, time lapsed videos during removal are requested. Photos after debris removal are also of interest.
RB-3b	Photos/videos of damaged walls and structures (1F3)	• Determine mode of explosion in 1F3. • Gain insight about highly energetic explosions in 1F3 compared to 1F1.	Understanding what happened; assist D&D efforts. Potential BWR improvements; Impacts BWR AM strategies and code models (venting and interconnection between units); Potential PWR impacts (e.g., modeling, AM strategies, etc.).	Some have been performed, but additional information may be obtained after debris removal	
RB-3c	Photos/videos of damaged walls and structures (1F4)	• Determine mode of explosion in 1F4.	Understanding what happened; assist D&D efforts. Potential BWR improvements; Impacts BWR AM strategies and code models (venting and interconnection between units); Potential PWR impacts (e.g., modeling, AM strategies, etc.).	Completed.	
RB-4	Photos/videos of damaged walls and components and radionuclide surveys (1F2)	• Cause of depressurization. • Cause of H_2 generation.	Understanding what happened; assist D&D efforts. Impacts BWR AM strategies (equipment utilization and venting); Improved BWR code simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.).	Completed.	TEPCO Holdings has dose distribution information. In addition, NRAJ completed gamma camera investigation of 1F2 refueling floor as independent investigations. This item has been addressed.

Table B-1. Information requests for the reactor building

Item	What/How Obtained	Why	Benefit /Use	When	Status
RB-5	Radionuclide surveys (1F1, 1F2, and 1F3)	• Leakage path identification. • Dose code benchmarks. • To develop lessons learned with respect to decontamination effectiveness.	Understanding what happened; assist D&D efforts. Improved BWR code simulations and dose code benchmarks. Insights regarding 'best practices' is of interest for developing improved BWR maintenance and operational practices, Accident Management (plant robustness, training, SAMG). Insights regarding 'best practices' is also of interest for developing improved PWR maintenance and operational practices and other potential PWR impacts (e.g., modeling, AM strategies, etc.). Information may also be beneficial for DOE cleanup activities.	Completed, but additional information may be obtained after debris removal.	TEPCO Holdings has survey information in 1F1, 1F2, and 1F3 RB. some concrete samples analyzed to investigate Cs permeation inside concrete floor. Dose rate distribution measurements on 1F2 and 1F3 including top of shield plug. Dose surveys obtained around 1F1, 1F2, and 1F3 pipe penetrations (outside end of penetrations through PCV) in RB. WW vent line in 1F1 extremely contaminated such as AC piping in RB 1st floor, SGTS filter train area, piping connected to stack. Dose rate around rupture disc of 1F2 WW vent line was performed. No contamination around rupture disc 1F2, but SGTS filter was highly contaminated. If additional isotopic composition of samples/swipes from drywell head are obtained, data are of interest. In particular, Ru information is of interest. A dose map of 1F1 after cleanup is also of interest. In addition, NRAJ completed surveys as independent investigations and to improve understanding of accident progression.
RB-6	Radionuclide surveys and sampling of ventilation ducts (1F4)	• Isotope concentration could be used for determining source of H_2 production for CCI.	Understanding what happened. Potential BWR plant improvements (hardened vent use, AM strategies, and multi-unit effects, etc.). Potential PWR impacts (e.g., modeling, AM strategies, multi-unit effects).	Completed.	TEPCO Holdings is not planning any additional examinations. This item is closed. If additional information become available, please provide.

Table B-1. Information requests for the reactor building

Item	What/How Obtained	Why	Benefit /Use	When	Status
RB-7	Isotopic evaluations of obtained concrete samples (1F2)	• Code assessments. • Possible model improvements for building retention assumptions.	Understanding what happened; assist D&D efforts. Improved BWR modeling and emergency planning; cross check of RN surveys. Potential PWR impacts (e.g., modeling, AM strategies, etc.).	Completed.	This has been addressed. JAEA has obtained surface RN concentrations and RN distribution from boring concrete samples. Surface radionuclide concentrations and distribution of radionuclides of boring core samples were obtained. If additional samples or surveys are obtained, isotopic composition is of interest (but the U.S. recognizes that additional information may not be obtained).
RB-8	Photos/ videos and inspection of seismic susceptible or radiation degraded components and structures (e.g., bellows, penetrations, welds, structures, supports, etc. in 1F1, 1F2, 1F3, and 1F4)	• To confirm with data that there were no seismic-induced failures • To determine with data if there are any radiation-degraded components and concrete structures; • To develop lessons learned regarding their performance under high radiation conditions	Understanding what happened; assist D&D efforts. Improved plant robustness; observed differences between 1F1 and 1F3. Potential PWR impacts (e.g., similar penetrations, structures, and components). Additional seismic data for large magnitude earthquakes that is specific to nuclear related components and systems is of interest for operating and new reactors. It may be possible to use results to discern differences between challenges from H_2 explosions and seismic events.	Now and later (as debris is removed); Note that debris currently precludes data from being obtained.	Images obtained by TEPCO Holdings have been archived per request of NRAJ for Unit 4 (see NRA website). TEPCO has published report on Units 5 and 6 and on Daini. TEPCO Holdings will review and provide additional images of interest. **1F1:** The IC main unit, major pipes, and major valves visually investigated to confirm whether there was any damage that could cause reactor to lose coolant. Since inside area of PCV inaccessible, IC, pipes, and valves outside PCV checked. **1F2:** No large abnormality was found in the robot camera's visual inspection. Visual inspection inside PCV performed in 1F1, 1F2, and 1F3 but inspection range limited. If additional information is obtained as part of planned D&D activities, please provide it (but the U.S. recognizes that additional information may not be obtained).

Table B-1. Information requests for the reactor building

Item	What/How Obtained	Why	Benefit /Use	When	Status
RB-9	a) DW Concrete Shield Radionuclide surveys (1F1, 1F2, and 1F3 - after debris removed)	• To understand leak-age amounts and locations.	Improved AM strategies (Plant improvements for BWRs and PWRs, training, and education). Improved codes. Understanding what happened; assist D&D efforts. Could reduce requirements in codes and standards for existing and new reactor designs.	Now and later (as debris is removed).	RN surveys obtained by TEPCO Holdings have been archived per request of NRAJ. TEPCO Holdings will review and provide additional information of interest. If additional information is obtained as part of planned D&D activities, please provide (but the U.S. recognizes that additional information may not be obtained).
	b) Photos/ videos and dose surveys around mechanical seals and hatches and electrical penetration seals (as a means to classify if joints in compression or tension)	• Potential leakage paths for RN and hydrogen release.[b] • To develop lessons learned regarding seal performance under high radia-tion/high tempera-ture conditions	Improved AM strategies (Plant improvements for BWRs and PWRs, which have similar seals). Improved codes. Understanding what happened with pressure sensors; Improved knowledge for D&D efforts and reduce requirements in codes and standards for existing and new reactor designs	Now and later (as debris is removed).	Images and RN surveys obtained by TEPCO Holdings have been archived per request of NRAJ. If photos are obtained as part of planned D&D activities, please provide (but the U.S. recognizes that additional information may not be obtained).
RB-10	Photos/videos and dose surveys of 1F1 (vacuum breaker), 1F1, 1F2, and 1F3 PCV leakage points (bellows, penetrations)	• Potential leakage paths for RN and hydrogen release. • To develop lessons learned regarding penetration perfor-mance under high radiation/high tem-perature conditions	Improved AM strategies (Plant improvements for more robustness, training, education); applicable to BWRs and PWRs (which have similar penetration designs). Improved codes. Improved understanding of events; assist D&D efforts.	Now and later.	Images and RN surveys obtained by TEPCO Holdings have been archived per request of NRAJ. TEPCO Holdings has provided additional information on 1F1. As additional testing is completed, the US would appreciate it.[c] Now, restoring works for PCV to stop water leakage are higher priority, and there is no plan to scrutinize the damaged area or degree of PCV. If additional photos or information is obtained, please provide (but the U.S. recognizes that additional information may not be obtained).

Table B-1. Information requests for the reactor building

Item	What/How Obtained	Why	Benefit /Use	When	Status
RB-11	Photos/ videos and dose information on 1F1, 1F2, and 1F3 containment hardpipe venting pathway, SGTS and associated reactor building ventilation system	• To assess performance of seals under high temperature and radiation conditions.[d] • To develop lessons learned regarding their performance under high radiation/high temperature conditions	Improved AM strategies (Plant improvements). Improved understanding of events, assist D&D efforts.	Completed.	**1F1:** Dose rate of venting pathway and the point in front of SGTS room. Because of high dose rate, access to SGTS room is difficult. **1F2 and 1F3:** Photos and dose rate of SGTS trains and venting pathway available. This item has been addressed. NRAJ has additional information that will become available.
RB-12	Photos/ videos at appropriate locations near identified leakage points in 1F1, 1F2, and 1F3.	• To discern reason for leakage from the reactor building into the turbine building. • To develop lessons learned regarding their performance under high radiation/high temperature conditions	Improved BWR AM strategies (Plant improvements); potential PWR impacts, depending on identified leakage path. Assist D&D efforts.	Completed.	This item has been addressed. No additional activities currently considered by TEPCO Holdings. If additional photos are obtained as part of planned D&D activities, please provide (but the U.S. recognize that additional information may not be obtained).

Table B-1. Information requests for the reactor building

Item	What/How Obtained	Why	Benefit /Use	When	Status
RB-13	Photos/ videos of 1F1, 1F2, and 1F3 main steam lines at locations outside the PCV	• To determine PCV failure mode. • To develop lessons learned regarding their performance under high radiation/high temperature conditions	BWR AM strategies (plant mods, etc.) and better simulations for training. Assist D&D efforts.	Now and later.	**1F2:** TEPCO Holdings has some visual information related to 1F2 MSIV. **1F3:** Water leak from near expansion joint (bellows) of MSL D in MSIV room was confirmed. The water level in the PCV is estimated at about 2 m above the reactor building first floor by converting the S/C pressure obtained by the existing pressure indicators to water head, and this was confirmed during first PCV entry investigation. This elevation is on the level of PCV penetrations for main steam lines, thus indicating the possibility of water leaks from the PCV penetration of MSL. TEPCO Holdings has some temperatures around MSIV recorded since September 2011 for 1F2 and 1F3. Some evidence also on 1F1 and 1F2 provided by Yamada at 4/28/16 meeting. This item has been addressed; However, if more information is obtained as part of planned D&D activities, please provide (but the U.S. recognizes that additional information may not be obtained).
RB-14	Perform chemical analysis of high radiation deposits or particles found inside the reactor building (1F1, 1F2, and 1F3); e.g., the white deposits from the HPCI room using ICP, FE-SEM, XRD, etc.	• Presence of Ca/Al/Si would indicate MCCI.	Assist D&D efforts for determining debris location.	Now and later	TEPCO Holdings has provided results from examinations of initially available samples from 1F2 RB during November 2018 meeting. The US suggests that future sample examinations provide information about the presence of Ca along with Al and Si.

Table B-1. Information requests for the reactor building

Item	What/How Obtained	Why	Benefit /Use	When	Status
RB-15	Examinations (water level and additional dose measurement) of 1F1 RCW surge tank	• During events at 1F1, contaminated water may have entered RCW and/or water may have flowed out of RCW into containment. • To develop lessons learned regarding component performance under high radiation/high temperature conditions	Determine the role of the RCW during 1F1 accident.	Now.	RN surveys obtained by TEPCO Holdings have been archived per request of NRA. TEPCO Holdings will review and provide additional information of interest. TEPCO Holdings has obtained some dose rate measurements in the area around the surge tank.

a. With the exception of general area views, photos and videos should be obtained with a reference length (ruler) at appropriate locations. In particular, it would be extremely useful for RB-1, RB-2, and RB-13; it is required for photos and videos to be most effective for RB-9 and RB-10.

b. For PWR containments, the containment actually grows radially as pressure and temperature are increased so penetrations that may have been in compression (e.g., hatches) may now be in tension.

c. **1F1:** Water leaks from a sand cushion drain pipe and an expansion joint (bellows) for vacuum breaker tube observed. The water leak from a sand cushion drain pipe was confirmed since the vinyl chloride pipe (connecting the sand cushion drain tube and drain funnel with an insertion-type joint) had been displaced. Water leaks could not be confirmed at other seven drain pipes, since the drain tubes had not been displaced. However, concrete seams (joints) below sand cushion drain piping were observed to be wet all around on the concrete wall, which indicates that leaked water is filled in the sand cushion area outside of PCV wall. The water leak from bellows of vacuum breaker tube is located in the direction of access opening of pedestal wall in the PCV floor where molten corium might spread out first.

1F2: It was confirmed SC water level changes together with torus room water level. This indicates water is leaking from the lower position of SC including suction piping. No water leakage from sand cushion drain pipes or vent pipe was observed. As of now, water leakage is not specified.

1F3: Water leak from near the expansion joint (bellows) of main steam line D in MSIV room was confirmed. The water level in the PCV is estimated at about 2 m above the reactor building first floor by converting the S/C pressure obtained by the existing pressure indicators to water head. This elevation is on the level of PCV penetrations for main steam lines, thus indicating the possibility of water leaks from the PCV penetration of MSL.

1F3: Water seeping from equipment hatch is inferred from the following observations.

- Rust was observed along with the hatch interface lower than DW water level (in November 2015). Upper part of the interface does not have the rust.

-The increasing dose rate on the floor towards the equipment hatch was observed (in November 2015), which indicates contaminated water had flown from DW side

- Equipment hatch rail was dry in December 2015. Current DW water level is lowest since 2011. The DW water level in 2011 was higher and water seeping from DW through equipment hatch seal would be higher.

- The observed high dose rate at the rail in front of shield plug for equipment hatch (in September 2011) would be attributed to water leak through equipment hatch seal.

- Water dripping due to rain fall observed (in November 2015, rainy day), which might be intruding from refueling floor.

No specific observation regarding gas phase leakage other than dose rate distribution on refueling floor and steam discharging from refueling floor.

d. Passage of high temperature gas from venting operations at 1F1 and 1F3 may have affected seals. The effluent vented from 1F1 and 1F3 would also have subjected these components to high radiation fields. Note that, at present, available evidence indicates that 1F2 may not have been successfully vented. The high radiation fields in components of the 1F2 reactor building ventilation system appears to have been caused by 1F1 vent effluent bypassing the vent stack shared by 1F1 and 1F2. Many PWRs have safety grade fan cooler units for post-loss of coolant accident containment heat removal; PWRs would be interested if there is anything to learn.

Table B-2. Information requests for the primary containment vessel

Item	What/How Obtained	Why	Benefit /Use	When	Status
PC-1	Photos/ videos[a] of drywell head, head seals, and sealing surfaces (1F1, 1F2, and 1F3). Procedures used to tension and torque the bolts used to close the drywell head bolts.	• Determine how head lifted. • Determine peak temperatures. • Look for indicators of degradation due to high radiation and high temperature hydrogen, including hydrogen-induced embrittlement.	AM Strategies; What happened with respect to the leak path; better simulations for training. Assist D&D efforts. Available information indicates that no changes in tensioning procedures are needed. Additional information regarding sealing surface and elastomer condition could provide insights of what occurred and inform consideration of potential failure modes.	Now (initial data and photos) and later (if head removed).	The US is interested in comparing procedures used by the US and TEPCO. Information obtained by TEPCO Holdings has been archived per request of NRA. TEPCO Holdings will review and provide additional information of interest. TEPCO Holdings observed that tensioning is done based on gap requirements; and no records are available. TEPCO Holdings has obtained photos indicating: **1F1:** Although top head may have moved during the accident, additional information from TEPCO indicates gap in region that could be observed is small (initial and after pictures are similar). Degradation of paint is also of interest. **1F2:** No large abnormality was found in the robot camera's visual inspection in the operating floor. Rubber boots remained standing on the shield plug. **1F3:** Deformation of part of shield plug was observed, which was found in the visual inspection after removing building rubbles. Additional photos may become available. The U.S. would appreciate any additional information (although the U.S. recognizes that this information may not be available). Visual images of deformation and RN samples (with isotopic content) are of particular interest.
PC-2	Photos/videos and radionuclide surveys/ sampling of IC (1F1)	• Evaluate for seismic damage. • Evaluate final valve position. • Gain insights about hydrogen transport.	AM Strategies (plant robustness, use of equipment in limited number of plants with ICs and new passive plants); better simulations for training. Assist D&D efforts.	Completed.	TEPCO Holdings has some photos (and no damage observed); no RN sampling planned (due to radiation levels). This item has been addressed.

Table B-2. Information requests for the primary containment vessel

Item	What/How Obtained	Why	Benefit /Use	When	Status
PC-3	a) Photos/ videos of relocated debris and crust, debris and crust extraction, hot cell exams, and possible subsequent testing (1F1 - 1F3)	• Code assessments • Possible model updates for mass, height, composition, morphology (e.g., coolability), topography of debris, spreading, splashing, and salt effects.	BWR AM Strategies (plant robustness, use of equipment, inform cavity flooding strategies) and better simulations for training. Potential PWR impacts (e.g., modeling.).[b] Assist D&D efforts.	Now and > 5 years (per TEPCO Holdings roadmap).	TEPCO Holdings has obtained some samples and some photos from inside of 1F1, 1F2, and 1F3 PCV, more are planned. When additional information is available, please provide.
	b) PCV liner examinations of debris (photos/videos and metallurgical exams; 1F1-1F3)	• Code assessments. • Possible model improvements for predicting liner failure and MCCI.	AM Strategies (improved plant robustness); better simulations for training. Assist D&D efforts. Information could inform life beyond 80 considerations.	Now and > 5 years (per TEPCO Holdings roadmap).	TEPCO Holdings has some PCV visual information. When additional information is available, please provide. TEPCO Holdings has provided results from examinations of initially available debris samples within the PCV. The US requests that future debris sample examinations provide information about the presence of Ca along with Al and Si.
	c) Photos/ video, RN surveys, and sampling of debris and water samples near the pedestal wall and floor (1F1-1F3)	• For benchmarking code predictions of vessel failure location and area, mass, morphology (e.g., coolability), and composition of ex-vessel debris, and MCCI.	BWR AM Strategies, better simulations, etc. Potential PWR impacts (e.g., modeling, AM strategies, etc.). Assist D&D efforts.	Now and later.	TEPCO Holdings has some information and may obtain additional information later. For 1F1, 1F2, and 1F3, robots with cameras and dose rate meters were inserted inside PCV and retained water in D/W was sampled for radioactivity analysis. Sediment (1F1) or relocated core components (1F2 and 1F3) have been observed. If debris samples obtained, a collaborative evaluation program may be possible.
	d) Concrete erosion profile; photos/videos and sample removal and examination (1F1-1F3)	• For benchmarking code predictions of MCCI.	BWR AM Strategies (plant mods, etc.) and better simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.). Assist D&D efforts.	Now and later.	TEPCO Holdings has no plans to obtain at this time. TEPCO Holdings may consider in the future. If end-state is observed, a collaborative program to evaluate samples may be possible.
	e) Photos / videos of RPV lower head and of structures and penetrations beneath the vessel to determine damage and corium hang-up (1F1-1F3)	• Code assessments. • Possible model improvements.	BWR AM Strategies (plant modifications, etc.) and better simulations for training (improved models for predicting containment pressure-temperature response); Potential PWR impacts (e.g., modeling, AM strategies, etc.). Assist D&D efforts.	Now and later.	TEPCO Holdings will obtain some information. The U.S. believes this information is very important for benchmarking models. Please provide additional information when available.

Table B-2. Information requests for the primary containment vessel

Item	What/How Obtained	Why	Benefit /Use	When	Status
PC-4	Photos/videos of 1F1, 1F2, and 1F3 recirculation lines and pumps	• To determine PCV failure mode and relocation path. • To develop lessons learned regarding performance under high radiation/high temperature conditions	AM Strategies (plant mods, etc.) and better simulations for training.	Completed.	TEPCO Holdings has some pressure and temperature measurements at PLR pump inlet since April 2011. No additional inspections planned. The U.S. continues to have interest in this visual information. However, the U.S. recognizes that additional information may not become available.
PC-5	Photos/videos of 1F1, 1F2, and 1F3 main steam lines and ADS lines to end of SRV tailpipes, including instrument lines	• To determine RPV failure mode.	BWR AM Strategies (plant mods, etc.) and better simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.).	Now and later.	TEPCO Holdings has not considered photographic exams. TEPCO Holdings has some temperatures around SRV and MSIV recorded since September 2011 for 1F2 and 1F3. The U.S. continues to have interest in photos to resolve questions regarding SRV failure versus main steam line rupture. In particular, some visual inspection of MSL would be very valuable. However, the U.S. recognizes that additional information may not become available.
PC-6	Visual inspections of 1F1, 1F2, and 1F3 SRVs and MSLs including standpipes (interior valve mechanisms)	• To determine if there was any failure of SRVs and associated piping.	BWR AM Strategies (maintenance practices, etc.), SRV functioning in test facility data, and better simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.).	Later.	TEPCO Holdings has not considered photographic exams. TEPCO Holdings has some temperatures around SRV and MSIV recorded since September 2011 for 1F2 and 1F3. The U.S. continues to have interest in photos to resolve questions regarding SRV failure versus MSL rupture. In particular, some visual inspection of MSL would be very valuable. However, the U.S. recognizes that additional information may not become available.

Table B-2. Information requests for the primary containment vessel

Item	What/How Obtained	Why	Benefit /Use	When	Status
PC-7	Ex-vessel inspections and operability assessments of 1F1, 1F2, and 1F3 in-vessel sensors and sensor support structures[c]	• Data qualification for code assessment. • Identification of vessel depressurization paths. • To develop lessons learned regarding performance under high radiation/high temperature conditions	Equipment qualification life (1F1 at 40 years; underwater cabling); better simulations for training.	Completed	TEPCO Holdings completed some examinations and re-calibrations; no additional examinations are planned. If additional information becomes available, it will be shared. Cable integrity examinations by TDR were performed for 1F1, 1F2, and 1F3; and cable damage was confirmed. In 1F2, it was confirmed TIP index tube was stuck. In 1F2, it was found SLC injection tube in RPV was stuck, which indicates blockage by molten core. -New thermocouple was inserted into nearby N-10 nozzle to reinforce RPV temperature monitoring in Oct. 2012. -Beforehand SLC line integrity was confirmed by injecting water and monitoring discharge pressure change. -Pressurized water of about 7MPa could not penetrate SLC line into RPV.
PC-8	Examinations and operability assessments of 1F1, 1F2, and 1F3 ex-vessel sensors and sensor support structures[d]	• Data qualification for code assessment. • Identification of vessel depressurization paths. • Understanding why the RPV A and B pressure signals decalibrated. • To develop lessons learned regarding their performance under high radiation/high temperature conditions	BWR and possible PWR equipment qualification life; better qualifications for training. Insights regarding survivability support revised severe accident strategies. Images of penetration seals associated with PCV pressure sensors are of interest because of potential reduction in PM and surveillance.	Completed, but images of penetrations associated with PCV pressure sensors are of interest.	No additional operability assessment planned, but additional visual information may become available. TEPCO Holdings has completed some evaluations and recalibrations. TEPCO Holdings provided additional information regarding sensor qualification envelop and conditions exposed to during the accident.

Table B-2. Information requests for the primary containment vessel

Item	What/How Obtained	Why	Benefit /Use	When	Status
PC-9	Photos/videos of 1F1, 1F2, and 1F3 PCV (SC and DW) coatings	• Assess impact for coating survivability. • To develop lessons learned regarding their performance under high radiation/high temperature conditions	BWR and possible PWR maintenance upgrades.	Now and later.	Visual examinations inside PCV performed in 1F1, 1F2, and 1F3, although inspection range limited. TEPCO Holdings plans to evaluate the integrity of concrete pedestals and PCV liner and will share information when it is available. The US requests this additional information when available and suggests that TEPCO Holdings evaluate the presence of coating materials in elemental evaluations of other samples.
PC-10	1F1, 1F2, and 1F3 RN surveys in PCV	• Dose code assessments. • Possible model improvements.	BWR and possible PWR AM strategies/better simulations (plate out). Assist D&D efforts	Now and later.	TEPCO Holdings has some sample evaluation and survey information and may obtain more data later. Radioactivity data were obtained from retained water in basement of each building. Sampling water in D/W was performed for 1F1, 1F2, and 1F3. Sampling drain water and dust of exhaust gas from drywell was performed for 1F1, 1F2, and 1F3. S/C water not evaluated. The U.S. remains very interested in isotopic information from RN surveys/samples for code assessments (but the U.S. recognizes that this information may not become available).
PC-11	Photos/videos of 1F1, 1F2, and 1F3 primary system recirculation pump seal and any potential discharge to containment	• To assess performance under high temperature/ high pressure conditions.[e] • To develop lessons learned regarding performance under high radiation/high temperature conditions	Improved BWR AM strategies (plant improvements). Improved understanding of events. Assist D&D efforts. Potential PWR impacts.[e]	Now and later. Some exams may be completed more easily at Daini.	Not currently considered by TEPCO Holdings; Information obtained by TEPCO Holdings has been archived per request of NRA. TEPCO Holdings will review and provide additional information of interest. The U.S. remains interested in additional photographs from Daiichi or Daini (but the U.S. recognizes that this information may not become available).

Table B-2. Information requests for the primary containment vessel

Item	What/How Obtained	Why	Benefit /Use	When	Status
PC-12	Photos/videos of 1F1, 1F2, and 1F3 TIP tubes and SRM/IRM tubes outside the RPV	• To determine if failure of TIP tubes and SRM/IRM tubes outside the RPV led to depressurization. • To develop lessons learned regarding performance under high radiation / high temperature conditions	BWR AM Strategies and maintenance practices, SRV performance insights, and better simulations for training. Potential PWR impacts (e.g., modeling, AM strategies, etc.). Assist D&D efforts.	Later.	An attempt was made to insert a fiber optic scope through the 1F2 TIP guide tube. The scope was stuck at the TIP indexer and could not get past that location. 1F2 SLC injection line blockage was confirmed (see PC-7). Also, see item PC-14 for SLC injection line stuck in RPV. The U.S. continues to have interest in this information. However, the U.S. recognizes that additional information may not become available.
PC-13	Photos/videos of 1F1, 1F2, and 1F3 insulation around piping and the RPV	• To determine potential for adverse effects on long-term cooling due to insulation debris. • To develop lessons learned regarding performance under high radiation / high temperature conditions	Improved BWR and PWR AM strategies (plant improvements).	Now and later.	Not currently considered by TEPCO Holdings; some photos may already be available. The U.S. continues to have interest in this visual information. However, the U.S. recognizes that additional information may not become available.
PC-14	Samples of conduit cabling, and paint from 1F1, 1F2, and 1F3 for RN surveys	• Dose code assessments. • Possible model improvements.	BWR and possible PWR AM strategies/Better simulations (plate out).	Now and later.	TEPCO Holdings has some sample information. The U.S. continues to have interest in this information but recognizes that additional information may not become available.
PC-15	Samples of water from 1F1, 1F2, and 1F3 for RN surveys	• Dose code assessments. • Possible model improvements.	BWR and possible PWR AM strategies/Better simulations. Assist D&D efforts.	Completed.	TEPCO Holdings has some sampling information. Sampling water in D/W was performed for 1F1, 1F2, and 1F3. Sampling drain water and dust of exhaust gas from drywell was performed for 1F1, 1F2, and 1F3. This item is closed.
PC-16	Photos/videos of melted, galvanized, or oxidized 1F1, 1F2, and 1F3 structures	• To provide indications of peak temperatures (for possible model improvements).	Improved AM strategies (Plant improvements).	Now and later. Exams may be completed more easily at Daini.	Some photos may be available. The U.S. continues to have interest in this visual information but recognizes that additional information may not become available.

Table B-2. Information requests for the primary containment vessel

Item	What/How Obtained	Why	Benefit /Use	When	Status
PC-17	Chemical and isotopic analysis of the upper layer of sediment on drywell floor at the X-100B penetration location in 1F1.The upper surface of the sediment is ~ 30 cm above drywell floor. Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes.	• Presence of concrete oxides would indicate MCCI • Possible model improvements • Testing has shown that the ability to cut core debris is strongly impacted by amount of concrete oxides present • Presence of short-lived fission product isotopes could indicate low-level recriticality. • Given the low level of decay heat present in 1F1, any low-level criticality could impact plant heat balance calculations.	Assist D&D efforts for recriticality prevention, debris stabilization, locating fuel-containing materials, and debris removal and storage. Improved accident management strategies.	Now and later	TEPCO Holdings is also interested in this information. The next robot examination[f] will include the use of neutron and gamma detectors and obtain additional samples. TEPCO Holdings has provided results from examinations of initially available 1F1 PCV samples. The US requests that future sample examinations provide information about the presence of Ca along with Al and Si.
PC-18	Evaluate nature of material below the sediment at the 1F1 X-100B penetration location to determine if fuel debris is present. Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes.	• Presence of concrete oxides or core material debris would indicate MCCI • Possible model improvements • Testing shows that the ability to cut core debris is strongly impacted by amount of concrete oxides present • Presence of short-lived fission product isotopes could indicate low-level recriticality. • Given the low level of decay heat present in 1F1, any low-level criticality could impact plant heat balance calculations.	Assist D&D efforts for recriticality prevention, debris stabilization, locating fuel-containing materials, and debris removal and storage. Improved accident management strategies.	Now and later	TEPCO Holdings is also interested in this information. The next robot examination[f] will include the use of neutron and gamma detectors and obtain additional samples. TEPCO Holdings has provided results from examinations of initially available 1F1 PCV samples. The US requests that future sample examinations provide information about the presence of Ca along with Al and Si.

Table B-2. Information requests for the primary containment vessel

Item	What/How Obtained	Why	Benefit /Use	When	Status
PC-19	Chemical analysis (XRF) of black material discovered on CRD exchange rail in 1F2 at X-6 penetration location	• Identification of material could provide an indicator of peak structure temperatures and potential for structure failure. • Possible model improvements.	Assist D&D efforts for determining debris location. Modeling improvements for ex-vessel holdup have been implemented in MAAP and informed accident management strategies and risk assessment metrics.	Completed.	Examination results were presented by TEPCO during our November 2018 meeting (Sample 2). This item has been completed.
PC-20	Chemical analysis of black material on 'existing structure' in 1F1 images at location 'D3'	• Presence of Si or core material debris would indicate MCCI • Possible model improvements. • Testing shows that the ability to cut core debris is strongly impacted by amount of concrete oxides present • Presence of short-lived fission product isotopes could indicate low-level recriticality. • Given the low level of decay heat present in 1F1, any low-level criticality could impact plant heat balance calculations.	Assist D&D efforts for recriticality prevention, debris stabilization, locating fuel-containing materials, and debris removal and storage. Improved accident management strategies.	Now.	TEPCO Holdings is also interested in this. Future robot examination may obtain such samples. The next robot examination[f] will include the use of neutron and gamma detectors and obtain additional samples. TEPCO Holdings has provided results from examinations of initially available 1F1 PCV samples. The US requests that future sample examinations provide information about the presence of Ca along with Al and Si. In addition, the US suggests that future examinations provide information about the presence of short-lived fission product isotopes.
PC-21	Images from examinations in 1F3 X-53 penetration	• Possible model improvements	Assist D&D efforts for determining debris location and improved accident management strategies.	Now.	TEPCO Holdings is also interested in this information. Some images have been obtained. The U.S. would appreciate any additional images that become available.

Table B-2. Information requests for the primary containment vessel

Item	What/How Obtained	Why	Benefit /Use	When	Status
PC-22	Chemical analysis of debris from locations at different axial and radial positions (bores, if possible). Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes. (1F1-1F3)	• Presence of concrete oxides would indicate MCCI • Gain insights about material relocations • Material properties important to tooling design (e.g., density and hardness) are known to be a function of material composition (e.g., the ability to cut debris is impacted by amount of concrete oxides present). • Potential concentrations of fuel. • Presence of short-lived fission product isotopes could indicate low-level recriticality. • Given the low level of decay heat present in 1F1, any low-level criticality could impact plant heat balance calculations • Possible model improvements.	Assist D&D efforts for recriticality prevention, debris stabilization, locating fuel-containing materials, and debris removal and storage. Potential modeling improvements for debris coolability during MCCI and inform accident management strategies and risk assessment metrics.	Now and later	TEPCO Holdings is also interested in this information, and the potential for bore samples is under evaluation. The next robot examinations will obtain additional samples (neutron and gamma detectors and visual information can be used for prioritization). The U.S. requests that bores be obtained from diverse locations (e.g., with high and low count rates, high and low debris heights, different colors, etc.). The US requests that future sample examinations provide information about the presence of Ca along with Al and Si.

a. With the exception of general area views, photos and videos should be obtained with reference length scales at appropriate locations. In particular, it would be extremely useful for PC-3(b), PC-3(e), PC-9, PC-12, PC-13.

b. Key to applicability for PWRs will be if melt composition does not significantly impact spreading; with different core materials, molten core debris may behave differently. If forensics can confirm basic properties or models, information could be applicable to all LWRs.

c. Ex-vessel inspections and evaluations [e.g., continuity checks, calibration evaluations, etc.] of in-vessel sensors [dP cells, water level gauges, TIPs, TCs, etc.] and sensor support structures, cables, removed TIPs, etc.; requires knowledge of sensor operating envelop.

d. Inspections and evaluations (e.g., continuity checks, calibration evaluations, etc.) of suppression pool, PCV, and ex-vessel sensors (e.g., containment air monitors, pressure sensors, TCs, etc.) and sensor support structures and cables; requires sensors operating envelop knowledge.

e. Some PWRs have inside containment recirculation systems for Emergency Core Cooling and Containment Spray. BWR recirculation pump seals and PWR reactor coolant pump seals have many material similarities; there may also be some information relevant to reactor coolant pump seals and their ability to function following recovery or provide core cooling with core debris in-vessel.

f. Scheduled for calendar year 2019.

Table B-3. Information requests for the reactor pressure vessel

Item	What/How Obtained	Why	Benefit /Use	When	Status
RPV-1	a) 1F1, 1F2, and 1F3 dryer integrity and location evaluations (photos/videos[a] with displacement measurements, peak temperature evaluations). If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.	• Code assess-ments. • Possible model improvements.	Improved AM strategies; Improved simulations for training. Assist D&D efforts.	Later.	TEPCO Holdings will conduct visual examinations. The U.S. remains interested in all the requested information but recognizes that it may not be available. If possible, laser-Induced Breakdown Spectroscopy methods might reduce costs for chemical evaluations in exams (ongoing R&D at JAEA may make it easier to obtain this information).
	b) Photos/videos, probe inspections of 1F1, 1F2, and 1F3 MSLs; interior examinations of MSLs at external locations. If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.	• Code assess-ments. • Possible model improvements.	Improved AM strategies; Improved simulations for training. Assist D&D efforts.	Later	TEPCO Holdings has no plans for any such exams. See PC-3 for water leakage information from MSL penetration through PCV. The U.S. remains interested in this information but recognizes that it may not be available.
	c) Photos/videos of upper internals and upper channel guides. If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.	• Code assess-ments. • Possible model improvements (for predicting peak tempera-tures, displace-ment, melting).	Improved AM strategies; Possible plant modifications; Improved simulations for training. Assist D&D efforts.	Later.	TEPCO Holdings will conduct visual exams. The U.S. remains interested in all the requested information but recognizes that it may not be available.

Table B-3. Information requests for the reactor pressure vessel

Item	What/How Obtained	Why	Benefit /Use	When	Status
RPV-2	Photos/videos of 1F1, 1F2, and 1F3 core spray slip fit nozzle connection, sparger & nozzles. If significant distortion observed, then metallurgical exams of samples would be of interest for D&D. Photos/videos of 1F1, 1F2, and 1F3 feedwater sparger nozzle and injection points. If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.	• Assess operability. • Assess salt water effects (including corrosion). • Applicable to BWRs and PWRs.	Improved AM strategies; Improved simulations for training; Possible use in BWR VIP, depending on plant condition. Assist D&D efforts.	Now and later.	TEPCO Holdings has some information) and will obtain more data. When water injected through CS line in 1F1, 1F2 and 1F3, it was confirmed that RPV bottom temperature responds. When water injected through FDW line in 1F1, 1F2, and 1F3, it was confirmed that RPV bottom temperature responds. The U.S. remains interested in this information but recognizes that it may not be available.
RPV-3	1F1, 1F2, and 1F3 steam separators' integrity and location (photos/videos with displacement measurements, peak temperature evaluations). If significant distortion observed, then metallurgical exams of samples would be of interest during removal for D&D.	• Code assessments. • Possible model improvements.	Improved AM strategies, Improved simulations for training. Assist D&D efforts.	Later	TEPCO Holdings will conduct visual, exams. The U.S. remains interested in all the requested information but recognizes that it may not be available.

Table B-3. Information requests for the reactor pressure vessel

Item	What/How Obtained	Why	Benefit /Use	When	Status
RPV-4	1F1, 1F2, and 1F3 shroud inspection (between shroud and RPV wall); Photos/videos of interest. If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.	• Code assess-ments. • Possible model improvements.	Improved AM strategies; Improved simulations for training. Possible use in BWR VIP. depending on plant condition. Assist D&D efforts.	Now and later.	TEPCO Holdings has some information and will conduct visual exams. 1F2 PLR pump responded after increasing water flowrate from FDW, indicating a certain amount of water is retained outside shroud. The U.S. remains interested in this information but recognizes that some information may not be obtained.
	1F1, 1F2, and 1F3 shroud head integrity and location (photos/videos). If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.	• Code assess-ments. • Possible model improvements.	Improved AM strategies; Improved simulations for training. Possible use in BWR VIP, depending on plant condition. Assist D&D efforts.	Later.	TEPCO Holdings will conduct visual exams. The U.S. remains interested in this information but recognizes that some information may not be obtained.
	Photos/videos of 1F1, 1F2, and 1F3 shroud inspection (from core region). If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.	• Code assess-ments. • Possible model improvements.	Improved AM strategies; Possible plant modifications; Improved simulations for training. Possible use in BWR VIP, depending on plant condition. Assist D&D efforts.	Later	TEPCO Holdings will conduct visual exams. The U.S. remains interested in this information but recognizes that some information may not be obtained.
	Photos/videos of 1F1, 1F2, and 1F3 core plate and associated structures.	• Code assess-ments. • Possible model improvements.	Improved AM strategies; Possible plant modifications; Improved simulations for training. Possible use in BWR Program VIP for weld integrity, depending on plant condition. Assist D&D efforts.	Later.	TEPCO Holdings will conduct visual exams and retrieve debris on the core plate. The U.S. remains interested in this information but recognizes that some information may not be obtained.

Table B-3. Information requests for the reactor pressure vessel

Item	What/How Obtained	Why	Benefit /Use	When	Status
RPV-5	Remote mapping of 1F1, 1F2, and 1F3 core through shroud wall from annular gap region (muon tomography and other methods, as needed).	• Code assessments. • Possible model improvements.	Improved AM strategies; Possible plant modifications; Improved simulations for training. Assist D&D efforts.	Now and later.	TEPCO Holdings has deployed and provided results from muon tomography and robot examinations. More remote examinations using robots (including laser mapping) are planned.
	Mapping of end state of core and structural material (visual, sampling, hot cell exams, etc.).	• Code assessments. • Possible model improvements for predicting debris composition, mass, and morphology (e.g., coolability, topography of debris, spreading, splashing, and salt effects.	Improved BWR and potential PWR AM strategies; plant modifications, and improved simulations for training. Assist D&D efforts.	Later.	TEPCO Holdings has not yet considered but will probably perform, as necessary for defueling and D&D. If samples can be obtained from RPV, a collaborative program to evaluate may be possible.

a. With the exception of general area views, photos and videos should be obtained with reference length scales at appropriate locations. In particular, it is required for photos and videos to be most effective for RPV-1(b), RPV- 2(a), RPV-3 and RPV-4(d)

B.2. Additional details for Information Requests

Table B-4. Additional details for Information Requests RB-9b and RB-10

• **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
RB-9b: Photos/videos and dose surveys around mechanical seals and hatches and electrical penetration seals **RB-10:** Photos/videos of 1F1 (vacuum breaker), 1F1, 1F2, and 1F3 PCV leakage points (bellows and other penetrations) High-resolution images (photos/videos) of PCV penetrations and other vulnerable areas (i.e., access hatches, piping/electrical penetrations, expansion joints/bellows). Images of similar locations from each unit (1F1, 1F2, 1F3, 1F4) allows for comparison of damage and end state between units. 1F4 photos will provide a good baseline of a vessel not over pressurized. Imaging should be sufficient to estimate whether damage has occurred. External PCV images may be sufficient. Images taken internal to the PCV and of disassembled penetrations (i.e., hatch sealing faces and seal material) are desired if obtained during D&D. History on penetration leakage or repairs correlated to images is also desired.
• **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
Safety - Desired for improving reactor safety analysis models and accident management. Operational - Provides for weak link assessment of penetration capacity under high radiation/high temperature conditions. Economic - Provide insight into seal performance capability; could be used to adjust maintenance and inspection D&D - Impacts D&D because of constraints on contaminated water release, airborne radionuclide release path. Can influence D&D method by identifying where containment is leaking and to what level containment can be flooded.
• **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
Locations of PCV failure and leakage can affect the accident progression with respect to timing, accident mitigation actions, venting, and radionuclide and combustible gas releases. This information can be used to validate and/or enhance the current understanding of the conditions required for PCV failure and the locations of such failures. It can also impact operations and maintenance considerations, such as gasket and seal material selection and replacement. Linkage of repaired or degraded penetrations performance in over design conditions can provide insights to improve realistic estimates of failures and investigate improvements in repair methods.
• **Methods/Tools Needed to Collect Information or Data:**
• High resolution imaging system - external to PCV • Dose survey meter or gamma camera (3D image). • Irradiation resistant high-resolution imaging system - internal to PCV • Personnel observations indicating leakage (water dripping, discoloration, puddles)
• **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and later (continued inspections of containment and identification of leakage points for units 1F1, 1F2 and 1F3). Base line information from 1F4 can be gathered now. History of penetration maintenance and repair can support investigation of radiological releases and flood-up plans
• **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
Identification of actual penetration characteristics (e.g. geometry, seal material) may be needed to apply observations to other units. Prediction of conditions of penetration during accident (i.e., stress, temperature, pressure). Although multiple scenarios may lead to the observed endstate, comparisons between predicted and observed endstates may allow identification of possible scenarios and elimination of other scenarios. U.S. industry should develop a list of high interest penetrations/areas because of maintenance benefits and provide to TEPCO Holdings. Tabletop exercises with operation and reactor safety experts should be conducted to develop potential penetration failure scenario list.

Table B-5. Additional details for Information Request RB-15

• **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
RB-15: Reactor Building Closed Cooling Water system (RCW) inspection 1F1 Water level measurement of RCW. Dose survey around RCW surge tank. Images of the RCW system inside of containment are desired if obtained during D&D.
• **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
Safety - Desired for understanding 1F1 accident progression and the potential role of the RCW during an accident. Operational - Provides insights about component performance under high radiation/high temperature conditions. D&D - Could influences D&D efforts by identifying leakage locations.
• **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
TEPCO Holdings and the U.S. expert panel have identified the potential the failure of the 1F1 RCW sump heat exchanger piping in containment. The RCW system may have influenced the accident progression by allowing releases from containment and/or supplying cooling water to the ex-vessel debris in containment. Understanding the status of the RCW system will aid in determining the role the RCW system had during the accident.
• **Methods/Tools Needed to Collect Information or Data:**
• Dose survey meter or gamma camera (3D image). • Water level may possibly be obtained from gauge on surge tank or a dip stick. If water level is lower than surge tank, alternate assessment methods and locations may be required.
• **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term, the RCW surge tank and reactor building floors appear accessible. The surge tank inspection could accompany any future investigation of the nearby IC. Long-term, images of the RCW inside of containment (sump heat exchanger piping) may be obtained during D&D or its planning.
• **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
Identifying the design water volume of the RCW system.

Table B-6. Additional details for Information Request PC-1

• **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
PC-1: Photos/ videos of drywell head, head seals, and sealing surfaces (1F1, 1F2, and 1F3). Procedures used to tension and torque the bolts used to close the drywell head bolts. This information is of interest both prior to event and during debris removal. • Visual - signs of asymmetric lift or leakage paths. Look for thermal deformation due to high temperatures/high radiation conditions over time. • RN Swabbing • Visual inspection of seal • Visual inspection of the head. Look for evidence of permanent strain in the head flange or bulging of the head hemisphere and for evidence of bending/bowing of the bolts along their length that could result from head flange strain and result in permanent leakage location even after PCV decompression. • Inspect shield plug - visual inspection of cracks.
• **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
Operational - Provides insights about degradation under high radiation/high temperature conditions. AM Strategies; What happened with respect to the leak path; better simulations for training. Improved understanding of PCV response to overpressure that could inform accident management, especially PCV venting strategies.
• **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
Determine how head lifted with emphasis on the state of the flange closure gap and any evidence of permanent strain/deformation such that permanent leak paths would persist beyond the simple elastic bolt stretching behavior. Determine peak temperatures. Look for indicators of degradation due to high temperature hydrogen, including hydrogen induced embrittlement.
• **Methods/Tools Needed to Collect Information or Data:**
• Mostly photographic
• **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
When reactor head is opened for decommissioning purposes.
• **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
None

Table B-7. Additional details for Information Request PC-3a

• **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
PC-3a: Photos/videos of debris and crust, debris and crust extraction, possible hot cell exams, and possible subsequent testing (1F1, 1F2, and/or 1F3) High-resolution images (photos/videos) of debris and crust both in the as-found state and during extraction, and chemical analysis to determine composition and oxidation state. Imaging should be sufficient to provide insights into material characteristics (i.e., particle bed versus crust material, and if crust material, the morphology and extent of cracking if possible). A sufficient number of samples should be selected to estimate the spatial variations in composition. Elemental analysis of samples should look for fuel, structural, and concrete components. Evaluations should determine the approximate proportions of Uranium/Zirconium/Stainless Steel/Boron from the drive tubes below the vessel head and the corium samples retrieved from the cavity region. In addition, samples from the cavity region should be analyzed for the presence of Al/Ca/Si/Mg that would provide evidence of MCCI.
• **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
Structural characteristics of the material are important for supporting tooling design for removal; chemical analysis important for criticality evaluations. These same data are important for improving reactor safety analysis models and accident management.
• **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
Benchmark and reduce uncertainty in models for predicting molten core concrete interaction (MCCI) phenomena. MCCI phenomena are important for assessing combustible gas generation during late phase accident progression, as well as the extent of attack on containment structures. It is important to reduce uncertainty in this phenomenon because it affects strategies for venting and water addition. Improved knowledge will be used to enhance accident management strategies.
• **Methods/Tools Needed to Collect Information or Data:**
• Irradiation resistant high-resolution imaging system • Hot cell elemental analysis system, and/or in-situ elemental analysis using Laser Induced Breakdown Spectroscopy (LIBS) and/or X-ray Florescence • Ultimately, D&D cutting and removal tools able to extract materials
• **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later (Sample removal possible within next 2 years).
• **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
Obtaining /using this information may require additional material property and coolability testing (Young's modulus, linear expansion, ultimate strength, hardness, tensile strength, etc.) for cutting tool development and for model development. Evaluation of this information may require composition information for concrete (to distinguish between sand and concrete).

Table B-8. Additional details for Information Request PC-3b

• **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
PC-3b: PCV liner examinations (photos/videos and metallurgical exams); (1F1-1F3) High-resolution images (photos/videos) of PCV liner, with particular emphasis in regions contacted by core debris. In areas that were contacted, the imaging should be sufficient to provide insights into the nature/extent of heat transfer and/or thermo-chemical attack on the liner (e.g., distortion/displacement and extent of ablation if that occurred). A sufficient number of samples should be selected in eroded areas to determine if the boundary temperature during erosion was determined by simple melting or by eutectic formation. Evaluations should determine the approximate proportions of Uranium/Zirconium/Stainless Steel/Boron from corium samples retrieved from the cavity region. In addition, samples from the cavity region should be analyzed for the presence of Al/Ca/Si/Mg that would provide evidence of MCCI.
• **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
For D&D, plugging leaks in the liner will reduce the extent of water leakage from the PCV and determining leakage locations via liner examinations is crucial to this process. These same data are important for improving reactor safety analysis models and accident management.
• **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
Benchmark and reduce uncertainty in models for predicting liner thermal heatup and attack by core debris for ex-vessel accident scenarios. Improved knowledge will be used to enhance accident management strategies.
• **Methods/Tools Needed to Collect Information or Data:**
• Irradiation resistant high-resolution imaging system. • Laser imaging systems to reconstruct liner distortion and/or ablation profiles.
• **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later.
• **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
None.

Table B-9. Additional details for Information Request PC-3c

• **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
PC-3c: Photos/video, RN surveys, and sampling of pedestal wall and floor (1F1-1F3). High-resolution images (photos/videos), RN surveys, and sampling of 1F1, 1F2, and 1F3 pedestal wall and floor. Imaging should be sufficient to provide insights into structural integrity and/or damage incurred during the accident. A sufficient number of samples should be selected to estimate the RN distribution on the pedestal wall and floor. Evaluations should determine the approximate proportions of U/Zr/SS/Boron from corium samples retrieved from the cavity region.
• **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
Determining the pedestal wall and floor structural integrity as well as RN distributions is important for safety evaluations of D&D activities.
• **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
Benchmark and reduce uncertainty in models for predicting structure heatup and degradation during a severe accident. It is important to reduce uncertainties in this area since heat sink inside the PCV can impact predictions of water availability to cool core debris. Improved knowledge will be used to enhance accident management strategies.
• **Methods/Tools Needed to Collect Information or Data:**
• Irradiation resistant high-resolution imaging system • Robotic methods for extraction of samples for determining RN distributions • Consider developing a robot-deployed ultrasonic detection system for evaluating erosion of pedestal wall due to MCCI within the pedestal. • Muon detection systems located below grade may also be able to detect the presence of core debris in the lower regions of the containment.
• **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later (Sample removal possible within next 2 years).
• **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
None.

Table B-10. Additional details for Information Request PC-3d

• **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
PC-3d: Concrete erosion profile; photos/videos and sample removal and examination (1F1-1F3) High-resolution images (photos/videos) of concrete erosion with possible sample removal and elemental analysis. Imaging should be sufficient to estimate the total volume of relocated core material and the damaged volume of concrete. In addition, imaging should be of sufficient resolution to characterize the morphology (e.g., cracks, gaps, etc.) of the debris and concrete. A sufficient number of samples shall be selected to estimate the spatial variations in composition and oxidation state of relocated materials. Elemental analysis of samples should look for fuel, structural, and concrete components. Evaluations should determine the approximate proportions of Uranium/Zirconium/Stainless Steel/Boron from the corium samples retrieved from the cavity region.
• **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
Required for D&D facilitate planning for debris removal, and also for evaluation of the mechanical integrity of critical structures such as the reactor pedestal. Desired for improving reactor safety analysis models and accident management.
• **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
Benchmark and reduce uncertainty in models for predicting molten core concrete interaction (MCCI) phenomena. MCCI is important in assessing combustible gas generation during late phase accident progression. It is important to reduce uncertainty in MCCI phenomena because it affects strategies for venting and water addition. Improved knowledge will be used to enhance accident management strategies.
• **Methods/Tools Needed to Collect Information or Data:**
• Irradiation resistant high-resolution imaging system • Hot cell elemental analysis system • D&D cutting and removal tools able to extract materials • Consider developing a robot-deployed ultrasonic detection system for evaluating erosion of pedestal wall due to MCCI within the pedestal. • Muon detection systems located below grade may also be able to detect the presence of core debris in the lower regions of the containment.
• **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later (Sample removal possible within next 2 years).
• **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
Obtaining /using this information may require additional material property and coolability testing (Young's modulus, linear expansion, ultimate strength, hardness, tensile strength, etc.) for cutting tool development and for model development. Evaluation of this information may require composition information for concrete (to distinguish between sand and concrete).

Table B-11. Additional details for Information Request PC-3e

• **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
PC-3e: Photos/videos of RPV lower head and of structures and penetrations beneath the vessel to determine damage and corium hang-up (1F1-1F3) High-resolution images (photos/videos) of structures and penetrations with retained corium. Imaging should be sufficient to estimate the total volume of relocated core material and the damage to structures and penetrations.
• **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
Required for D&D facilitate planning for debris removal and for evaluation of the mechanical integrity of critical structures such as the reactor pedestal. Desired for improving reactor safety analysis models and accident management.
• **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
Benchmark and reduce uncertainty in models for predicting the mode(s) and associated size(s) of RPV failure and the mass and heat content of material that relocates from the RPV, which in turn, affects PCV gas temperature, PCV pressure, and the potential for MCCI.
• **Methods/Tools Needed to Collect Information or Data:**
• Irradiation resistant high-resolution imaging system • Hot cell elemental analysis system • D&D cutting and removal tools able to extract materials
• **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later (Robotic examinations underway).
• **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
Initial findings from 1F2 and 1F3 suggest that a non-negligible amount of core debris may be held up on structures below the reactor vessel. System analysis codes should be exercised assuming a range of core debris holdup in a situation that is not cooled by water to investigate the impact of heat sources not covered by water on PCV gas phase temperature and pressure.

Table B-12. Additional details for Information Request PC-5

• **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
PC-5: Photos/videos and temperatures of 1F1, 1F2, and 1F3 main steam lines and ADS lines to end of SRV tailpipes, including instrument lines.
• **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
BWR AM Strategies (plant mods, etc.) and better simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.).
• **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
To determine RPV failure mode. Initial examinations should focus on identifying failure mode(s) and location(s). For example, if images indicate that vessel lower head failure occurred, images should be of sufficient resolution to determine if the failure was a gross unzipping or a limited area. If images suggest that vessel depressurization was due to penetration failure, images should be of sufficient resolution to determine the number, type(s) [e.g., control rod drive, instrument tube, and/or drain line], and failure mode(s) [e.g., tube ejection and/or tube rupture]. Evaluations of MSLs and ADS lines should also focus on identifying failure mode(s) and location(s). Initial images may not be able to detect failure locations. Hence, dose surveys, gamma camera (3D) images, and temperature measurements may be needed to detect where radiation has leaked from the RPV.
• **Methods/Tools Needed to Collect Information or Data:**
• Irradiation resistant high-resolution imaging system (1 mm to 1 cm gaps or cracks). • Dose survey meter or gamma camera (3D image). • Thermal imaging to observe hot spots (> 100 °C increases)
• **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later.
• **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
None.

Table B-13. Additional details for Information Request PC-6

• **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
PC-6: Visual inspections of 1F1, 1F2, and 1F3 SRVs including standpipes in the torus and drywell (interior valve mechanisms)
• **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
BWR AM Strategies (maintenance practices, etc.), SRV functioning in test facility data, and better simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.).
• **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
To determine if there was any failure of SRVs and associated piping.
• **Methods/Tools Needed to Collect Information or Data:**
• Irradiation resistant high-resolution imaging system
• **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later.
• **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
None.

Table B-14. Additional details for Information Requests PC-17, PC-18, PC-19, PC-20, and PC-22

• **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
PC-17: Chemical analysis of upper layer of sediment on drywell floor at the X-100B penetration location in 1F1. The upper surface of the sediment is ~ 30 cm above drywell floor. Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes. **PC-18:** Evaluate nature of material below the sediment at the 1F1 X-100B penetration location to determine if fuel debris is present.[a] Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes. **PC-19:** Chemical analysis (XRF) of black material discovered on CRD exchange rail in 1F2 at X-6 penetration location **PC-20:** Chemical analysis of black material on 'existing vertical wall structure' in 1F1 picture outside pedestal doorway **PC-22**: Chemical analysis of debris from locations at different axial and radial positions (bores, if possible). Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes. Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes. (1F1-1F3) These five information requests are for determining the chemical composition of materials observed at locations in 1F1 (i.e., sediment and underlying material on the drywell floor below the X-100b penetration, and on existing vertical structure near the pedestal doorway), and black material discovered on the CRD exchange rail in 1F2 from the X-6 penetration. Elemental analysis of samples should look for fuel, structural, and concrete components and should also include a measurement of oxygen content if possible. In addition, evaluations should consider the potential for recriticality.
• **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
Required for D&D; desired for improving reactor safety analysis models and accident management.
• **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
Benchmark and reduce uncertainty in models for predicting vessel failure, in-vessel cladding oxidation and hydrogen production, holdup on ex-vessel structures, and MCCI phenomena. Vessel failure, holdup on ex-vessel structures, and MCCI phenomena are important for assessing combustible gas generation during late phase accident progression. It is important to reduce uncertainty in these phenomena because they affect strategies for venting and water addition. Additional analysis of PC-19 can be used to assess the extent of in-vessel cladding oxidation. Data from PC-18 evaluations can be used to determine if core debris is present and the X-100B location, thereby providing insights on the extent of core debris location which is also a critical uncertainty impacting accident management strategy. Knowledge gained from these analyses will be used to enhance these strategies. Data from PC-17 can be used to determine if the composition of this sediment varies with height. Recent chemical analysis results indicate a high presence of Na but little Cl, indicating the potential for NaCl decomposition and potential formation of CsCl which could impact source term evaluations.
• **Methods/Tools Needed to Collect Information or Data:**
• Hot cell elemental analysis system and/or in-situ elemental analysis using Laser Induced Breakdown Spectroscopy (LIBS) and/or X-ray Florescence (XRF). • Robotics systems for collecting samples, and for probing the sediment at X-100B location to determine the (loose material) sediment depth. • Sample examinations should also consider identifying short-lived fission product isotopes. If present, this would indicate low-level recriticality and thus impact approaches for debris removal and storage. • In addition, future robot entries could be instrumented with a neutron detector (to augment gamma detector) that also provide of low-level criticality, if it is occurring. • Any low-level criticality could impact plant heat balance calculations that are currently underway given the low overall level of decay heat that is currently present in the plants.
• **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later.
• **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
Evaluation of this information requires composition information for concrete (to distinguish between sand and concrete) and would benefit from chemical analysis of seashore sand located at the site.

a. See "Technical Supplement for PC-18 Evaluation".

Technical Supplement for PC-18 Evaluation

Examinations at the X-100b location in 1F1 (located ~ 130 degrees counter-clockwise from the pedestal doorway opening) indicate a layer of material covering the drywell floor that is ~ 30 cm deep. This material was identified during the initial entries through the X-100b penetration in 2012 and was reconfirmed during later entries in 2016 that provided additional data on the actual depth of the material. It is known that additional sediment had not accumulated at this location over the intervening four years because unique surface characteristics (i.e., grayish blue material thought to be lead) were still present. The upper surface of the material was determined to be loose sediment. It is not known whether this sediment extends down the entire 30 cm depth, or whether the sediment is a partial layer covering other material such as core debris.

There are a variety of potential sources for this sediment material that may include decomposed/flaked paint, thermal insulation, cable insulation, sand/sediment from low quality seawater injection, aerosol from core concrete interaction, among others. If the material is sand entrained with the seawater that was injected or concrete aerosol from core-concrete interaction, then it may be possible to determine the origin based on the relative proportions of dominant concrete oxides such as SiO_2, CaO, Al_2O_3, and MgO in the sediment. For sand from seawater injection, analysis of a sample of beach sand obtained at the site would provide definitive data for direct comparison with elemental analysis data obtained from a sample of the sediment. In lieu of this information, the composition of sand from 12 different beaches along the east and west coasts of Japan have been reported in the literature.[56] The compositions of key compounds varied considerably; i.e., 61.4-99.2 wt% SiO_2, 0.04-5.8 wt% CaO, 1.3-19.0 wt% Al_2O_3, and 0-2.0 wt% MgO. In terms of mass ratios of key elements, the resultant ratio for Si-Al is determined to range from 2.7 to 67 and for Si to Ca is determined to range from 6.9 to 1600.

Fortunately, the composition of concrete from the Daiichi site has also been measured for two samples to provide data for comparison to these ranges; see Table B-15.[57] Iron shown in Table C-4 is not considered in the current discussion as it could arise from corrosion (rust) of steel within the PCV, of which there is a massive amount. The corresponding mass ratios for Fukushima Daiichi concrete for the key elements in the two concrete samples are Si/Al: 3.6-4.2, and Si/Ca: 2.7-3.5. The Si/Al ratio for the concrete versus sand samples from around the island of Japan cannot be discriminated. However, the range of Si/Ca ratios does not overlap. In particular, the range boundaries are separated by a factor of ~ 2. Thus, if the Si/Ca ratio is lower and in the range of 2.7-3.5, it is likely concrete aerosol from MCCI. Conversely, if it is higher, ~7 or above, it is likely sand from seawater injection. Aerosol from core-concrete interaction also nominally contains a small amount of fuel (U) which would also be a discriminating factor.

Table B-15. Composition data from analysis of two concrete samples at 1F site.[57]

Sample Number	Mass%			
	Al	Ca	Fe	Si
1	7.0 ±1	7.8 ±1	3.6 ±1	25 ±1
3	6.5 ±1	9.1 ±1	3.3 ±1	27 ±1

Table B-16. Additional details for Information Request PC-21

• **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
PC-21: Images from examinations in 1F3 X-53 penetration High-resolution images (photos/videos) of external surfaces of RPV (especially of vessel failure locations); of material collected on structures beneath vessel (e.g., cables, control rod drives, support structures, gratings; and of concrete erosion on floor of PCV. Imaging should be sufficient to estimate the total volume of relocated core material at each location and the damaged volume of the vessel, any ex-vessel structures, and the concrete. In addition, imaging should be of sufficient resolution to characterize the morphology (e.g., cracks, gaps, etc.) of the debris and concrete. Measurements of dose rates and collection of samples for elemental analysis is desired. Ultimately, a sufficient number of samples shall be selected to be able to estimate the spatial variations in composition. Elemental analysis of samples should look for fuel, structural, and concrete components.
• **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
Required for D&D; desired for improving reactor safety analysis models and accident management.
• **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
Benchmark and reduce uncertainty in models for predicting vessel failure, holdup on ex-vessel structures, and molten core concrete interaction (MCCI) phenomena. Vessel failure, holdup on ex-vessel structures, and MCCI phenomena are important for assessing combustible gas generation during late phase accident progression. It is important to reduce uncertainty in these phenomena because they affect strategies for venting and water addition. Improved knowledge will be used to enhance accident management strategies. Inspections of the lower head walls at the three units are of significant value for understanding a) the active modes of vessel breach, b) the possibility for occurrence of a more benign gradual, progressive vessel breach, and 3) the role of accident management strategies (i.e., water injection to the RPV) on ameliorating challenges to containment as a result of vessel breach. Existing assessments of BWR containment response assume a number of prompt challenges to containment integrity upon vessel breach that do not appear to have occurred during the Fukushima Daiichi accident. Understanding why these challenges did not occur during the Fukushima Daiichi accidents is of immense value for not only refining our understanding of severe accident consequences, but also for providing a balanced perspective on severe accident risk to inform public policy debates around low carbon energy technologies. Inspections of debris on the containment floor are also of critical value to assess the conditions under which MCCI occurs at reactor-scale, specifically the role of ex-vessel debris discharge transients from a failed RPV lower head. Presently our state-of-the-art knowledge would tend to indicate much more severe ex-vessel damage progression would have occurred at 1F1 given the extended period over which no water addition to containment occurred. In addition to this observation, inspections of 1F2 indicate that limited damage to structures near the floor of the reactor pedestal occurred despite spreading of debris released from the RPV over this area. Substantial accumulation of debris within the 1F3 reactor pedestal has also been observed. The implications for assessing reactor-scale challenges to containment during late phase severe accident progression, in particular MCCI and ex-vessel debris coolability, is crucial to provide enhanced insights of relevance to refinement of risk characterization during this phase of an accident.
• **Methods/Tools Needed to Collect Information or Data:**
• Irradiation resistant high-resolution imaging system • Hot cell elemental analysis system • Systems to obtain dose rate measurements and collecting fluid or small particles during FY2017 examination (if it is possible). • Ultimately, D&D cutting and removal tools able to extract materials
• **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later (Sample removal possible within next 2 years).
• **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
Obtaining /using this information may require additional material property and coolability testing (Young's modulus, linear expansion, ultimate strength, hardness, tensile strength, etc.) for cutting tool development and for model development. Evaluation of this information may require composition information for concrete (to distinguish between sand and concrete).

Table B-17. Additional details for Request RPV-1b

• **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
RPV-1b: Photos/videos, probe inspections of 1F1, 1F2, and 1F3 MSLs; interior examinations of MSLs at external locations. If significant distortion observed, then metallurgical exams of samples would be of interest for D&D. Interior examinations of MSLs at external locations, looking for evidence of thermal/pressure strain and/or rupture, including nature of any ruptures such as fish mouth or more global rupture. Would like to know the approximate size of any rupture failure locations.
• **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
Improved AM strategies; Improved simulations for training.
• **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
• Code assessments and validation of current structural yielding modeling used in codes • Possible model improvements.
• **Methods/Tools Needed to Collect Information or Data:**
• Visual inspection
• **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later.
• **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
None.

Table B-18. Additional details for Information Requests RPV-4 and RPV-5

- **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**

RPV-4:

a) 1F1, 1F2, and 1F3 shroud inspection (between shroud and RPV wall); Photos/videos of interest. If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.

b) 1F1, 1F2, and 1F3 shroud head integrity and location (photos/videos). If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.

c) Photos/videos of 1F1, 1F2, and 1F3 shroud inspection (from core region). If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.

d) Photos/videos of 1F1, 1F2, and 1F3 core plate and associated structures

RPV-5

a) Remote mapping of 1F1, 1F2, and 1F3 core through shroud wall from annular gap region (muon tomography and other methods, if needed)

b) Mapping of end state of core and structural material (visual, sampling, hot cell exams, etc.)

A sufficient number of samples of core material should be examined to determine the approximate proportions of Uranium/Zirconium/Stainless Steel/Boron from any upper core remnants, core plate accumulations, drive tube accumulations above bottom of vessel, and any accumulations on the lower vessel head region. Results can be used to determine roughly the tendency for spatial separation of lower melting and metallic rich core debris materials from the more ceramic remnants and by implication, the temporal separation of relocation events. The same information is needed for the drive tubes below the vessel head and the corium samples retrieved from the cavity region. This information is needed to validate code assumptions of phase interactions during core degradation.

- **Benefits - Safety, Operational, Economic, D&D, or other benefits:**

BWR reactor safety analysis models have very significant uncertainties related to in-core damage progression modeling. These inspections can provide information that can help resolve the generally agreed upon largest uncertainties in BWR severe accident modeling. These uncertainties influence the understanding of containment response during a severe accident and are thus relevant to informing accident management.

- **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**

Resolve large uncertainties for in-core damage progression at BWR reactor-scale. These inspections are relevant to addressing areas where testing has been unable to reproduce key areas of BWR in-core debris relocation behavior at reactor-scale. The pathways by which debris relocate within the core-region influence the potential for rapid pressurization of containment to occur (e.g., due to rapid steam or hydrogen generation). The acquisition of knowledge to reduce uncertainties in this area can refine severe accident models, enhancing the effectiveness of accident management training.

- **Methods/Tools Needed to Collect Information or Data:**

  - Irradiation resistant high-resolution imaging system
  - Hot cell elemental analysis system, and/or in-situ elemental analysis using Laser Induced Breakdown Spectroscopy (LIBS) and/or X-ray Florescence
  - Ultimately, D&D cutting and removal tools able to extract materials

- **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**

Near-term and/or later (Sample removal possible within next 2 years).

- **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**

Obtaining /using this information may require additional material property and coolability testing (e.g., oxidation state of in-core debris). Refined understanding of mechanical properties of retrieved in-core debris, however, are of significant benefit to the design and development of cutting tools. Refined understanding of in-core damage progression will require effort to refine analytical models for this phase of a severe accident.

APPENDIX C. Selected Presentations at FY2020

This appendix contains presentations from participants wishing to have them published in this report. Presentations are organized similar to the structure found in Section 2 (i.e., presentations provide by Japanese organizations are found in Appendix C.1, presentations for major topic areas identified by US organizations are found in Appendix C.2, and other topics of interest (Appendix C.3). Section 2 highlights key points discussed during these and other presentations during the meeting. The agenda for this meeting is found in Appendix A.1.

C.1. Presentations from Japan

C.1.1. Nuclear Damage Compensation and Decommissioning Facilitation Corporation

C.1.1.1. Technical Strategic Plan 2019 for Decommissioning

Technical Strategic Plan 2019 for Decommissioning of the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company Holdings, Inc.

November 18, 2019

Hiroji Wakabayashi

Nuclear Damage Compensation & Decommissioning Facilitation Corporation (NDF)

©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

NDF's role

Chronology

- **March 11, 2011:** Great East Japan Earthquake & 1F NPS severe accidents
- **September 2011:** Nuclear Damage Compensation Facilitation division established
- **August 2014** : Decommissioning Facilitation division appended
- **October 2017** : Decommissioning reserve fund management function appended

We support the Japanese government and TEPCO to facilitate

① nuclear damage compensations (Financial support)
② 1F NPS decommissioning (Technological support)

NDF 2

©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

Table of Today's Contents

©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

1.Introduction

- Strategic Plan 2019 provides strategic recommendation on how to retrieve fuel debris from the first implementing unit.
- Mid-and-long-term directions that look at the overall efforts of the Fukushima Daiichi NPS, including waste management, are presented.

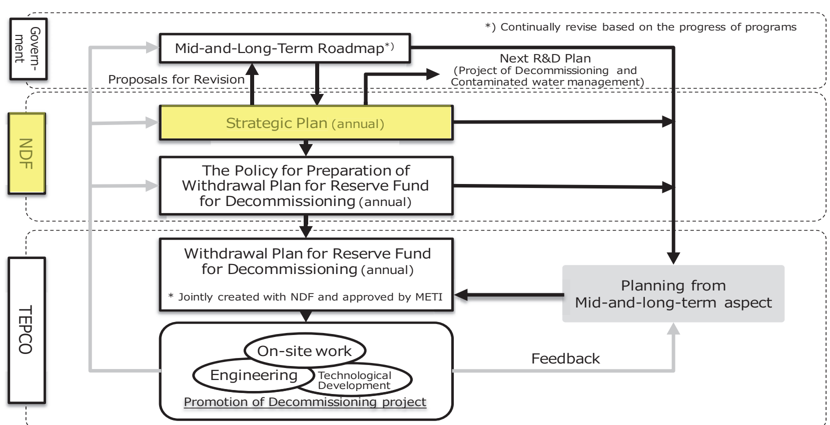


Fig. Positioning of the Strategic Plan based on the system of the Reserve Fund

©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

2. Risk Sources at Fukushima Daiichi NPS and Future Risk Reduction Strategy (1/3)

Basic policy on decommissioning

To continuously and quickly reduce the radioactive risks

Progress in decommissioning

Fuel debris retrieval

- Unit 2 (Feb 2019) Deposit contact investigation in the PCV revealed that the deposits are movable at the bottom of the PCV pedestal and on the platform.

Waste management

- (Jun 2019) Waste storage and management plan of solid waste was revised.

Contaminated water management

【REMOVING】 (continuing) Contaminated water is being purified by multi-nuclide removal equipment, etc.
【ISOLATING】 (Sep 2018) All areas of the land-side impermeable wall were frozen for isolating groundwater.
【PREVENTING LEAKAGE】(Mar 2019) Transfer of the water purified by the purification equipment to welded tanks was completed.*
【Treatment of stagnant water in buildings】(in 2018) Disconnection of the communication section between Units 1 and 2 was achieved.

* As for the treatment of the water stored in the welded tanks, comprehensive discussion including social standpoints has been made in the government-led subcommittee

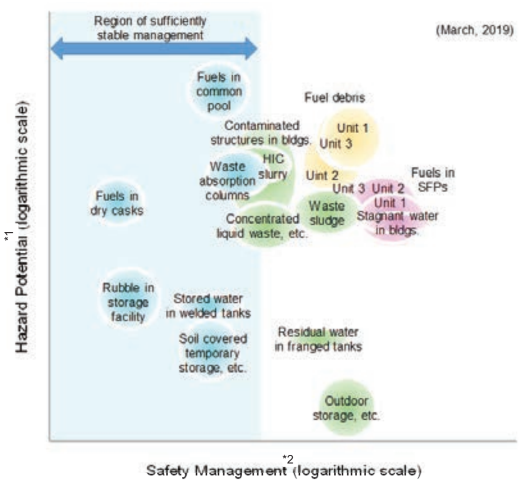
Fuel removal from spent fuel pools

- Unit 1 (continuing) Removal of rubble on the operating floor is ongoing.
- Unit 2 (Nov 2018 to Feb 2019) Investigations were conducted on the contamination state of the operating floor.
- Unit 3 (Apr 2019) fuel removal from SFP was started.

©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

2. Risk Sources at Fukushima Daiichi NPS and Future Risk Reduction Strategy (2/3)

- The interim goal of the risk reduction strategy is to bring the risk levels into the "Region of sufficiently stable management" (areas in pale blue).



Example of Risk Levels for Fukushima Daiichi NPS

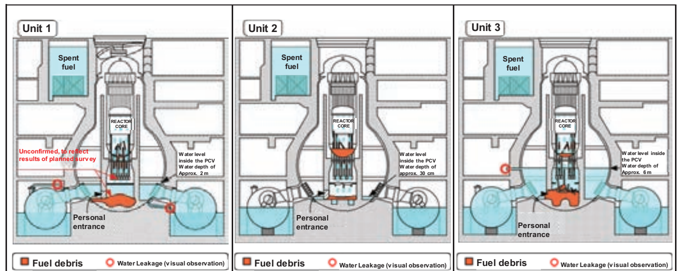
*1 : An index of impact of the event, that depends on inventory (radioactive material), form of the risk source and time allowable until the manifestation of risk in case of loss of safety function.
*2 : An index of likelihood that an event will occur, that depends on integrity of facility and on the packaging and monitoring status of risk source.

©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

2. Risk Sources at Fukushima Daiichi NPS and Future Risk Reduction Strategy (3/3)
(Status of the Estimated Distribution of Fuel Debris)

Comprehensive analyses and evaluations about ①&② have been done based on the following four information:
①Distribution of fuel debris
②Access routes and the status of surrounding structures
(The figures on the right show the distribution of fuel debris.)

- Actual measurements taken during the accident (Plant parameters, etc.)
- Severe accident progression analysis
- PCV internal investigation, muon-based fuel debris detection technology
- Scientific knowledge (Tests, etc.)



	Unit 1	Unit 2	Unit 3
Core Region	· Little fuel debris remains	· Little fuel debris remains (Stub-shaped fuels might exist in peripheral region)	· Little fuel debris remains
RPV Lower Head	· A small amount of fuel debris is present · A small amount of fuel debris is present in the inside and on the outer surface of the CRD housing	· Large amount of fuel debris is present · A small amount of fuel debris is present in the inside and on the outer surface of the CRD housing	· Fuel debris remains on the RPV lower head party · A small amount of fuel debris is present in the inside and on the outer surface of the CRD housing
Bottom of PCV Pedestal inside	· Most of the fuel debris is present	· A certain amount of fuel debris is present on the floor of the pedestal inside	· Amount of fuel debris in Unit 3 is more than that in Unit 2.
Bottom of PCV Pedestal Outside	· Fuel debris may have spread on the pedestal outside through the personal entrance	· The possibility of fuel debris spreading on the pedestal outside through the personal entrance is low.	· Fuel debris may have spread on the pedestal outside through the personal entrance
Radiation dose in operation site*	· Radiation dose rate around penetration X-6 on the first floor of R/B is high (630mSv/h).	· Radiation dose rate on the first floor of R/B had reduced by about 5mSv/h.	· Radiation dose rate on the first floor of R/B is higher than tens of mSv/h.

* Data provided by TEPCO

Prepared based on the Achievement Report 2017, Subsidy for "The Government-led R&D program on Decommissioning and Contaminated Water Management (Advancement of the comprehensive internal PCV condition analysis)" provided by IRID, The Institute of Applied Energy, June 2018

©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

3. Technological strategies toward decommissioning of the Fukushima Daiichi NPS(1/6)
([1] Fuel debris retrieval (1/5))

Goals

(1) **Retrieve fuel debris safely after thorough and careful preparations**, including safety measures, and bring them to **the state of stable storage under full management**.

(2) Toward determination of the fuel debris retrieval methods for the first implementing unit in FY 2019, and start of fuel debris retrieval work for the first implementing unit in 2021, **necessary approaches will be taken according to the Policy on Fuel Debris Retrieval**.

Policy on fuel debris retrieval

① **Step-by-step approach**
- Flexibly coordinate the direction while proceeding with the retrieval work

② **Optimization of the overall decommissioning work**
- Examine fuel debris retrieval work as a comprehensive project, including coordination with other works

③ **Combination of multiple methods**
- Combine the optimum retrieval methods for each unit, depending on the locations where fuel debris is considered to be present

④ **Approach focused on partial submersion method**
- Make efforts to focus on a more feasible partial submersion method

⑤ **Prioritizing fuel debris retrieval by access to the bottom of the PCV from the side**
- It should be considered that the accessibility to fuel debris and the removal of spent fuel can be accomplished in parallel

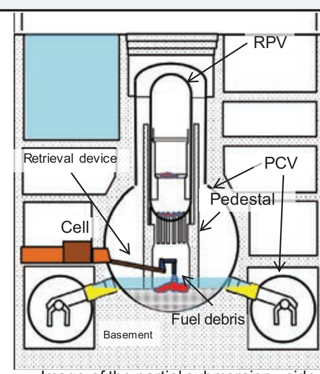


Image of the partial submersion - side access method

©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

3. Technological strategies toward decommissioning of the Fukushima Daiichi NPS(2/6) ([1] Fuel debris retrieval (2/5))

Strategies

Approach to risk reduction in fuel debris retrieval work

- **Careful operation after adequate preparation**
  - ・Retrieval remotely by partial submersion method is unprecedented work in the world,
  - ・The information on the condition inside the PCVs is limited.
- **Start with small-scale retrieval using existing safety systems and confinement wall without significant modification**
  - ・A series of continuous operation from retrieval through storage will be possible,
  - ・Early confirmation of the internal condition inside PCV and effectiveness of applied system for further operations will be expected.
- **Effects expected in the first implementation Unit**
  - ・Applied devices, installations and safety system including remote operations can be verified,
  - ・TEPCO can use this occasion as a process of mastering their operation,
  - ・Information helpful to understand the situation inside PCV can be obtained.
- **Further development of retrieval in each Unit after necessary actions such as**
  - ・actual applicability on site of safety system, data collection for safety analysis, further improvement of site environment including rad. dose reduction and further R&D activities.

NDF

9

©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

3. Technological strategies toward decommissioning of the Fukushima Daiichi NPS(3/6) ([1] Fuel debris retrieval (3/5))

Strategies

Outline of the strategic recommendation for determining fuel debris retrieval methods for the first implementing unit

Progress of R&D and PVC internal investigations
- Deposits in Unit 2 are movable.
- Development of the arm-type access equipment has been taking form.

Progress of work environment improvement
- Radiation dose in Unit 2 is relatively low.

Progress of site-wide planning
- Spent fuel removal from SFP can be performed in parallel with fuel debris retrieval operation.

Study on the retrieval concept
- Utilize existing safety systems and arm-type access equipment that actual site applicability is getting in sight .
- Start from small-scale retrieval by methods such as gripping and sucking, and also prepare plans for crushing of fuel debris.
- Contain retrieved debris in small containers and transfer them, contain them into canisters, and then store them at the dry temporary storage facility within site .

Study results of the actual site applicability for each unit
- Unit 1: Although the air-tightness of existing safety systems is relatively high, the radiation dose in the work environment is high, and the access route has been unidentified.
- Unit 2: The air-tightness of existing safety systems is high, and the radiation dose in the work environment is relatively low. The access route has been mostly confirmed.
- Unit 3: The access route has been mostly confirmed, but the air-tightness of existing safety systems is lower than other units, and the radiation dose in the work environment is relatively high.

: Preliminary engineering

Comprehensive assessment
- In Unit 2, fuel debris can be retrieved "safely", "reliably" and "promptly" using an arm-type access equipment, and the risk resulting from all fuel debris in Units 1 to 3 can be "promptly" reduced.

Points to be recommended for determining fuel debris retrieval method for the first implementing unit

[1] Fuel debris retrieval should be started on a small-scale basis by methods such as gripping and sucking.
[2] Next step of fuel debris operation are to be studied using information and experience accumulated through previous operation .
[3] Retrieved fuel debris should be transferred to the on-site temporary dry storage facilities at the Fukushima Daiichi NPS.
[4] Unit 2 is most appropriate as the first implementing unit from the view point of optimization of the overall decommissioning work, and so on .
[5] The method for expanded scale of fuel debris retrieval should be determined by proceeding with engineering including the assessment of safety based on progress of R&D, PCV internal investigation, Improvement of site condition as well as information & experience accumulated through previous operation.

NDF

10

©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

3. Technological strategies toward decommissioning of the Fukushima Daiichi NPS(4/6) ([1] Fuel debris retrieval (4/5))

Challenges

Comprehensive understanding of PCV conditions by continued internal investigations, etc.

- Internal investigations and sampling must be carefully prepared over a certain period of time.

Unit 1:	Investigations on the distribution of deposits (including a small-amount sampling) are scheduled to begin in the second half of FY 2019.
Unit 2:	Deposit contact investigations were conducted in the second half of FY 2018 (conducted on 2/13). Investigations on the distribution of deposits (including a small-amount sampling) are scheduled to begin in the second half of FY 2019.
Unit 3:	The necessity of further investigations by means of measures for lowering the water levels in PCVs and using an submersible-type access robot are being examined.

Fig. Deposit contact investigations in Unit 2*

* Source: TEPCO

Approach to the commencement of fuel debris retrieval from the first implementing unit (Unit 2)

- The method of retrieving fuel debris from Unit 2 should be determined in line with the recommendations.
- In determining the method of fuel debris retrieval in Unit 2, the following issues need to be considered.
  - ✓ Removal of obstacles in consideration of enhancing the prevention of the scattering of radioactive materials
  - ✓ Reduction of the on-site radiation dose on the 1st floor of the reactor building
  - ✓ Adjustment of work interference in cases where the operations of fuel removal from SFP and fuel debris retrieval are carried out in parallel

©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

3. Technological strategies toward decommissioning of the Fukushima Daiichi NPS(5/6) ([1] Fuel debris retrieval (5/5))

Key issues and future schedule

Fiscal Year	2019	2020	2021	2022	2023
Milestones	Determination of the fuel debris retrieval methods for the first implementing unit ▼		Start of fuel debris retrieval from the first implementing unit ▼		
(1) Internal investigation etc.					
・The Government-led program on Decommissioning and Contaminated Water Management			Continue as necessary		
・Engineering					
・Internal investigation etc.	Preparation for investigation/ Internal investigation / Characterization of fuel debris (from retrieved sampling, etc.)				
(2) Fuel debris retrieval					
・The Government-led program on Decommissioning and Contaminated Water Management			Continue as necessary		
・Engineering					
・Fuel debris retrieval		Preparation for small-scale retrieval →		- Small-scale retrieval - Preparation for expanded-scale retrieval	

: On-site operation
: Technical reviews for the on-site construction, etc., for each item
: Research and development

©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

the fuel stored in the Common Spent Fuel Storage Pool

3. Technological strategies toward decommissioning of the Fukushima Daiichi NPS (6/6) ([2] Fuel removal from spent fuel pools)

Goals

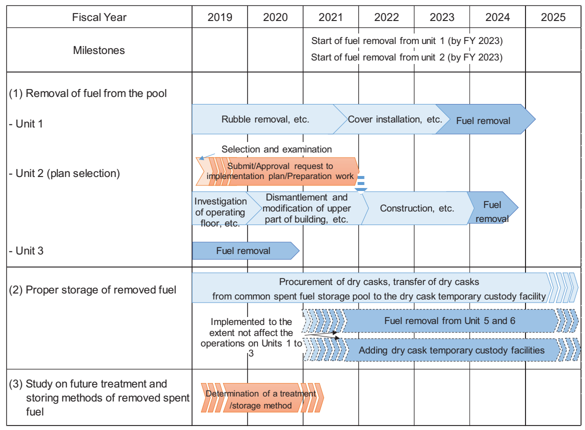
(1) Risk assessment and management for the progress of the work will be carried out properly, and measures for safety and security including the prevention of the scattering of radioactive materials will be undertaken thoroughly. **For Units 1 and 2, the removal of fuel in SFP will start by 2023. For Unit 3, the removal is scheduled to be completed in FY 2020.**

(2) By transferring the fuel stored in the common spent fuel storage pool to dry cask temporary custody facility in order to **secure the capacity** in the common spent fuel storage pool, the fuel removed from the spent fuel pools of Units 1 to 4 is to be **appropriately stored** in the common spent fuel storage pool.

(3) Based on **the assessment of the long-term integrity and investigations for future treatment of the removed fuel, the future treatment and storage methods of them will be fixed around 2020**.

Strategies and Challenges

- The evacuation order was lifted in the municipalities where the nuclear power plants is located, and efforts for the return of residents and reconstruction have been started. Careful efforts with more emphasis on safety are required.
- Unit 1: Careful removal of rubble and continued implementation of measures to prevent radioactive dust dispersion during removal
- Unit 2: In addition to the conventional method of completely dismantling the upper part of the operating floor, a method of accessing from the south side of the reactor building without dismantling, to the extent possible, is being considered
- Unit 3: The removal began this April and is expected to be completed by the end of FY 2020



: On-site operation
: Technical reviews for the on-site construction, etc., for each item

©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

4. Handling important items related to the comprehensive approach and smooth promotion of the project

Comprehensive approach to the project

- Formulate a consistent long-term plan for the entire decommissioning process covering from the current status to the short, medium, and long term plans, and comprehensively manage various efforts in line with this plan
  - Securing the lead time
  - Optimized allocation of limited resources (people, things, money, time, and space)
- In making specific use of the long term plan, set decision points at appropriate times while revising it flexibly based on the assessment of work progress etc.

Handling critical enablers for smooth operation of the project

- The government, industry and universities associated with the nuclear industry as a whole to make steady efforts to foster and secure future researchers and engineers

R&D initiatives

- TEPCO places much more importance on its own technological development that is directly linked to actual site applicability.
- Management of R&D activities associated with engineering schedule, that the R&D results will be applied to the site in a timely and appropriate manner
- It is also important for universities and public research institutes to share awareness of issues identified at the decommissioning site.

Enhancement of international cooperation

- Learning lessons from the legacy sites and treatment for preceding accident reactors to apply them to decommissioning through utilization of technologies and human resources cultivated in various nations, as a risk reduction strategy
- Establishment and strengthening of a long-term partnership with overseas decommissioning organizations
- NDF proactively participates in international joint activities, such as applying knowledge acquired from accident of Fukushima Daiichi and decommissioning process to other issues in order to respond to new concerns from overseas.

©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

5． Summary(1/3)

1． Fuel Debris Retrieval

Goals

- Safe retrieval and stable storage under full management
- Retrieval method for the first unit is to be determined in FY2019 and its actual work starts in 2021

Policy for direction progress

- 5 items are indicated→①Step by step ②Optimization of the overall decommissioning work③Combination of multiple methods④Methods focusing on partial submersion method⑤Prioritization of side access method

Strategies for success

- Approach to risk reduction and recommendation for retrieval method of the first unit are proposed.
  ・Retrieval method: an arm-type access equipment having a better outlook on the site applicability from points of view 「safe」 「reliableness」 「promptitude」
  ・The first implementing unit: unit 2 is the optimum unit from the overall decommissioning work and possible to retrieve safely, reliably, and promptly.

©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

5.Summary(2/3)

- ✓ Safe: The radiation dose level in working environment is relatively lower than other units and the airtightness with existing safety system is higher than other units.
- ✓ Reliableness: Some possible debris deposits are movable and access route to reach them has been identified.
- ✓ Promptitude: It has a possibility to start fuel debris retrieval work promptly and leads to earlier acquisition of information and experience.

2.Fuel removal from spent fuel pools

Goals

- For unit 1&2, the removal is to start in about FY2023 and for unit 3, the completion of it is aimed within FY2020 on the premise that measures have to be taken thoroughly for safety.
- The stable storage of removed fuels in the common SF pool after securing adequate capacity.
- The methods of the future storage and treatment of those fuels are to be determined in about FY2020 on the assessment of the long-term integrity and investigation for future treatment.

©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

5.Summary(3/3)

Strategies for success

- The return of residents and reconstruction have been started and accordingly careful efforts with more emphasis on safety are required.

  ・Unit 1: Careful removal of rubble and the implementation of measures to prevent dust dispersion during its removal.
  ・Unit 2: In addition to the method of completely dismantling the upper part of the operating floor, a method of accessing from the south side of RB is being considered.
  ・Unit 3: The removal is expected to be completed by the end of FY2020.

©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

End

Thanks for your kind attention!

©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

C.1.1.2. Fundamental Concept for Fuel Debris Analysis

Fundamental Concept for Fuel Debris Analysis in Japan

18-19 November, 2019
Reactor Safety Technology Expert Panel Forensics Meeting
Argonne National Laboratory (ANL) Offices, Washington, DC

Junichi Nakano
Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF)

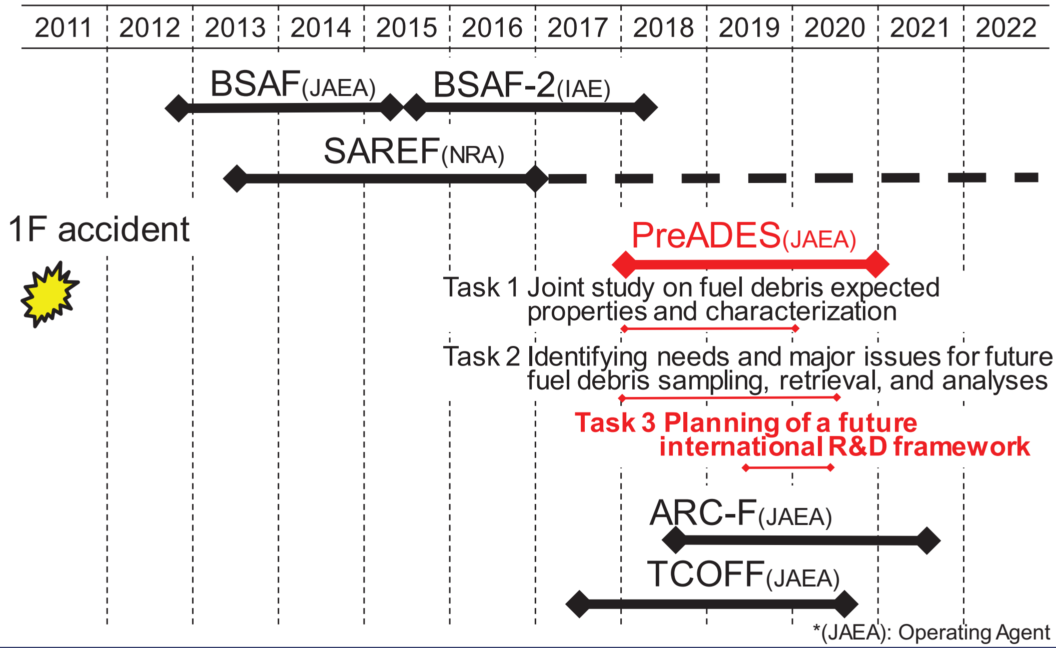
NDF
無断複製・転載禁止 原子力損害賠償・廃炉等支援機構
©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

Background

- The NDF and Tokyo Electric Power Company Holdings (TEPCO) have discussed about analysis for fuel debris and investigation at Fukushima Daiichi NPS (1F) site since January 2019 as preparation for sampling.
- In the middle of the discussion, objectives, issues and concept, etc. for fuel debris analysis are introduced.
- Please understand the issues may be changed depending on progress of the discussion and the decommissioning process.

NDF
無断複製・転載禁止 原子力損害賠償・廃炉等支援機構
©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

OECD/NEA Projects related to 1F accident



©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

Basic principles toward the analysis and investigation

(i) To proceed the decommissioning of Fukushima Daiichi NPS (hereinafter referred to as “1F”) safely and steadily is of primary importance. With these efforts, it is necessary to achieve “decommissioning as soon as possible”. For that sense, the analysis and the investigation on the decommissioning and contaminated water management of 1F (hereinafter referred to as “1F Analysis and Investigation”) should be conducted, to the extent that can proceed the decommissioning in safe and steady.

(ii) At the same time, it is also necessary to proceed the 1F Analysis and Investigation from the viewpoint of ascertaining the causes of the 1F accident and improving the nuclear safety for future (hereinafter referred to as “Forensic”). Therefore, due consideration is to be given to the necessity of the 1F Analysis and Investigation from the viewpoint of Forensic, on the premise of the safe and steady decommissioning of 1F.

(iii) The 1F Analysis and Investigation is to be planned on the premises of realistic working situations and difficulties of the site, giving the highest priority on the safety for local residents, surrounding environment and workers. In addition, it must be proposed after clarifying the concreteness of technology commensurate with it.

(iv) While clarifying what is the information obtained from the 1F Analysis and Investigation used for and what will it contributes to, it must be conducted in reasonably acceptable range as 1F decommissioning project, considering its significance and the responsibility associated with it.

(v) Taking into account of Japan’s responsibility to the international society, as a country where the 1F accident occurred, information obtained in the 1F Analysis and Investigation should be provided in a proactive way. There is a possibility for institutions requesting additional information to bear a reasonable burden.

©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

Decommissioning is the highest priority

To proceed the decommissioning of the 1F safely and steadily, it is important to understand properties of fuel debris and current condition at the 1F site. Results of analysis and investigation will contributes to the followings.

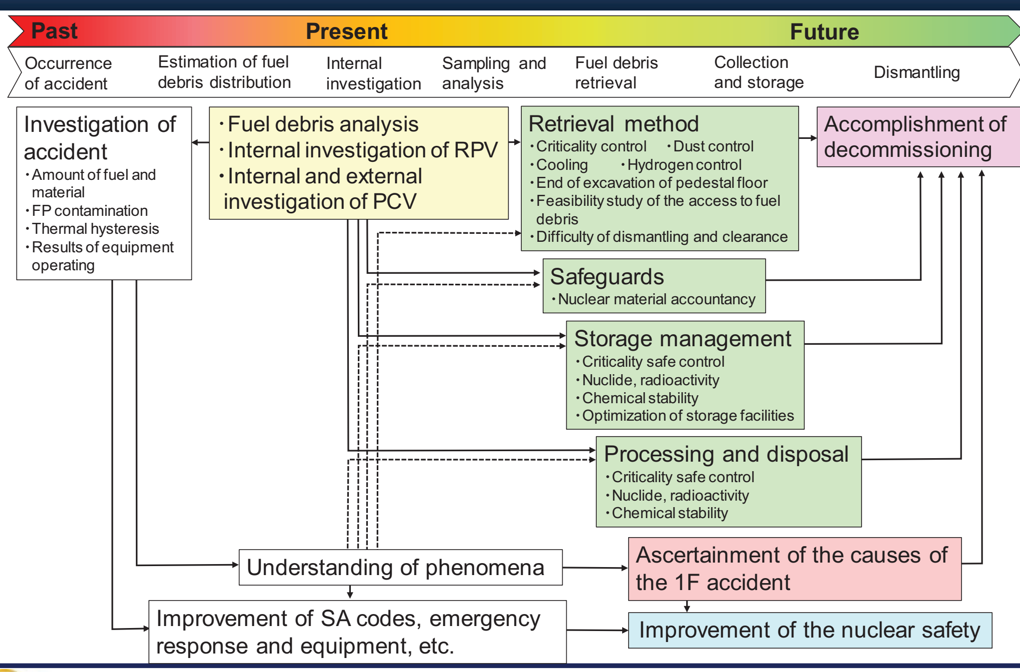
(1) For **accomplishment of decommissioning**, fuel debris analysis and investigation at the 1F site should be conducted while considering issues for retrieval method, safeguards, storage management, and processing and disposal in the decommissioning process.

(2) For **ascertainment of the causes of the 1F accident**, through understanding phenomena, investigation of the 1F accident should be conducted.

(3) **Improvement of severe accident codes** leads to the nuclear safety for future.

無断複製･転載禁止 原子力損害賠償･廃炉等支援機構
©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

Objectives for analysis and investigation



無断複製･転載禁止 原子力損害賠償･廃炉等支援機構
©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

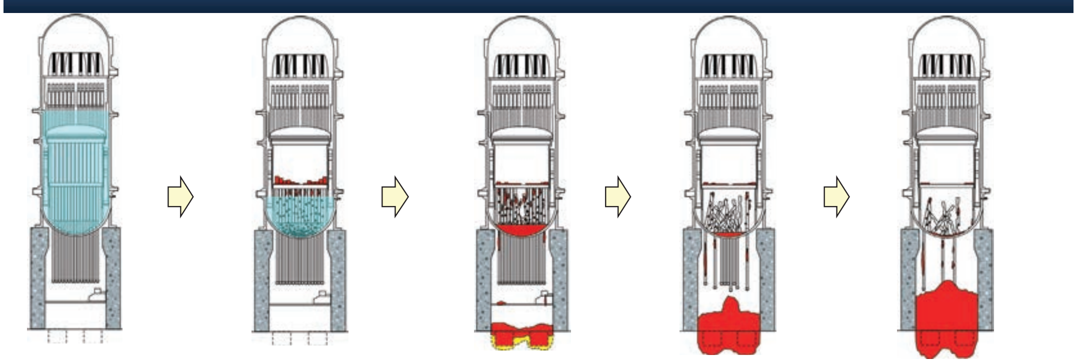
Peculiarities of accident at the 1F

- The Fukushima Daiichi NPS was the first severe accident of the Boiling Water Reactors (BWR) in the world. There was no experience of BWR until then.
- Reactor support structure, i. e. core shroud, fuel support piece, core plate assembly, etc., were included in the RPV. Internal structure is complex.
- Melting point decreased by the eutectic reaction of UO_2 with stainless steel. Melting point is unknown.
- There is no thermal record because of blackout at the 1F.
- Did water injected from fire engines reach the core area ? When did water reach the core area ?
- Does sea salt (NaCl) in sea water injected in the accident affect fuel debris characteristics?

> It is important to conduct analysis of fuel debris and investigation of the 1F site in order to understand phenomena happened at the 1F and current condition.

無断複製・転載禁止 原子力損害賠償・廃炉等支援機構
©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

Process of fuel melting and condition of each unit



①Before accident ②Water level drops down and fuel melting starts. ③RPV is damaged and fuel debris falls down.(Unit 2) ④Spill from RPV is in progress. (Unit 3) ⑤Most of fuel flows from RPV. (Unit 1)

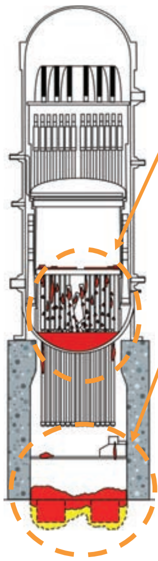
Table 1 Operating time of core cooling system in severe accident

2011	11 March	12 March	13 March	14 March
Unit 1	IC, Operating for approx.17 min			
Unit 3	RCIC, HPCI, Operating for approx. 34 hours			
Unit 2	RCIC, Operating for approx. 70 hours			

Ranking of damaged core and fuel melting is corresponding to operating time of core cooling system in severe accident.

無断複製・転載禁止 原子力損害賠償・廃炉等支援機構
©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

Issues of fuel debris analysis in Unit 2



Contents of U and Pu in 548 fuel assemblies; U:55.4%, Pu:0.4%

→ Contents are diluted depending on mixture of Zr and stainless steel.

(1) Fuel debris at the bottom of the RPV

- Content of fuel assembly (U, Pu, Gd, Zr), control rod (B, C, Fe, Cr, Ni), and structural material (Fe, Cr, Ni) according to location.
- Isotope ratio of U and Pu. Density. Particle size.
- Existence of boride phase (possibility of hardening).
- <u>Existence and amount of fission products (FP) in non-melt fuel.</u>
- Amount of fuel debris in CRGT and CRD housing.

(2) Fuel debris on the floor of pedestal

- Content of fuel assembly, control rod, and structural material.
- Isotope ratio of U and Pu. Density. Particle size.
- Content of concrete, i. e. Ca, Si, etc.
- <u>Content of object fallen on the steel grating.</u>
- <u>Original location and path of upper tie plate fallen on the pedestal.</u>
- Amount of fuel debris in drain pit.
- Depth attacked by Molten-Core-Concrete-Interaction (MCCI).

(3) Others

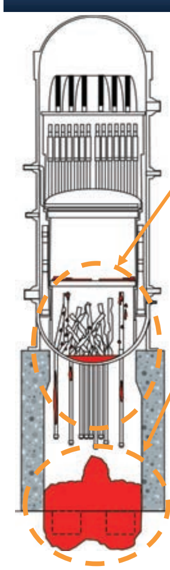
- Effects of sea water injection and existence of Na, Mg and Cl.
- Compound and amount of ^{137}Cs and ^{134}Cs.
- Ratio of O/U (degree of aging).
- In analysis of a very small amount of nuclide, it is necessary to discuss.
- Discussion of nuclide related to radioactive waste management will start from the 3rd phase.

NDF

9

on

Issues of fuel debris analysis in Unit 3



Content of U and Pu in 548 fuel assemblies; U:55.4%, Pu:0.4%

→ Contents are diluted depending on mixture of Zr and stainless steel.

※ 32 MOX assemblies were loaded. Based on inventory estimation, contents of U and Pu are the same as those loaded in Unit 2.

(1) Fuel debris at the bottom of the RPV

- Content of fuel assembly (U, Pu, Gd, Zr),Control rod (B, C, Fe, Cr, Ni), and structural material (Fe, Cr, Ni) according to location.
- Isotope ratio of U and Pu. Density. Particle size.
- Existence of boride phase (possibility of hardening).
- Amount of fuel debris in CRGT and CRD housing.

(2) Fuel debris on the floor of pedestal

- Content of fuel assembly, Control rod, and structural material.
- Isotope ratio of U and Pu. Density. Particle size.
- <u>A projection from the center of the floor (CRD changing machine may be buried?).</u>
- Existence of boride phase (possibility of hardening).
- Content of concrete, i. e. Ca, Si, etc.
- Depth attacked by Molten-Core-Concrete-Interaction (MCCI).

(3) Others

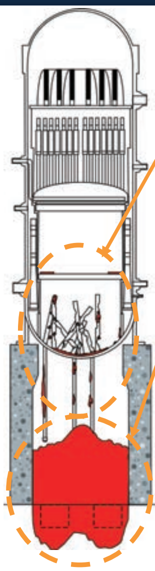
- Effects of sea water injection and existence of Na, Mg and Cl.
- Compound and amount of ^{137}Cs and ^{134}Cs.
- Ratio of O/U (degree of aging).
- In analysis of a very small amount of nuclide, it is necessary to discuss.
- Discussion of nuclide related to radioactive waste management will start from the 3rd phase.

NDF

10

無断複製・転載禁止 原子力損害賠償・廃炉等支援機構
©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

Issues of fuel debris analysis in Unit 1



Contents of U and Pu in 400 fuel assemblies; U:55.5%, Pu:0.4%
→ Contents are diluted depending on mixture of Zr and stainless steel.

(1) Fuel debris at the bottom of the RPV
- Content and compound of a small amount of fuel debris which adhered to the bottom of the RPV.
- Isotope ratio of U and Pu. Density. Particle size.
- Existence of boride phase (possibility of hardening).
- Amount of fuel debris in CRGT and CRD housing.

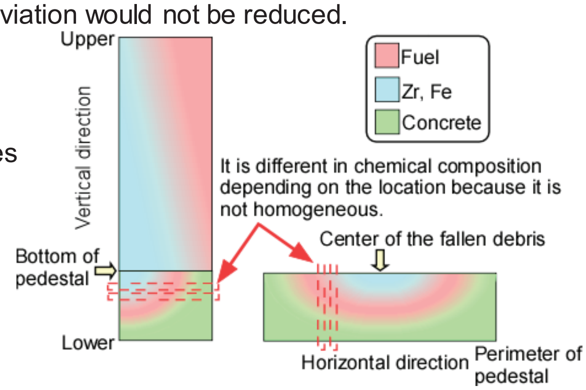
(2) Fuel debris on the floor of pedestal
- Content of fuel assembly (U, Pu, Gd, Zr), control rod (B, C, Fe, Cr, Ni) and structural material (Fe, Cr, Ni) according to location.
- Isotope ratio of U and Pu. Density. Particle size.
- Existence of boride phase (possibility of hardening).
- Content of concrete, i. e. Ca, Si, etc.
- Depth attacked by Molten-Core-Concrete-Interaction (MCCI).

(3) Others
- Effects of sea water injection and existence of Na, Mg and Cl.
- Compound and amount of ^{137}Cs and ^{134}Cs.
- Ratio of O/U (degree of aging).
- In analysis of a very small amount of nuclide, it is necessary to discuss.
- Discussion of nuclide related to radioactive waste management will start from the 3rd phase.

無断複製・転載禁止 原子力損害賠償・廃炉等支援機構
©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

Standard deviation of results of analysis

- To improve precision of analysis, in general, many samples are obtained and standard deviation is reduced. However, time, resource and cost are needed according to the number of samples. It is considered that results may not be gotten timely.
- Molten fuel catches internal structure, falls down and spreads in severe accident. Depending on location, temperature, and reaction process, concentrations of contents are changing seamlessly. Fuel debris is not homogeneous. If many samples were obtained, standard deviation would not be reduced.
- It is important to consider results of analysis with information of location comprehensively, since results of analysis show contents and properties depending on picking location.



Conceptional illustration of fuel debris

無断複製・転載禁止 原子力損害賠償・廃炉等支援機構
©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

Conclusions

- The 1F decommissioning is the highest priority in all objectives for fuel debris analysis. Results of fuel debris analysis and investigation at the 1F are expected to contribute to ascertaining the causes of the 1F accident and improving the nuclear safety for future.
- Basic issues of fuel debris analysis in each unit were considered. They will be discussed.
- It is important to understand results of analysis with exact information of location.

Thank you for your attention!

無断複製・転載禁止 原子力損害賠償・廃炉等支援機構
©Nuclear Damage Compensation and Decommissioning Facilitation Corporation

C.1.2. Tokyo Electric Power Holdings, LLC

C.1.2.1. Current status and recent investigations

Current status and recent investigation result of Fukushima Daiichi

2019/11/18

Shinya MIZOKAMI
Tokyo Electric Power Company Holdings, Inc.

1. Future and recent investigation plan

- To obtain the information about conditions inside the PCV, the characteristics of fuel debris and the potential impacts of removal process is vital for fuel debris removal. Therefore, the following sequence of tasks by which the scale of operations will be gradually enlarged shall be employed.
  "Primary Containment Vessel (PCV) internal investigation (including sampling)"→"Small-scale removal of fuel debris"→"Large scale removal of fuel debris".
- As a first step of the PCV internal investigation, the implementation of further investigations is being considered to obtain additional knowledge about each unit in preparation for fuel debris removal.

  【Unit 1】
  - PCV internal investigation by inserting a submersible boat-like access/investigation device through the X-2 penetration (scheduled for FY2019)

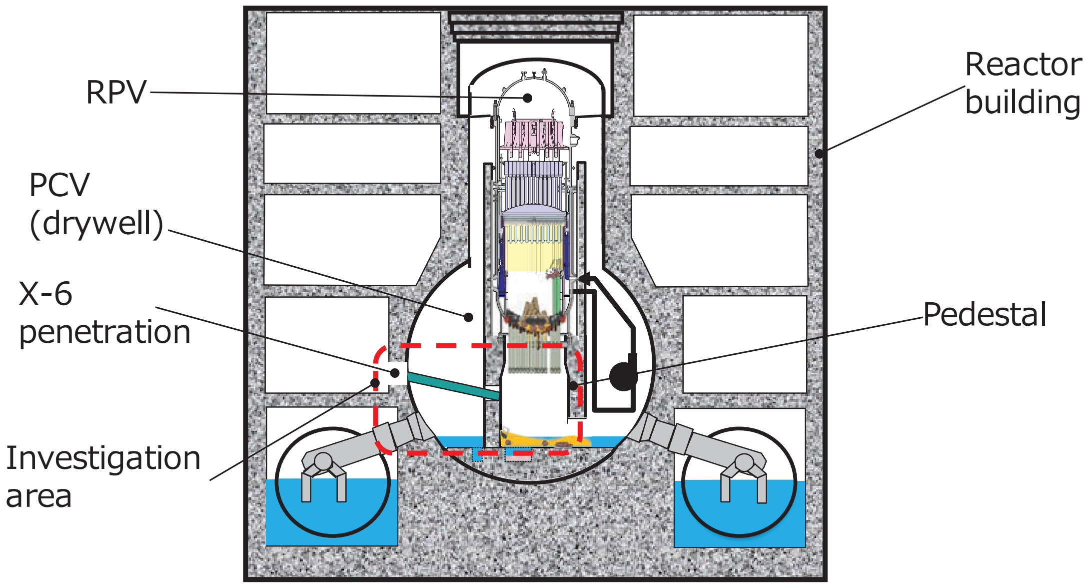
  【Unit 2】
  - PCV internal investigation using the guide pipe through the X-6 penetration used for the investigation in January, 2018 (completed 2019.2.13)
  - PCV internal investigation through the X-6 penetration using an arm-like access/investigation device (scheduled for FY2019)

  【Unit 3】
  - Methods for lowering water levels are being examined in preparation for fuel debris removal. At the same time, whether or not additional investigations using the submersible ROV used for the previous investigation are necessary is also being examined.
- A small amount of deposit samples will be collected from the bottom of the PCV during the PCV internal investigations of Units 1 and 2 scheduled for FY2019.

1

2. Investigation area in unit 2 PCV investigation(1/6)

TEPCO

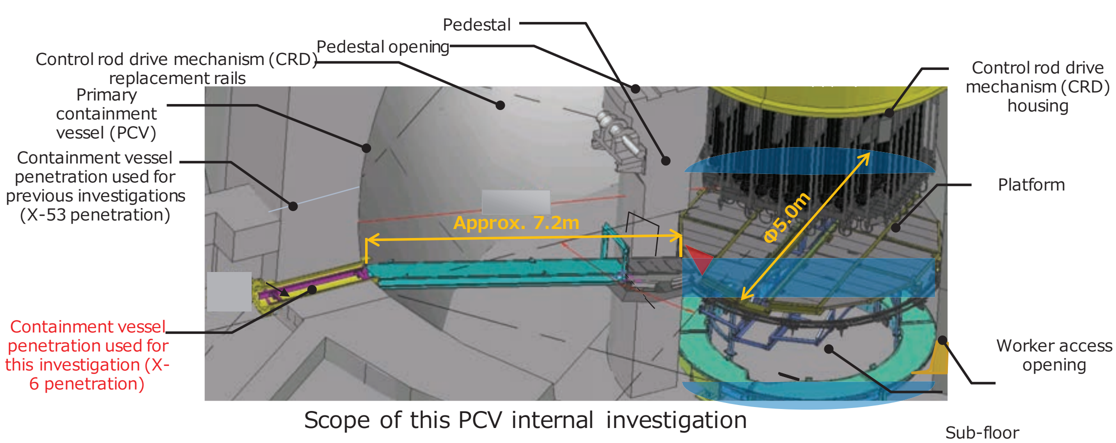


2

IRID
TEPCO

2. Investigation area in unit 2 PCV investigation(2/6)

【Investigation Plan】: The area underneath the platform, where fuel debris may exist, was checked.



Scope of this PCV internal investigation

C.1.2.2. 1F1 Water Suspension Tests

Reactor Cooling Water Temporary Suspension Test at Unit 1 -rapid communication-

2019/1 1 /18

TEPCO

Shinya MIZOKAMI
TEPCO HD

Summary

TEPCO

- Objective ot the test
  - ✓ Modification of the emergency procedure in 1F
    (Current procedure is based on much conservative evaluation)
  - ✓ To confirm real plant reaction when water injection was stopped.
  - ✓ It is expected that obtained data bring the information for reasonable evaluation of temperature, such as energy transportation to gas phase, etc.。
- Test condition and results
  - ✓ The suspension period was about 49 hours. And it was confirmed that RPV temperature behavior was gradual.
    - ➢ Temperature rise: very small
    - ➢ No side effects on radioactive dust, noble gas(Xe 135), etc.

	RPV bottom	PCV
No injection period (10/15 11:00～10/17 12:00)	0.2℃	0.6℃
Whole testing period (10/15 11:00～10/30 14:00)	0.4℃	0.7℃

- Future plan
  - ✓ Detail evaluation by comparing obtained data and current BE energy balance model.
  - ✓ Similar test in unit 3 will be conducted until March 2020.

1

Test results（rapid communication）

TEPCO

- Unit 1 cooling water injection was suspended about 49 hours.
  Temperature increasing rate in RPV and PCV was around 0.01℃/h.
  There was no unexpected reaction and abnormal temperature increase.

<Testing operation>

- 2019/10/15 10:41～10:54 3.0 m³/h → 0.0 m³/h
- 2019/10/17 11:37～11:48 0.0 m³/h → 1.5 m³/h

- 2019/10/21 10:09 1.6 m³/h → 2.1 m³/h
- 2019/10/23 10:03 2.1 m³/h → 2.5 m³/h
- 2019/10/24 14:09～14:19 2.5 m³/h → 3.0 m³/h

<temperature increasing rate during the test (2019/10/15～10/17)>

Increasing rate	Measured temperature			TC
Around 0.01℃/h	26.5℃ (10/15 11:00)	→	27.0℃ (10/17 12:00)	TE-1625T7

<RPV cooling condition>

- We set the limiting condition that temperature increase is less than 15 ℃. However, every thermo couples never reached the limit.

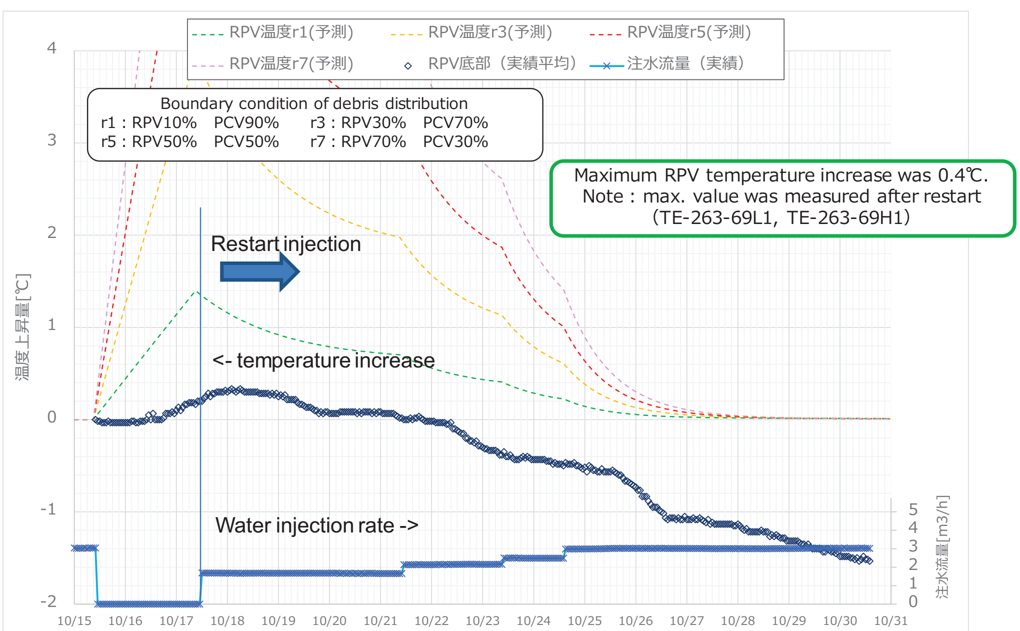
<other behavior>

- No significant increase on dust measurement in PCV gas treatment system.
- Short live noble gas, such as Xe-135, indicator of criticality, never increased.

2

RPV bottom temperature behavior

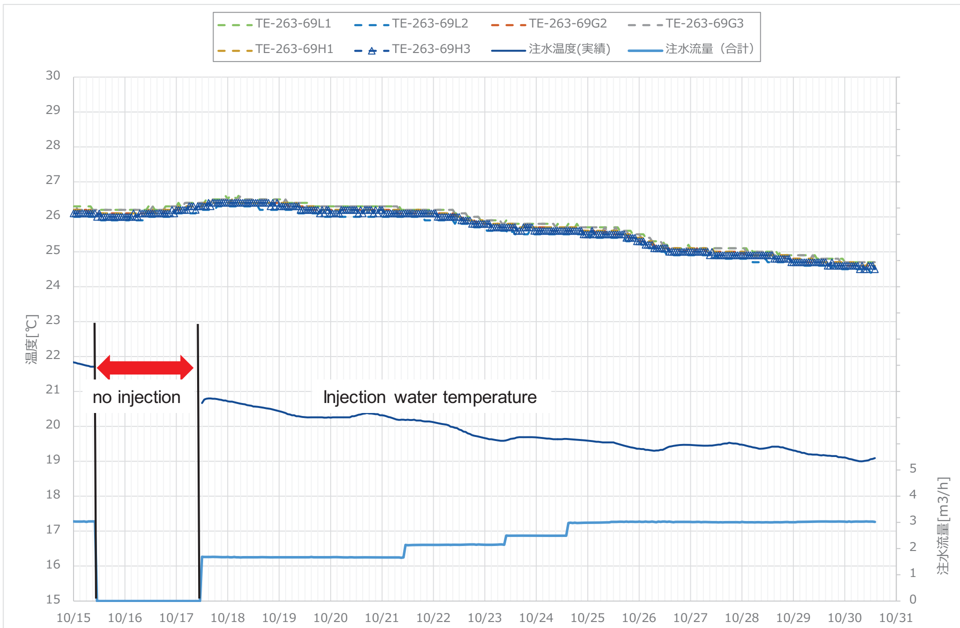
TEPCO



3

RPV bottom temperature behavior (measured value)

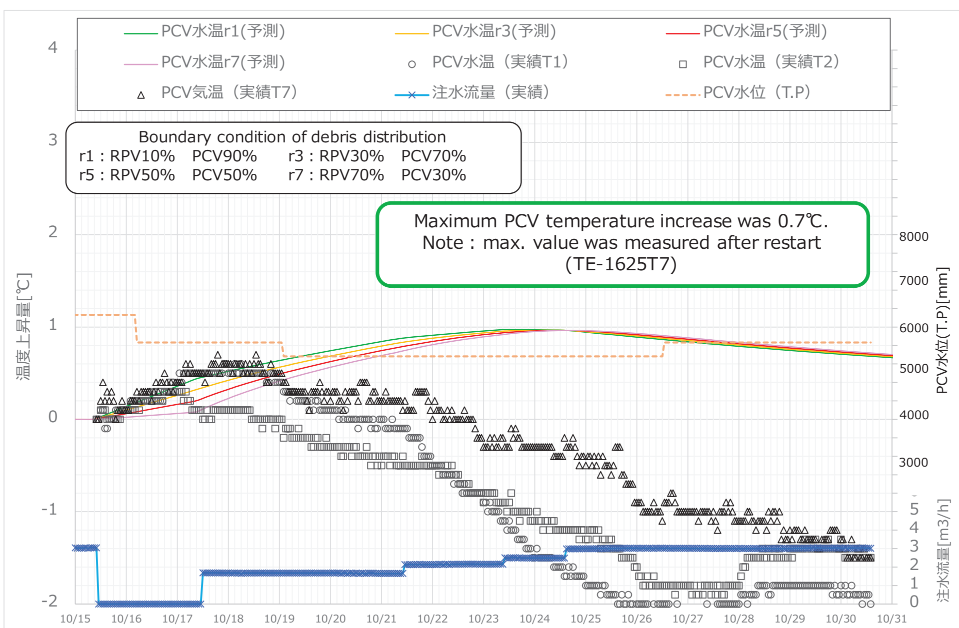
TEPCO



4

PCV temperature behavior

TEPCO

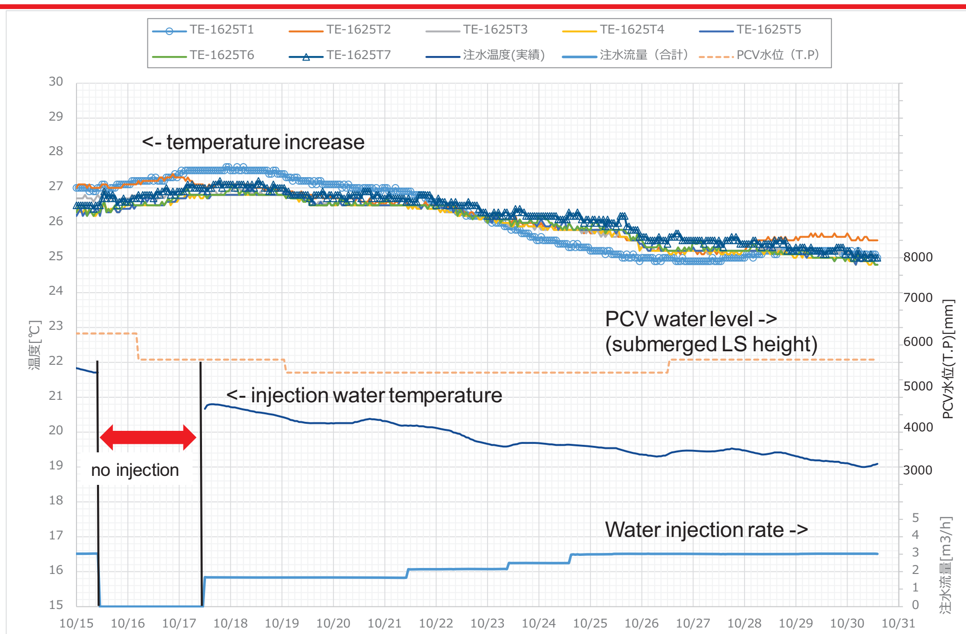


※予測温度は試験開始時の実績温度を基準として記載 ・PCV水位は水没している上端の水位計を記載

5

PCV temperature behavior (measured value)

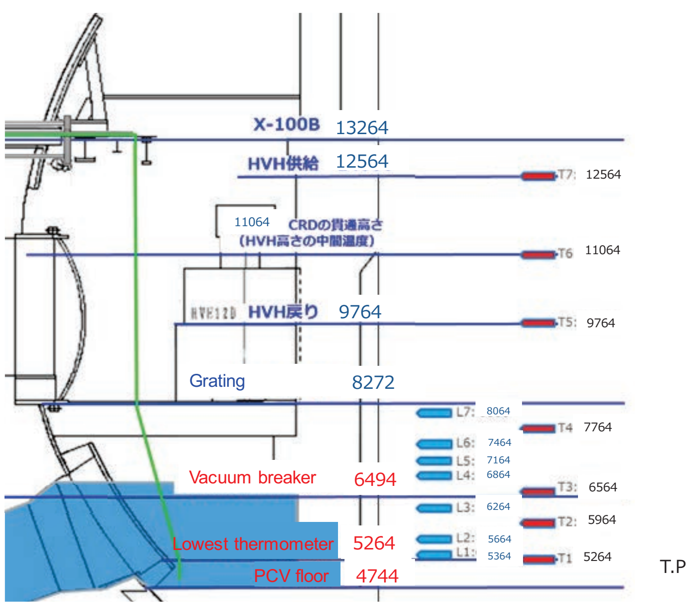
TEPCO



6

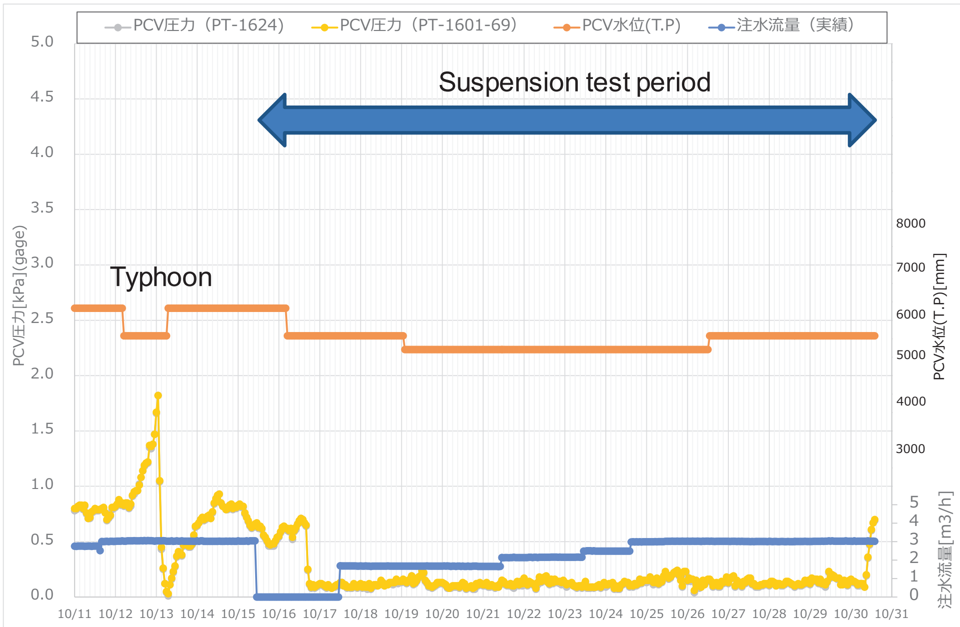
(ref) Unit 1 water level LS and thermometer arraignment

TEPCO



7

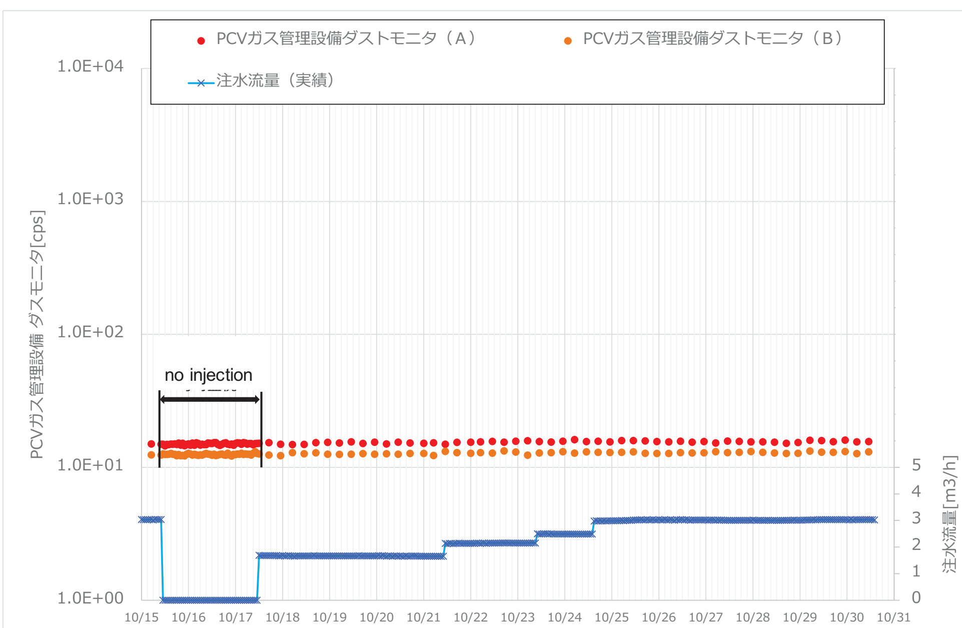
PCV pressure behavior



・PCV水位は水没している上端の水位計を記載　水位はT.Pで記載

8

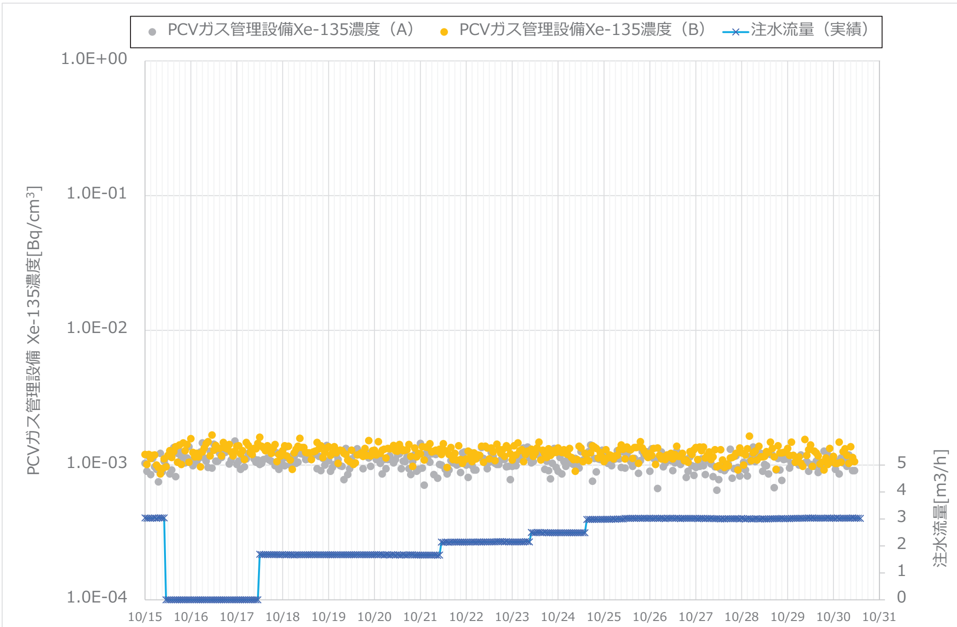
(ref.) Dust monitor behavior



9

(ref.) Noble gas monitor behavior

TEPCO



10

TEPCO

11

C.1.2.3. Severe Accident Modeling

Findings on Fukushima Dai-ichi NPP Severe Accident and Implication to SA Code Validation

August 20, 2019

TEPCO

18th International Topical Meeting on Nuclear Reactor Thermal Hydraulics
Marriott Portland Downtown Waterfront, Portland OR, United States

Shinya MIZOKAMI
Tokyo Electric Power Company Holdings, Inc.

1

1.1 Introduction

TEPCO

- On March 11, 2011, Great East Japan Earthquake(GEJE) and successive Tsunami hit the Fukushima Daiichi NPS(1F). These caused the total Station Blackout, loss of AC/DC power. Then, unit 1 to 3 lost all cooling functions and fell into severe accident.
- Just after the accident, nobody could explain the accident progressions correctly. One of the reason is that the consequence of accident was different from knowledge accumulated before 2011.
- Now, 8 years past since the accident. Researchers in the nuclear field made massive efforts to understand the 1F accident by data analysis, code calculation, robot investigation, etc.. Therefore, we know the accident progression better than before.

2

1.1 Introduction

TEPCO

- BSAF, Benchmark Study on Accident of Fukushima Daiichi, is the OECD/NEA benchmark project.

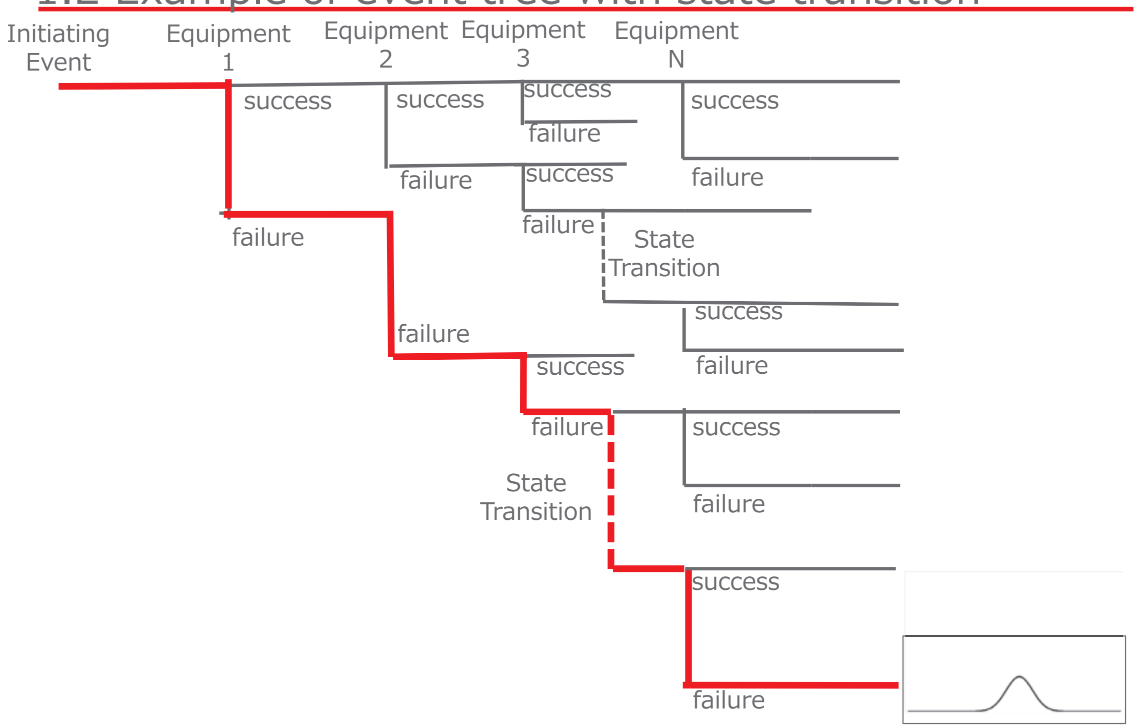
- Phase 1 completed 2015 and Phase 2 completed 2018. Phase2 summary report will be published soon.
  Phase1: https://www.oecd-nea.org/nsd/docs/2015/csni-r2015-18.pdf
- Phase 1 was conducted as like as blind test, because there were little information about accident progression.
- During the phase 2, many information was provided from Fukushima Daiichi such as robot investigation, muon tomography, etc. These helped to reduce scenario uncertainty.
- You can confirm BSAF achievement in BSAF session.

- In this presentation, I will show state of the art knowledge of the accident. And I will talk about implication to SA code validation.

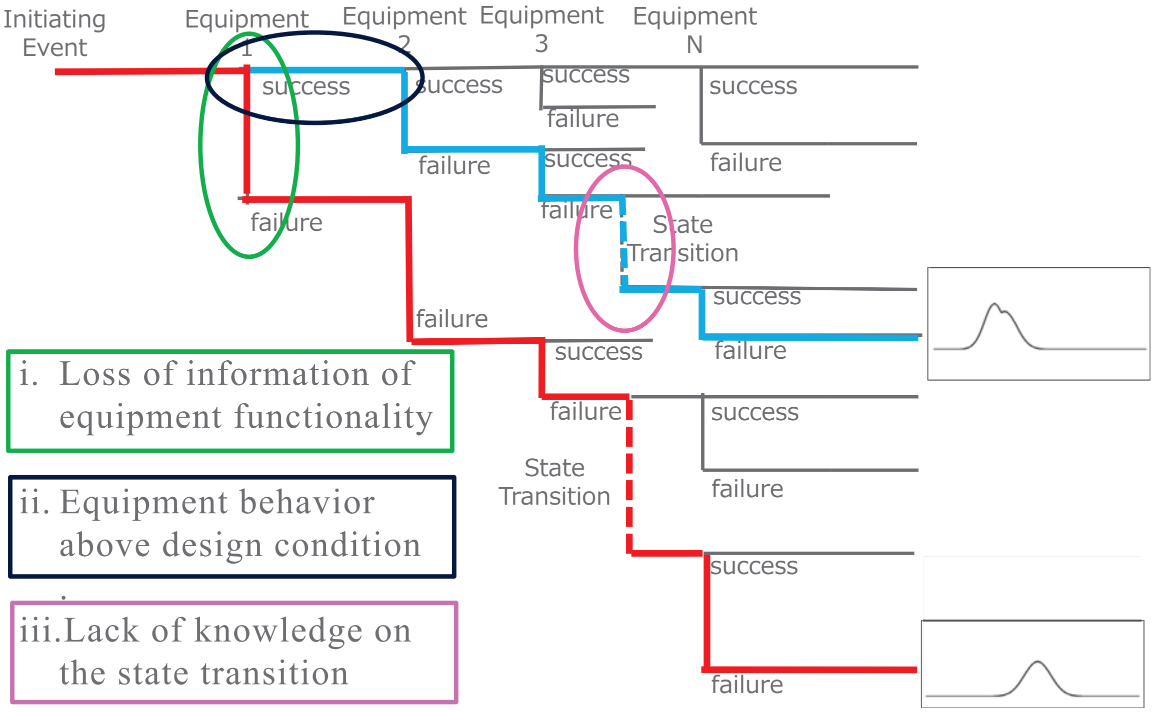
3

1.2 Example of event tree with state transition

TEPCO



1.3 Focusing point in the event tree



1.4 1F accident analysis and V&V

Model V&V

ASME V&V series, AESJ standard, there are many methodologies for V&V. However, Model V&V should be applied for the code calculation to the problem that the scenario is well defined.

It seems that 1F accident can not be applied V&V methodology.

Product V&V

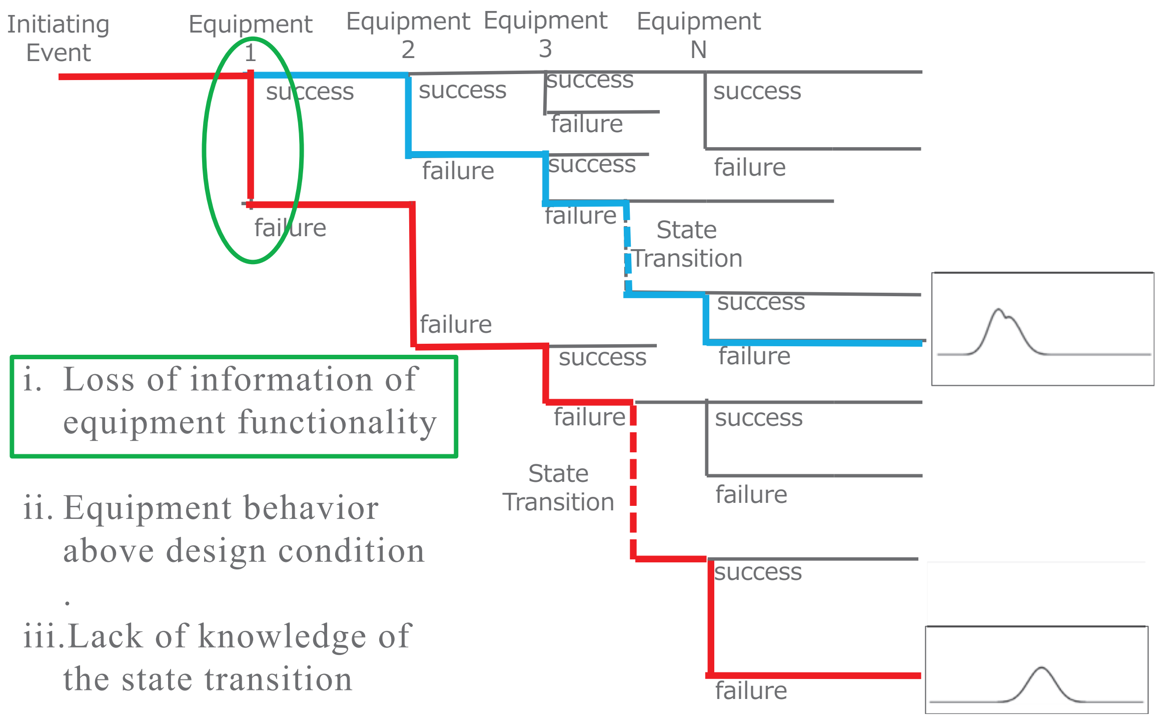
"**Validation.** The assurance that a product, service, or system meets the needs of the customer and other identified stakeholders. It often involves acceptance and suitability with external customers. Contrast with verification."

"**Verification.** The evaluation of whether or not a product, service, or system complies with a regulation, requirement, specification, or imposed condition. It is often an internal process. Contrast with validation."

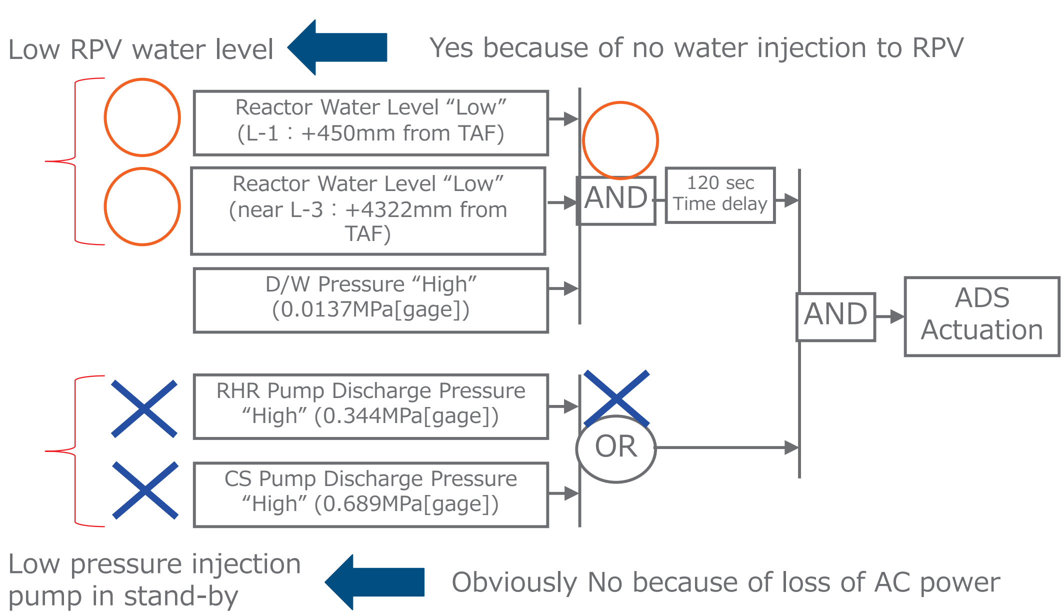
·········IEEE PMBOK guide

Utility needs the code which can explain 1F accident reasonably. And key is Phenomena Identification including state transition.

2.1 Loss of information of quipment functionality



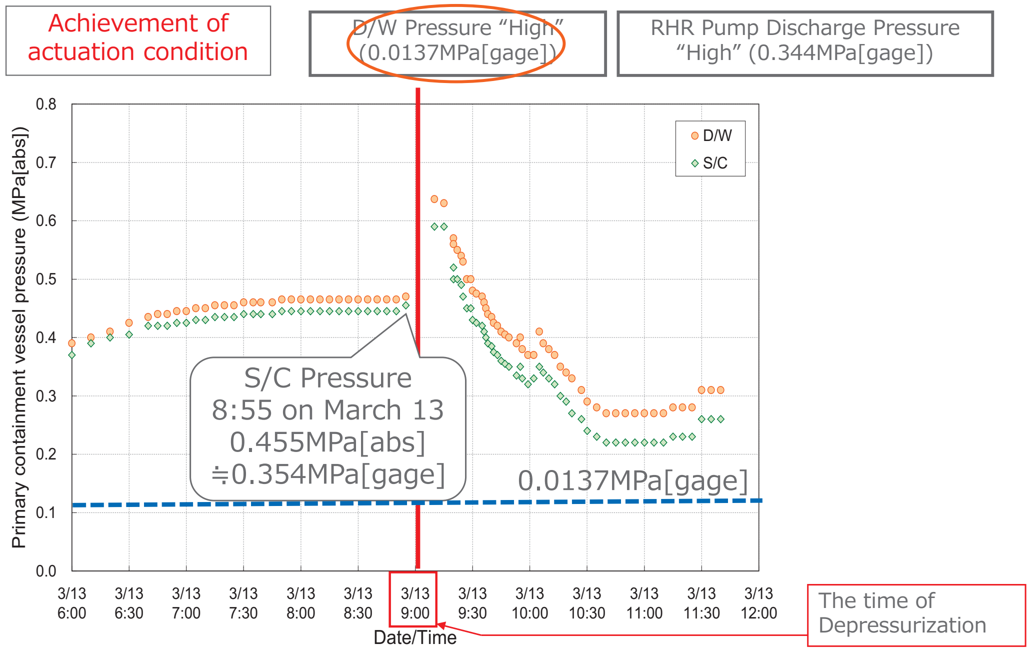
2.2 Whether ADS was activated or not in unit 3?



※Normal water level : 5327mm from TAF

2.3 PCV pressure behavior around depressurization

TEPCO

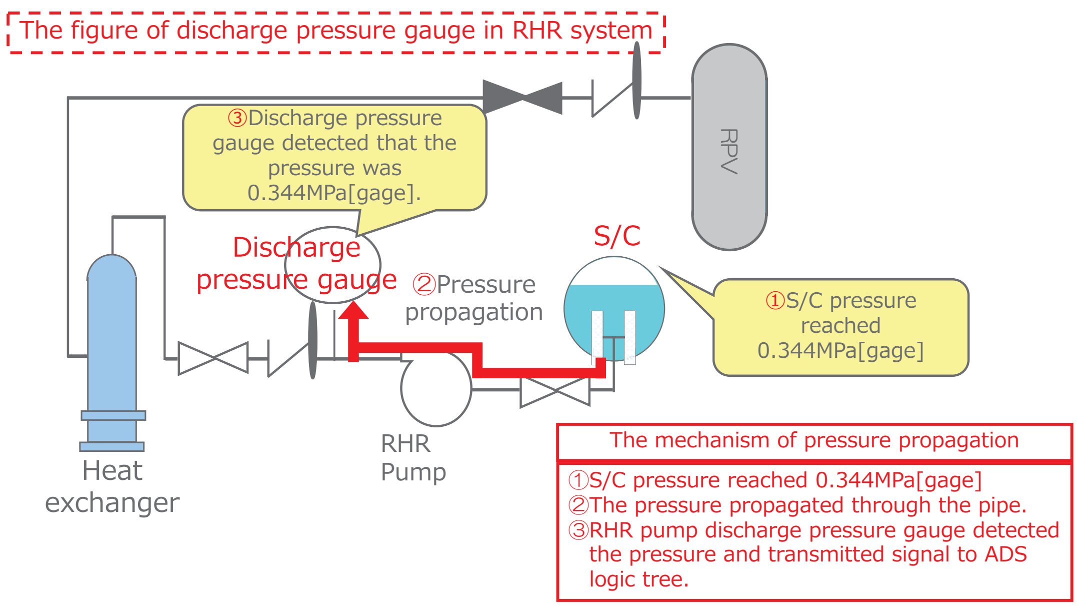


8

2.4 Discharge pressure of RHP pump

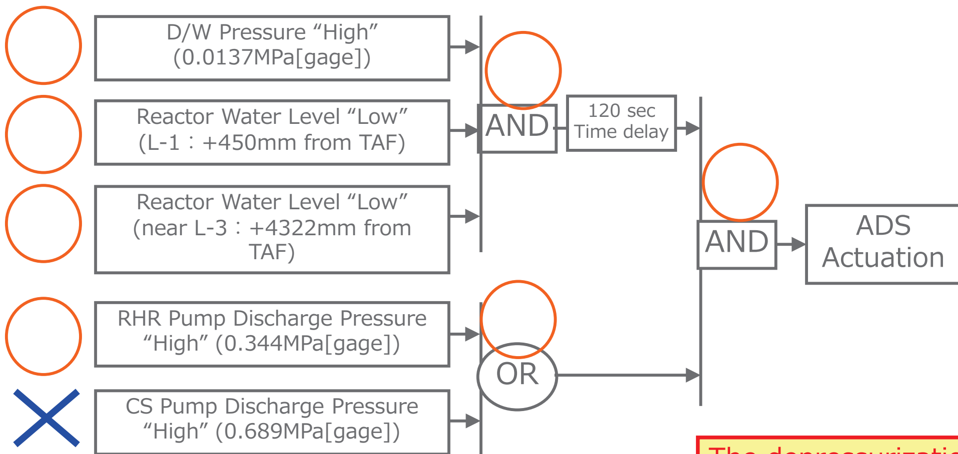
TEPCO

Actuation condition of ADS "Establishment of RHR pump discharge pressure"



9

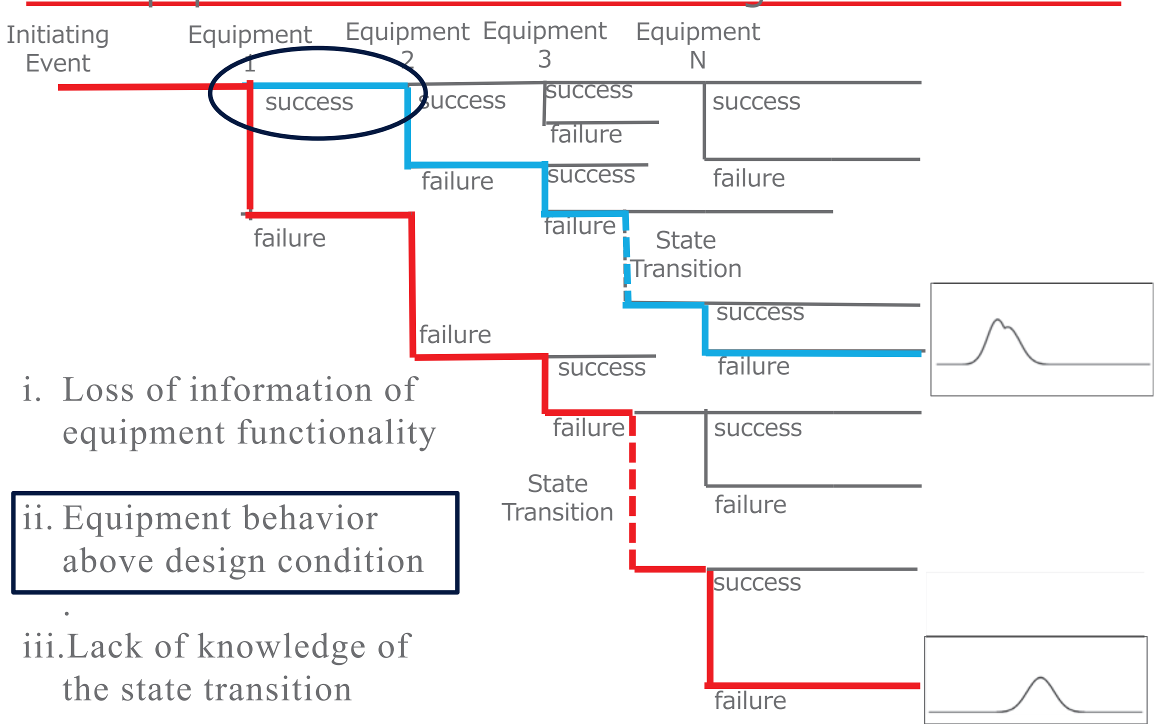
2.5 ADS was activated in unit 3



- At around 9:00 on March 13, electrical source of SRV control panel might be remained, because the SRV condition indicator lamp blinked on and off.
- At around 9:00 on March 13, it is possible that nitrogen in accumulator necessary for ADS actuation remained enough for ADS actuation, because ADS actuation had not worked after the accident probably.

The depressurization of RPV pressure at around 9:00 on March 13 might be caused by ADS actuation of SRVs.

3.1 Equipment behavior above design condition



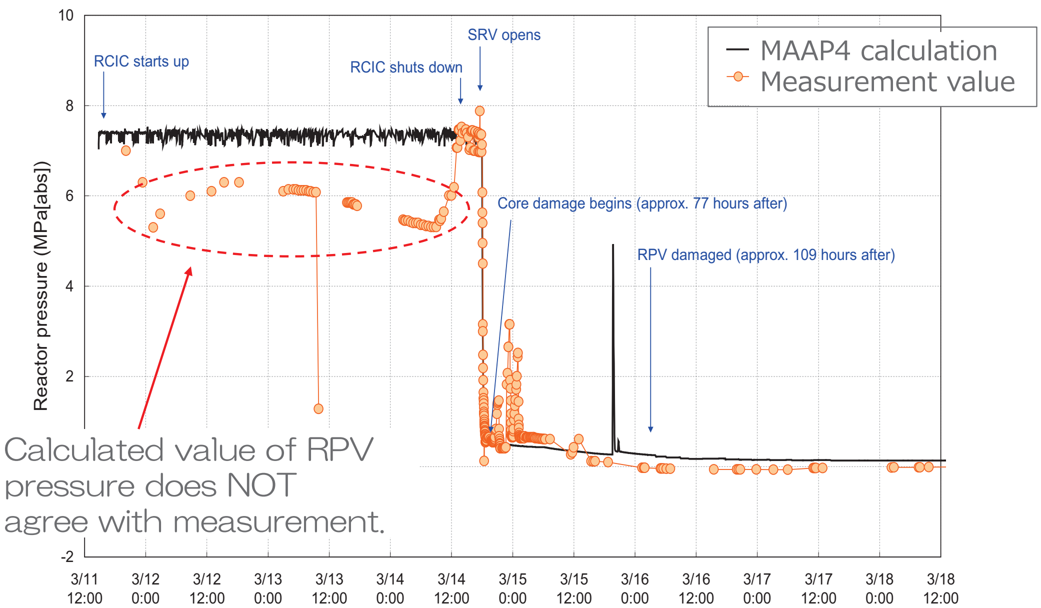
i. Loss of information of equipment functionality

ii. Equipment behavior above design condition

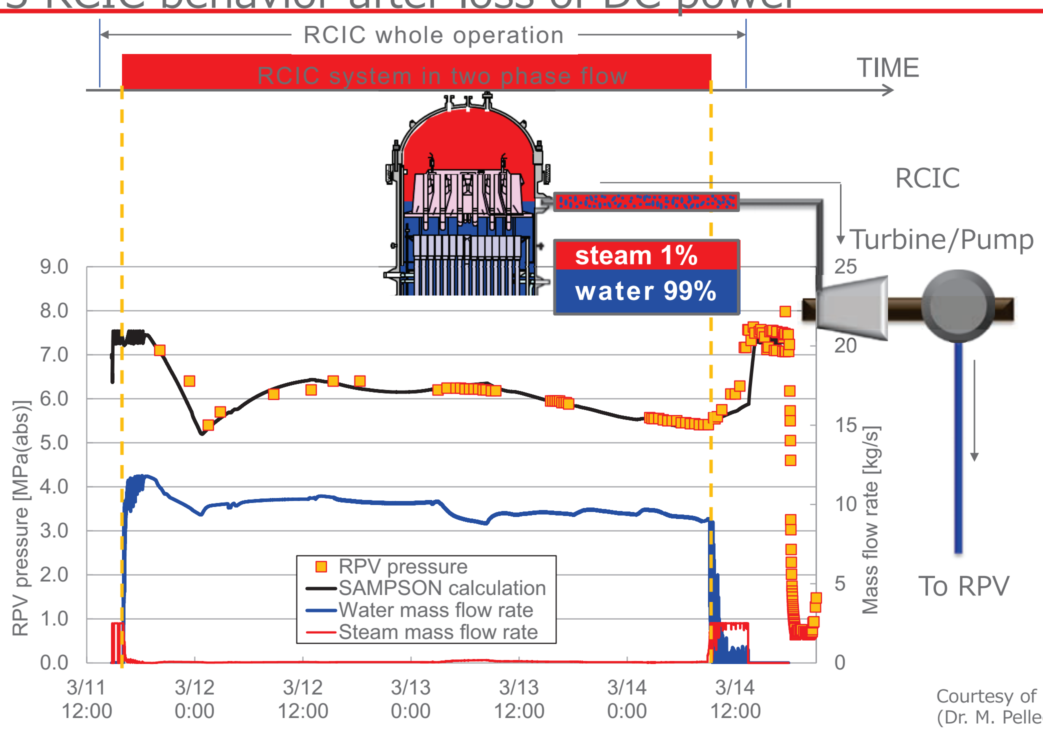
iii. Lack of knowledge of the state transition

3.2 How RCIC system worked in unit 2?

Unit 2 RCIC continued water injection about 70 hours after the Earthquake, even though DC power was not provided to RCIC system.

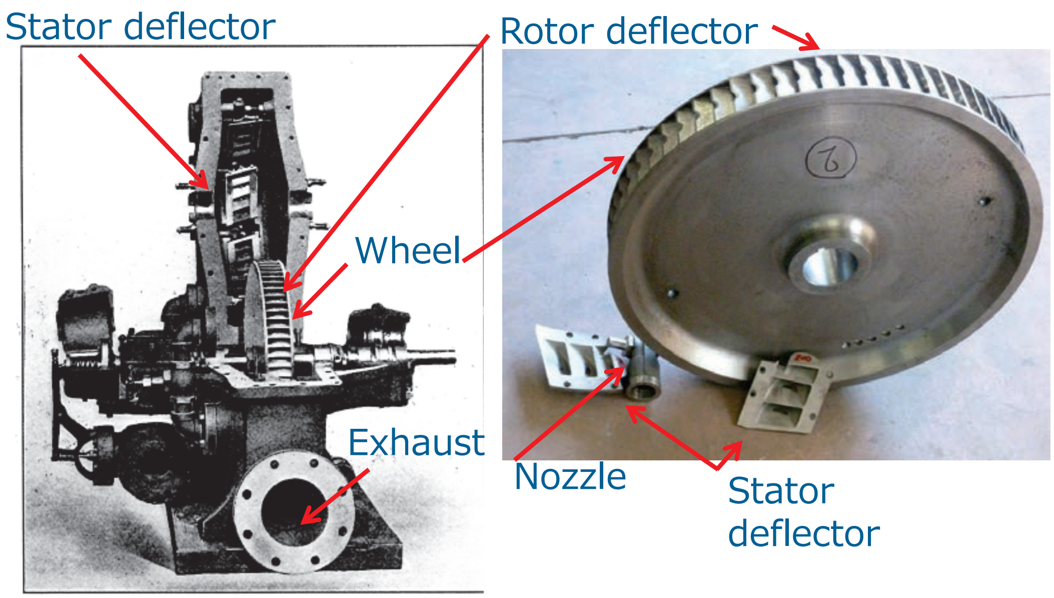


3.3 RCIC behavior after loss of DC power



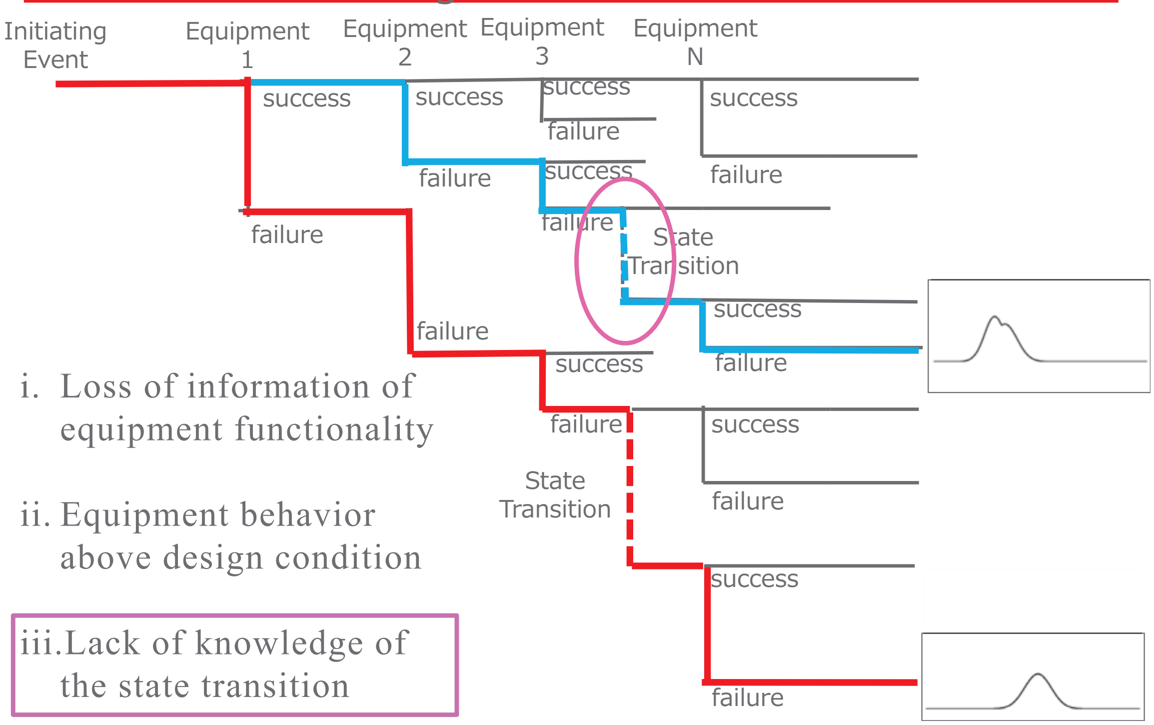
Courtesy of IAE (Dr. M. Pellegrini)

3.4 RCIC turbine system



This simple system can continue operation even by water rich steam.
There is a possibility to enhance nuclear safety by using this knowledge.
Of course, it is required to develop the CET database outside of Tech. Spec..

4.1 Lack of knowledge of the state transition

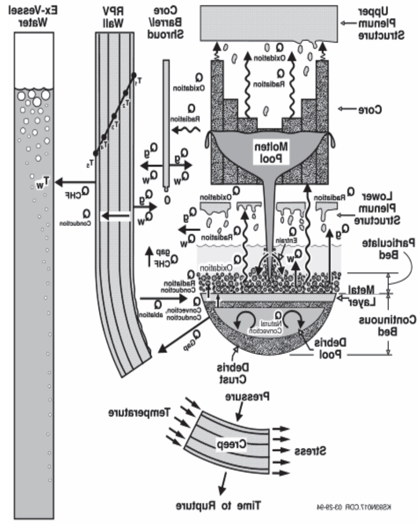


i. Loss of information of equipment functionality

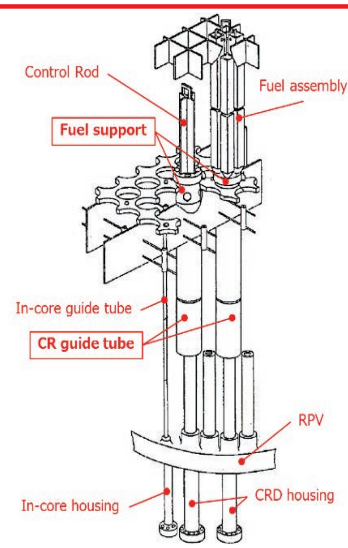
ii. Equipment behavior above design condition

iii. Lack of knowledge of the state transition

4.2 How RPV breached in Fukushima Daiichi reactors?



RPV model in MAAP



Structure inside and outside RPV in real world

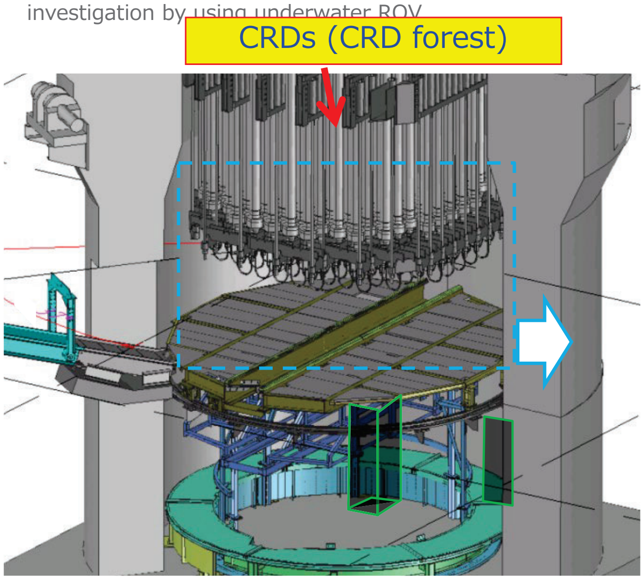
Because there is no BWR RPV break model with CRD housing on RPV bottom, CRD ejection scenario was proposed just after accident.

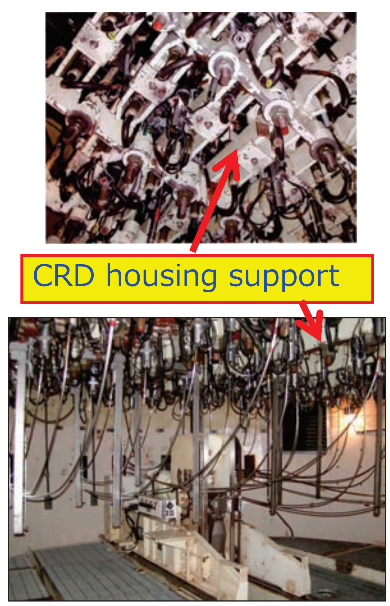
16

17

(Ref) Structure inside PCV

- Structures inside PCV can become obstacles for investigation by using underwater ROV





※Unit 2 before accider

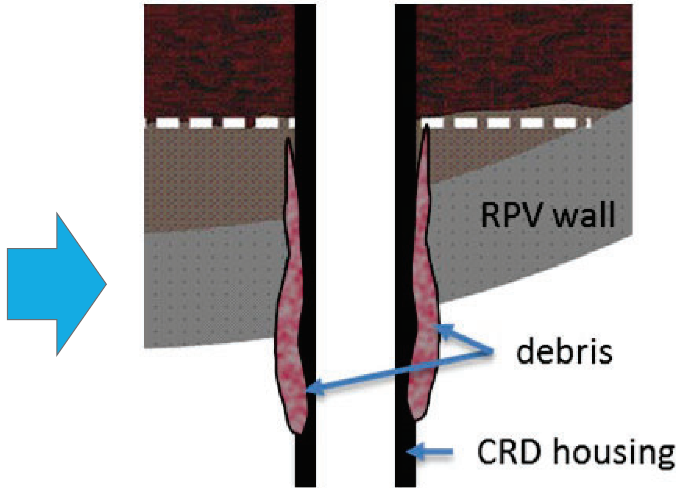
18

(ref) Inside Unit 5 PCV pedestal

4.3 Unit 2 PCV investigation result

Photos provided by: International Research Institute for Nuclear Decommissioning (IRID) 19

4.4 First revise of RPV breach model



Unit 2 investigation showed that CRD still exist and supported by supporting structure. Therefore, we introduced the debris slumping model through small gap between CRD housing and RPV wall in 2016.

4.4 Upper tie plate on PCV floor

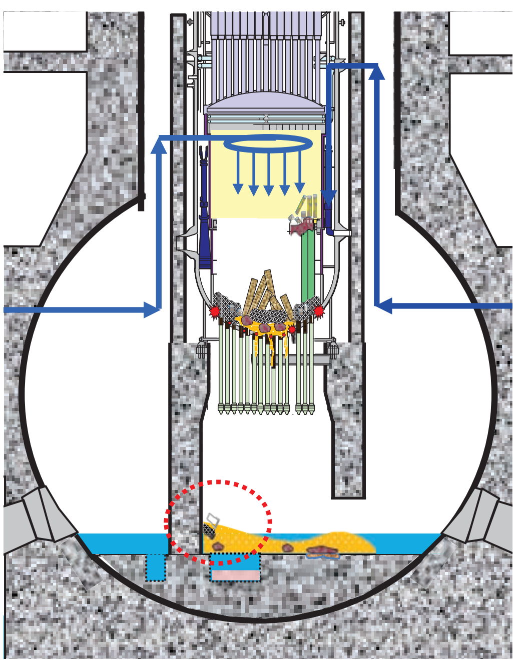
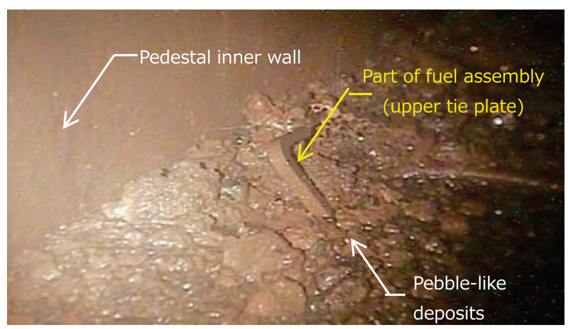


Image source: International Research Institute for Nuclear Decommissioning (IRID)
Image rendering: Tokyo Electric Power Company Holdings, Inc.



4.5 Unit 3 PCV internal investigation

IRID 22 TEPCO

The investigation inside the pedestal (VT) was conducted using a submersible remotely operated vehicle (submersible ROV) in July, 2017.

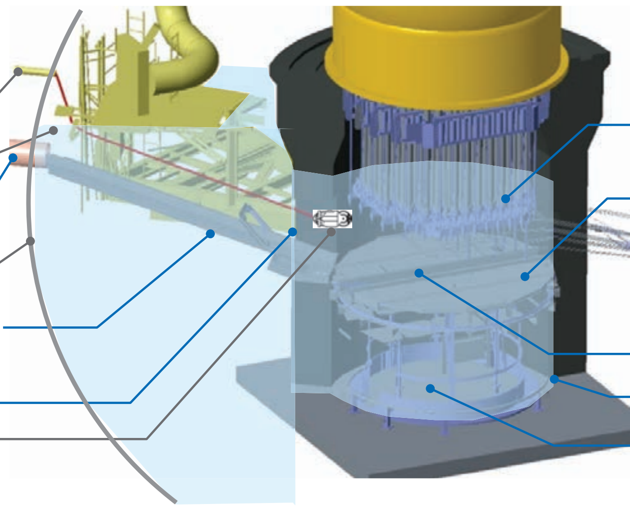
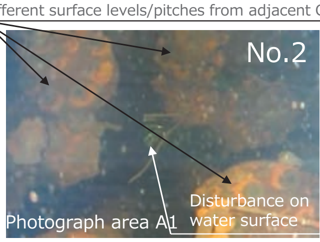


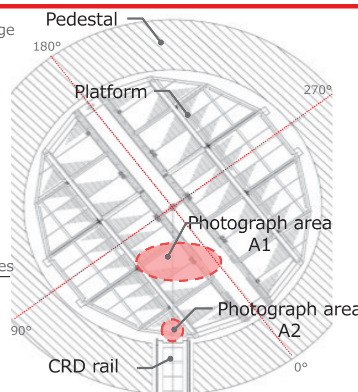
Diagram of the investigation

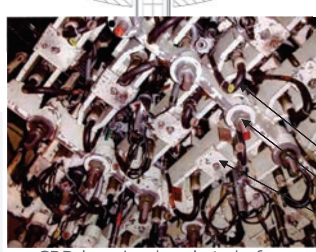
4.6 Unit 3 investigation results near the CRD housing

IRID 23 TEPCO

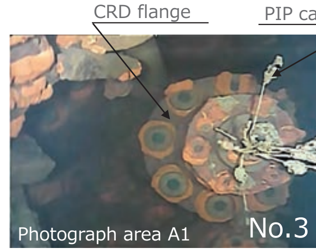
Different surface levels/pitches from adjacent CRD flanges



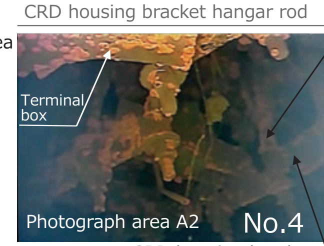




CRD housing brackets before the accident (Unit 3)



CRD housing bracket hangar rod



<Camera angle : All Upward>

- The CRD housing brackets were damaged or have fallen off at multiple points.
- CRD flange surfaces have different level and pitch from adjacent ones.
- A disturbance on the water surface was seen from a gap in the CRD housing (suggesting that water may be dripping from above) (Refer to Supplement 1 for the other places where disturbances on the water surface were observed) .

Image provided by International Research Institute for Nuclear Decommissioning (IRID)

4.7 Short movie (CRD and fluctuation of water surface)

24

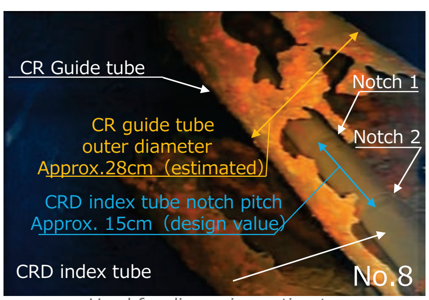
TEPCO

Image from the International Research Institute for Nuclear Decommissioning (IRID)
Image processing by Tokyo Electric Power Company Holdings

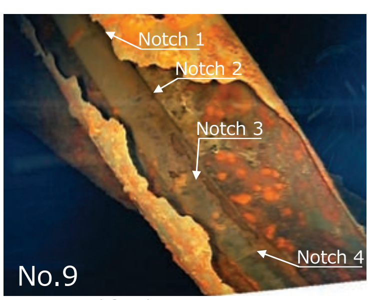
4.8 Observation of the CR guide tube outside RPV

IRID 25

TEPCO



Used for dimension estimate



Used for dimension estimate

Equip-ment	Material	Melting point
CR guide tube	Stainless Steel (SUS304)	Approx. 1450℃
CRD index tube	Stainless Steel (XM-19) (nitridization)	Approx. 1450℃

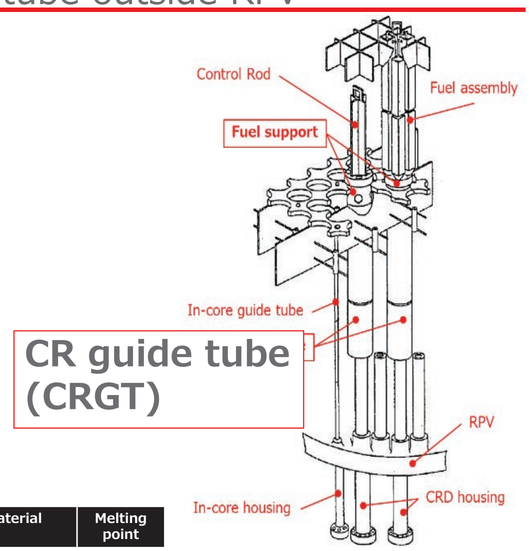
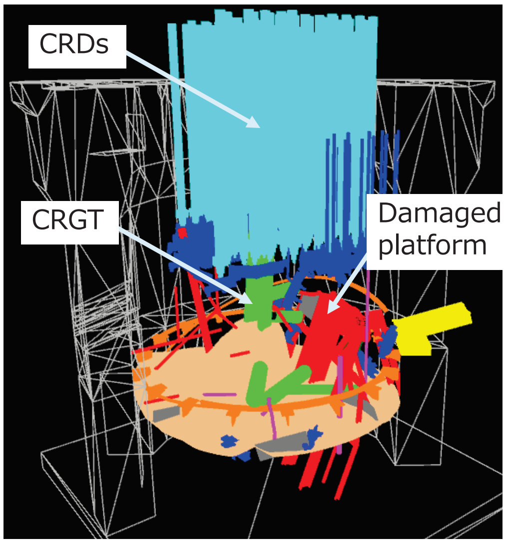
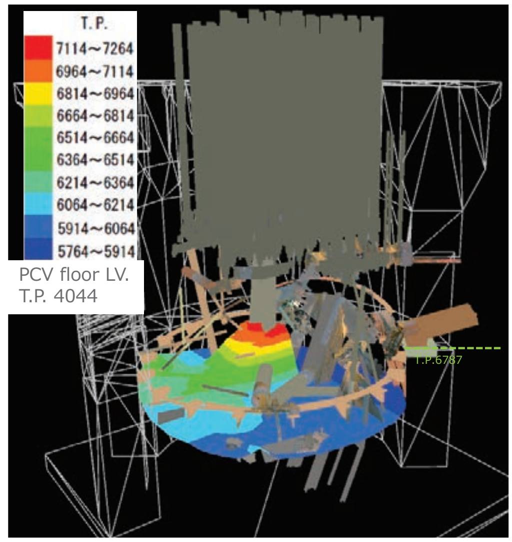


Image provided by IRID
Image processed by TEPCO

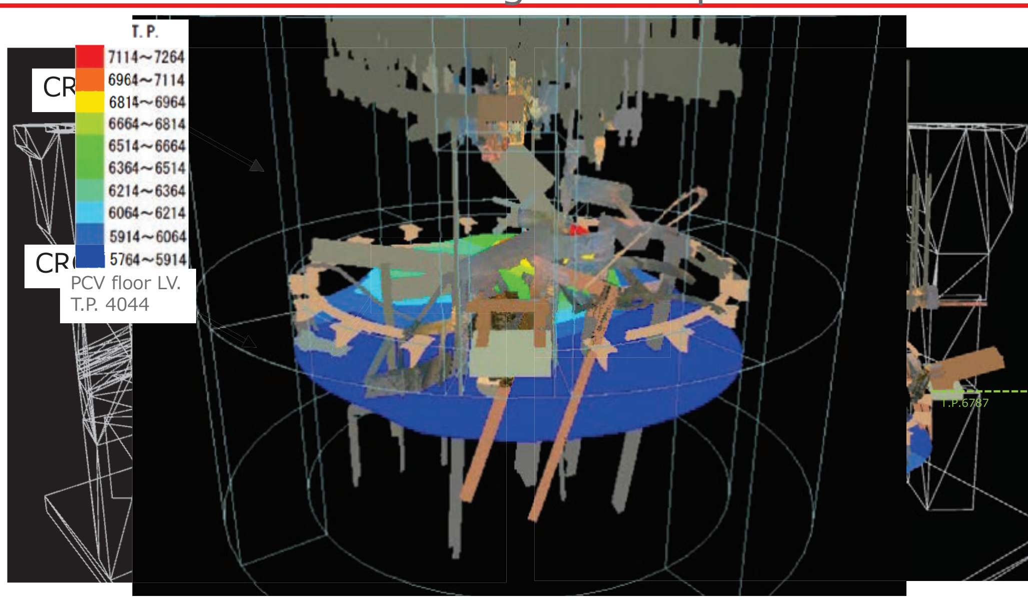
4.9 3D reconstruction image inside pedestal





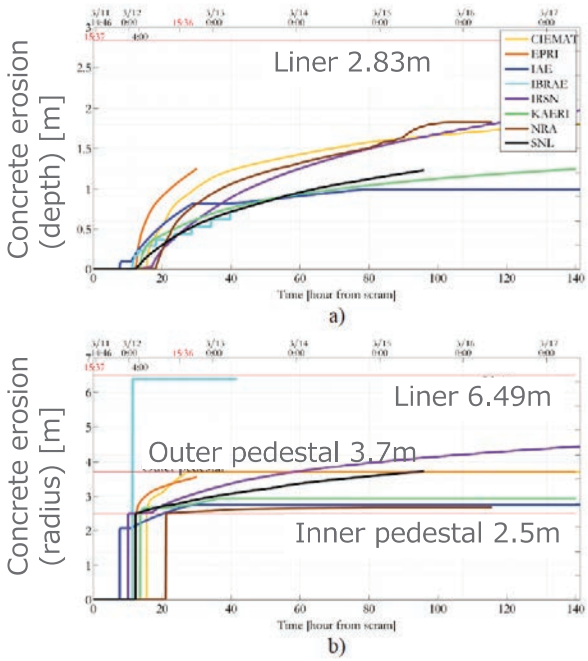
The height of debris like deposits is about 2m to 3m.
How deep is the MCCI ablation? This is also one of the difference from calculation results by SA code.

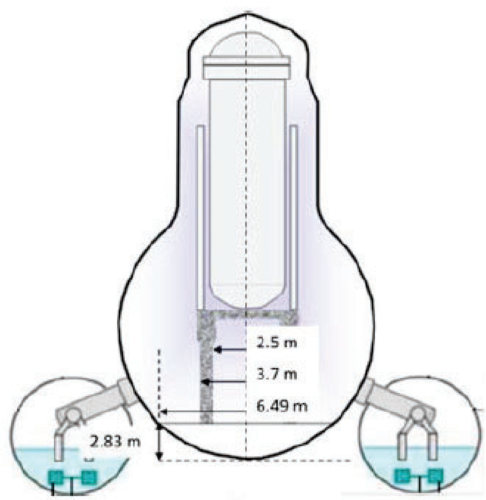
4.9 3D reconstruction image inside pedestal



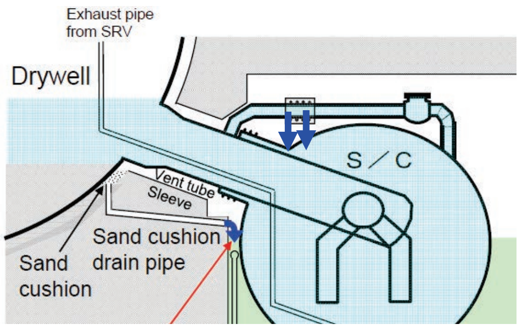
The height of debris like deposits is about 2m to 3m.
How deep is the MCCI ablation?
This is also one of the difference from calculation results by SA code.

4.10 MCCI calculation results in BSAF1 report





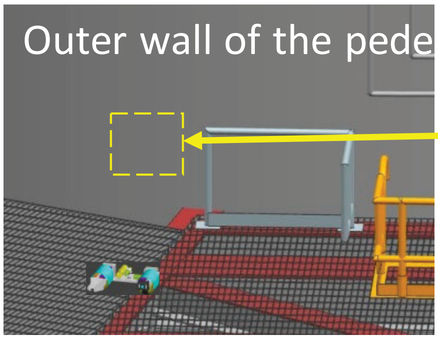
4.11 Indication of PCV failure in unit 1



Water leakage through sand cushion drain pipe means that there must be a leak hole on the PCV boundary.

But, major water leak path is other point at the vacuum breaker line.

4.12 Integrity of unit 1 PCV pedestal



Unit 1 PCV investigation results show that, at least, outer wall of the pedestal upper part could be seen.

TEPCO is now preparing to insert boat like robot to investigate the accumulated deposits on the PCV floor and lower part of the pedestal. It is expected to reveal real accident phenomena related to MCCI.

5. Conclusion

- In this presentation, state of the art knowledge and facts derived from real investigation results and measurement of Fukushima Daiichi accident are shown with some example.
- TEPCO, Japanese nuclear industries, and international experts in SA fields are learning much about SA phenomena from decommissioning activity in Fukushima Daiichi.
- You can confirm the current status of understandings of Fukushima Daiichi accident in BSAF session presentations.

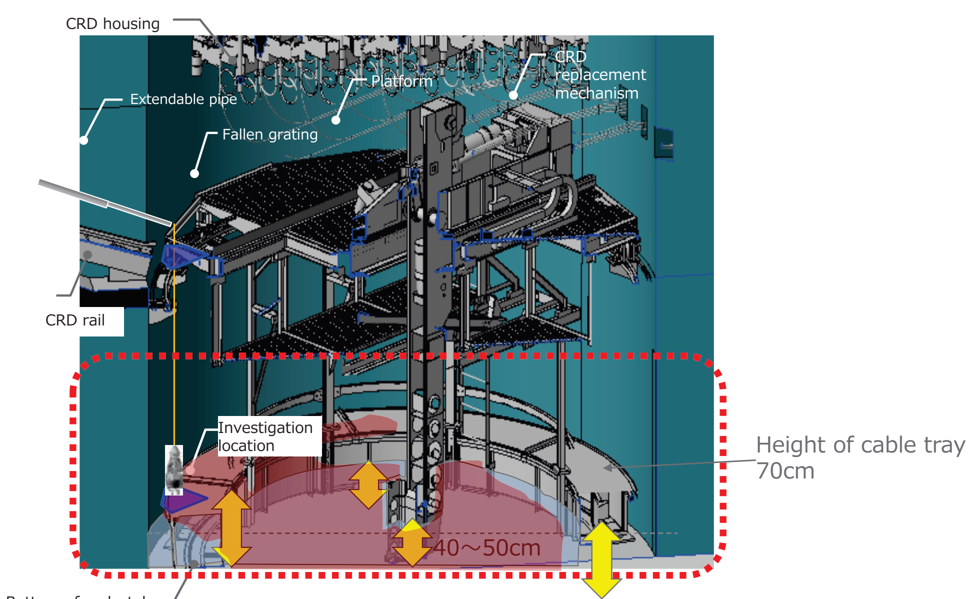
- It is better to use the information derived from 1F for SA code validation to reflect lessons learned from the accident, at least from the viewpoint of product V&V.

- TEPCO will continue investigation and decommissioning work. And Information from such activities will be provided at http://fdada.info/ .

Thank you for your kind attention.

Fine.

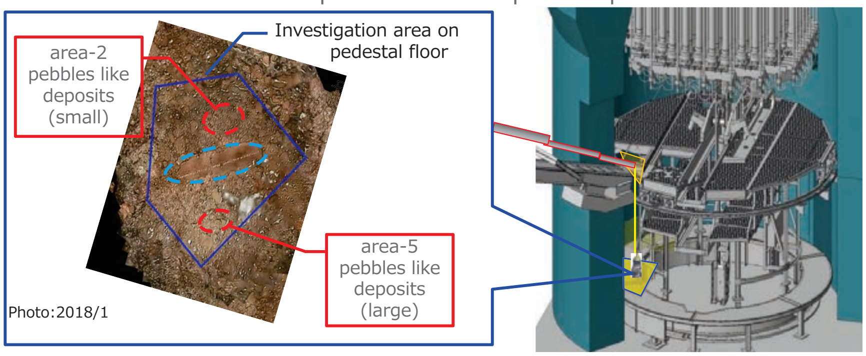
2. Investigation area in unit 2 PCV investigation(4/6)



※When viewing this area, fallen objects, such as parts of a fuel assembly (top tie plate), could be seen in the vicinity and the height of deposits in this area was higher than that in the surrounding area. Therefore, it is possible that the area directly above this location marks one of the paths by which fuel debris fell.

3． Investigation result : picking trial （1/2）

- It was confirmed that “pebbles” can be picked up.



Picking trial at area 2

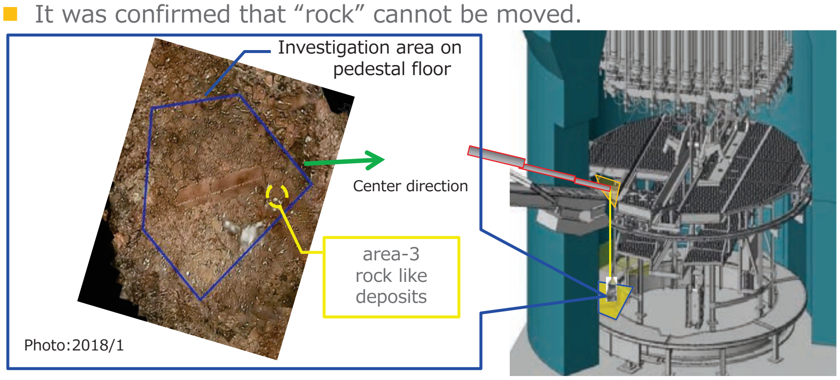
Picking trial at area 5

34

35

3. Investigation result : picking trial (2/2)

- It was confirmed that "rock" cannot be moved.



Holding the rock by finger

After pull up the finger

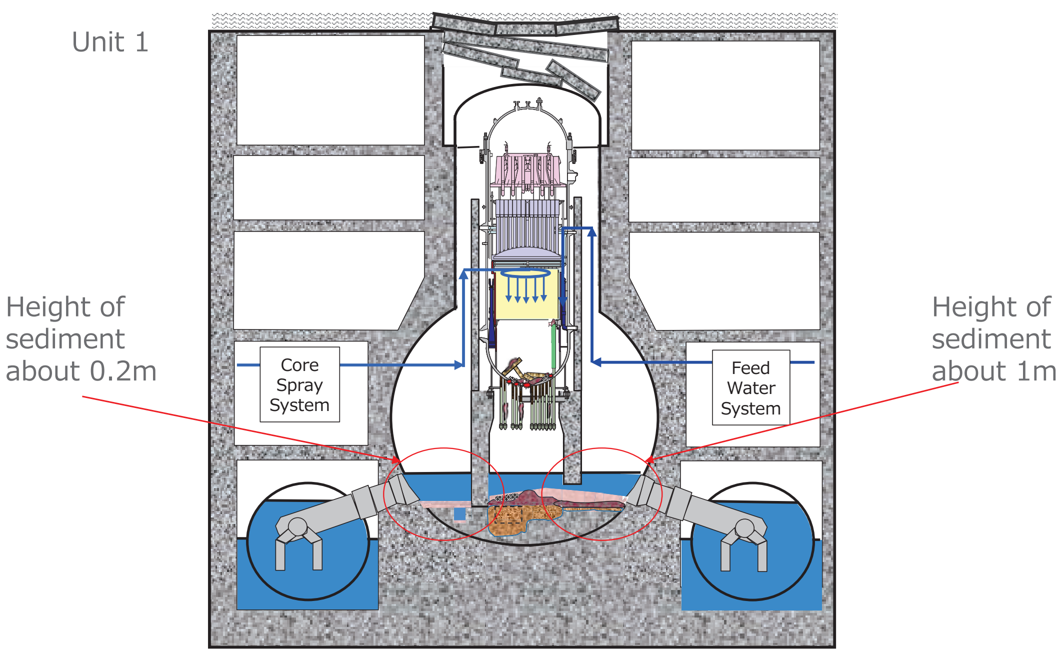
Picking trial at area 3

TEPCO

38

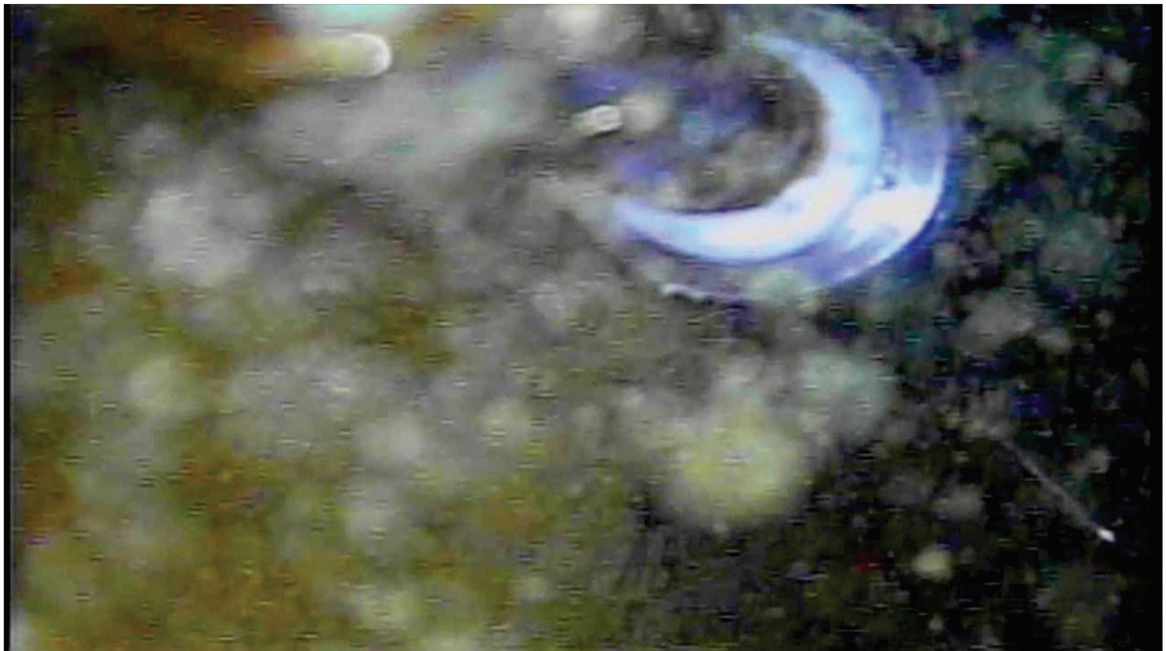
Sediments observed in unit 1 PCV floor outside of pedestal

TEPCO



39

Sediments observed during 2nd PCV entry in 2015



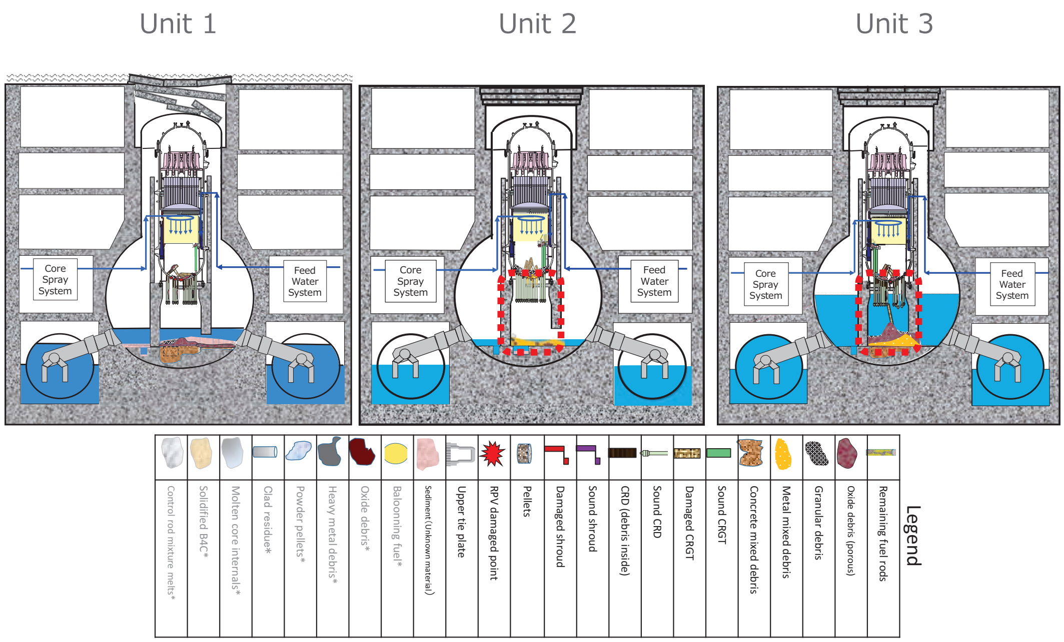
©Tokyo Electric Power Company Holdings, Inc. All Rights Reserved 無断複製・転載禁止 東京電力ホールディングス株式会社

Video image of sampling of sediments

©Tokyo Electric Power Company Holdings, Inc. All Rights Reserved 無断複製・転載禁止 東京電力ホールディングス株式会社

5. Estimation of the debris distribution for unit 1 to 3

TEPCO



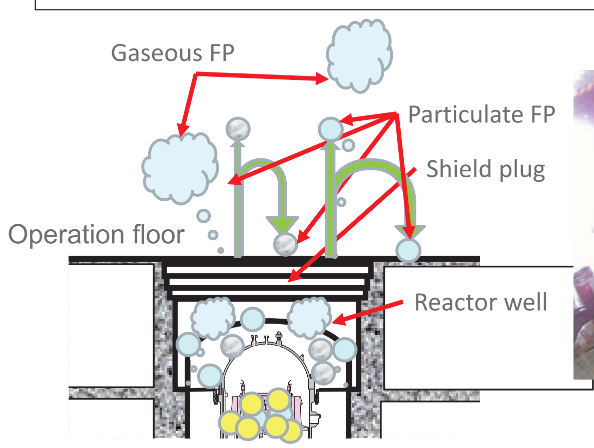
*Not used in Illustration of Estimated Unit 1-3

42

11. Sample 5: Unit-2 operation floor curing sheet

TEPCO

- ✓ The curing sheet covered Unit 2 top shield plug which was considered as the main release path of FP: top head flange ⇒ shield plug ⇒ reactor building.
- ✓ Possible existence of cesium bearing particles as found in environment.



Unit-2 operation floor

IRID JAEA NFD Power the Future

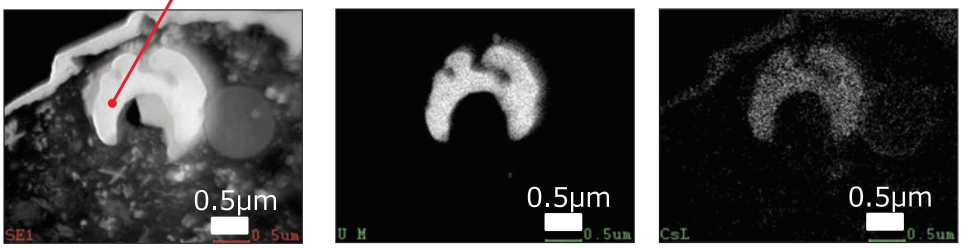
©Tokyo Electric Power Company Holdings, Inc. All Rights Reserved.

43

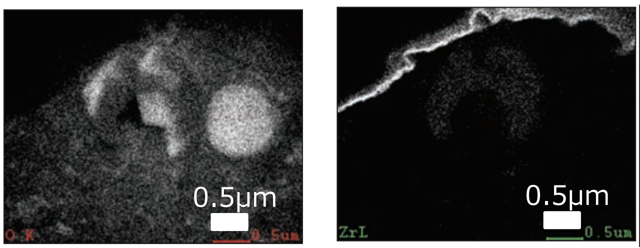
13. Sample 5: Unit-2 operation floor UO_2 particle TEM

TEPCO

Cubic-UO_2



SE image U Cs



O Zr

- ✓ UO_2 phase contains little Zr.
- ✓ UO_2 possibly came from evaporation process and condensed on the spherical SiO_2 sphere which might separate during sample preparation

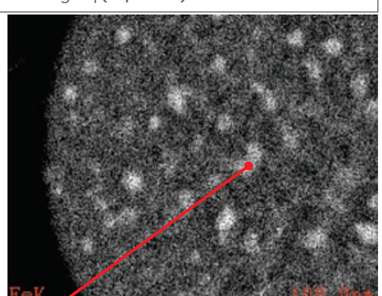
IRID JAEA NFD Power the Future

©Tokyo Electric Power Company Holdings, Inc. All Rights Reserv 無断複製・転載禁止 東京電力ホールディングス株式会社

14. Sample 5: Unit-2 operation floor Fe rich particle TEM

TEPCO

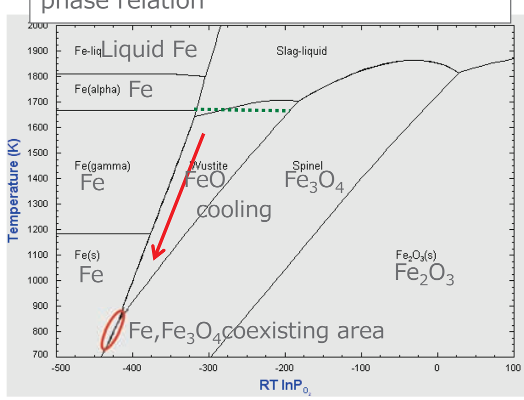
Electron diffraction revealed coexistence of metallic Fe (BCC) and Fe_3O_4(Spinel)



Metallic Fe micro particles

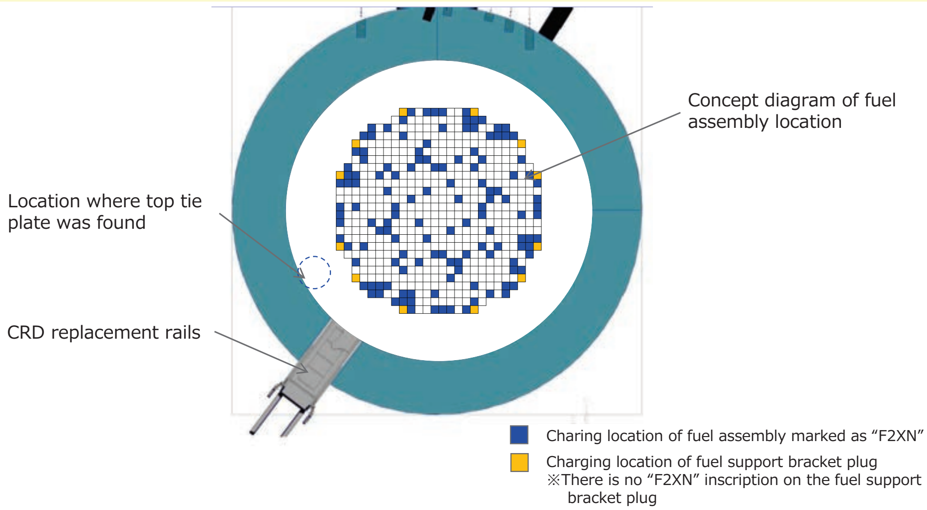
SEM/EDS(Fe)

Fe/Fe_3O_4 coexisting is possible in phase relation



- Spherical geometry suggests this is formed by evaporation-condensation process. In the particle, Fe and Fe_3O_4 coexist and contain small amount of uranium. (Left)
- It is considered that the condensation phase is FeO, and separating into Fe and Fe_3O_4 during cooling in environment of P_{H_2}/P_{H_2O} = 0.01～1 .
- The formation mechanism is probably as follows: Evaporation as $Fe(OH)_2$ → Condensation as FeO → Phase separation into Fe and Fe_3O_4

・ 132 out of the total 548 fuel assemblies that were in the reactor begin with the inscription F2XN and are followed by a serial number. The locations of these fuel assemblies are shown in the diagram below.



C.1.3. Japan Atomic Energy Agency

Current situation of OECD/NEA, Preparatory Study on Analysis of Fuel DEbriS (PreADES) project

18-19 November 2019

Akira NAKAYOSHI

Japan Atomic Energy Agency
International Research Institute for Nuclear Decommissioning

This slide includes results obtained under research program entrusted to International Research Institute for Nuclear Decommissioning, including Japan Atomic Energy Agency, by Agency for Natural Resources and Energy, Ministry of Economy, Trade and Industry (METI) of Japan.

JAEA PreADES

Background

1) OECD/NEA
(Organisation for Economic Co-operation and Development/Nuclear energy Agency)

Intergovernmental agency that facilitates co-operation among countries with advanced nuclear technology infrastructures to seek excellence in nuclear safety, technology, science, environment and law. NEA, which is under the framework of the Organisation for Economic Co-operation and Development, is headquartered in Paris, France.

2) NEA-CSNI (Committee on Safety of Nuclear Installations), **senior expert group was set-up**

SAREF (Senior Expert Group on Severe Accident Opportunities Post-Fukushima Research & Evaluation for Fukushima) was set up (Jun 2013) to address safety research knowledge gaps (near–term & long-term) related to the Fukushima Daiichi nuclear accident.

3) From SAREF deliberations *near-term* projects were proposed (2016-17):
PreADES (Preparatory Study on Analysis of Fuel debris)

"Near-term projects" which can start relatively quickly in preparatory phase. An example is to collect and analyse basic information and track information on damaged state and maintain information channels between the CSNI and relevant Japanese organizations, and monitor feasibility of extraction, transportation, examination, etc. of samples to be taken.

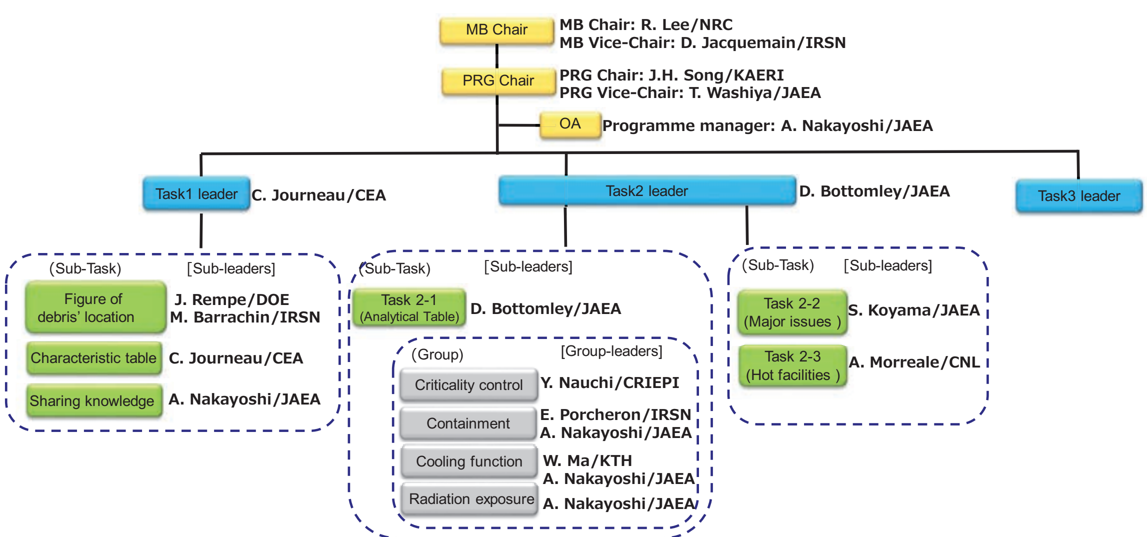
Structure of PreADES

- Management board (MB):
  - MB Chair: Richard Lee, NRC
  - MB Vice-Chair: Didier Jacquemain, IRSN
- Program Review Group (PRG)
  - PRG Chair: Jin Ho Song, KAERI
  - PRG Vice-Chair: Tadahiro Washiya, JAEA
- Programme Manager/Operating Agent: Akira Nakayoshi, JAEA
- 7 countries and 15 organizations join to PreADES

  Canada(CNL), France(CEA IRSN EDF), Japan(JAEA NRA CRIEPI), Korea(KAERI KINS), Sweden(SSM), Switzerland(PSI), United States(NRC DOE EPRI) , EC/JRC

2

Structure of PreADES



3

Outline of each Task in PreADES

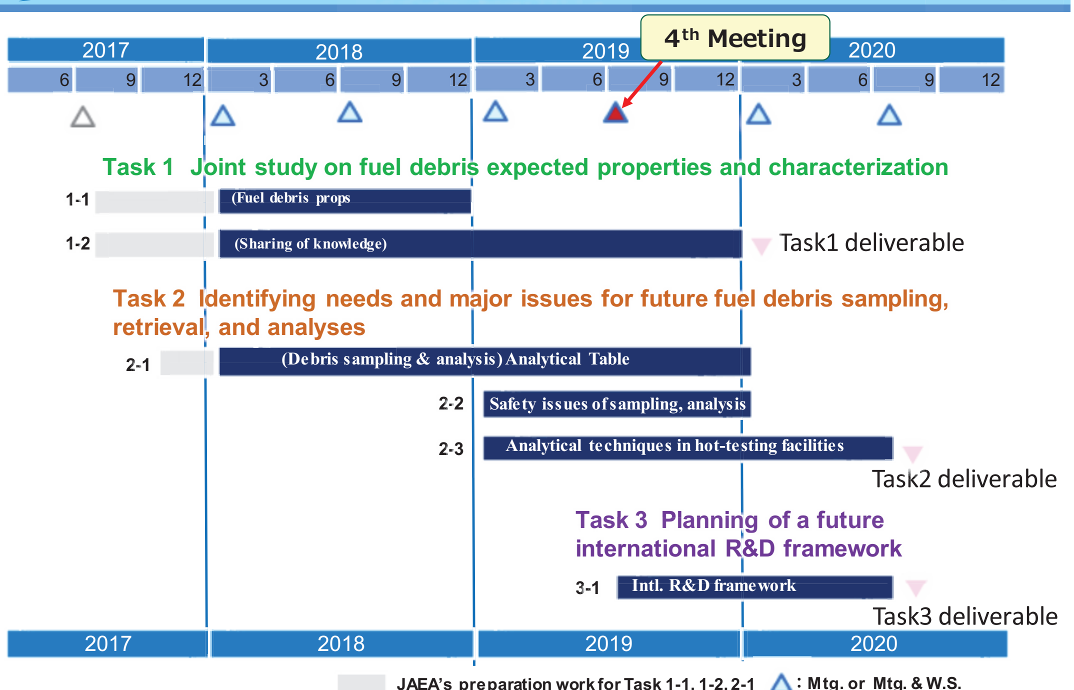
- **Task 1:** Joint study on fuel debris characterization
  - Discuss fuel debris location in 1F units 1-3
  - Discuss debris properties at various location

    Properties : Microscopic and Macroscopic view points

- **Task 2:** Needs identification and issues discussion for future fuel debris analysis
  - Discuss needs for analysis, from decommissioning point of view; Criticality, Containment Cooling function, Radiation exposure
  - Discuss major issues for sampling; Analysis methods, Facility demands, Transportation, Accounting, etc.
  - Discuss hot testing facilities; Capable of fuel debris analysis
- **Task 3:** Planning future international R&D framework

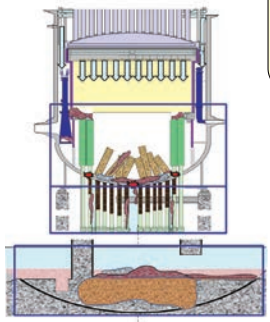
4

Schedule



5

Structure of Task 1



2) Characteristic table for micro

-Characterisation of debris from microscopic point of view

(Mainly, properties of single-phase material)

3) Characteristic table for macro

-Characterisation of debris from macroscopic point of view

(Mainly, features of multi-phases material/mixture)

1) Figure of debris' location

- Image of debris' location in reactor

* Provided by TEPCO

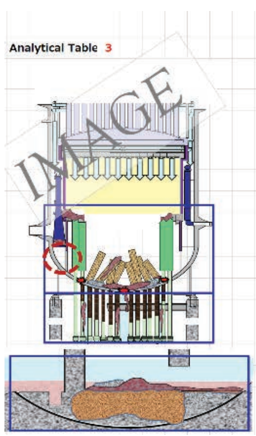
These will be summarised and shared as basic information in order to discuss Analytical table (Task 2) & Task 3

6

Task 1: Microscopic Properties

- **Characteristic table- (Micro table)**

Density (g/cm^3)	Vickers Hardness (GPa)	Elastic Modulus (GPa)	Fracture Toughness (MPa m$^{1/2}$)	Thermal Conductivity (W/mK)	Specific Heat (J/g K)	Melting Point (°C)	Behaviour at high temperature	Stability



* Provided by TEPCO

Characteristic Table micro

Categories / Material / Phases	Density (g/cm^3)	Vickers Hardness (GPa)	Elastic Modulus (GPa)	Fracture Toughness (MPa m$^{1/2}$)	Thermal Conductivity (W/mK)	Specific Heat (J/g K)	Melting Poin t (°C)	Behaviour at high	Stability
Oxide									
UO_2	11.0	5.9	193	0 - 1.5	10	0.28	2850		UO2 should be oxidised under oxygen existing, and it causes volume change, cracks.
ZrO_2-C	6.1	15	200	0 - 10	1 - 3	0.61	2700		
$(U,Zr)O_2$-C	6.1 - 11.0	5.9 - 11	200	0 - 3.0	1 - 3	0.28 - 0.61	2500 - 2850		
$(U,Zr)SiO_4$	4.6 - 8.3	11	166	0 - 1.2	6.1	0.45 - 0.82	2500		
Metallic									
U									
Zry-2									There are explosion cases of Zr fine
α-Zr(O)									
SUS/Fe									
Fe2(Zr,U)									
Others									
B4C									
ZrB2									

7

Task 1: Macroscopic Properties

- **Characteristic table- (Macro table)**

Composition*3	U, Pu Concentration*2*3 (wt%)	SUS Concentration*2*3 (wt%)	B_4C Concentration*2*3 (wt%)	……..	Porosity *2*3 (vol%)	Bulk density (g/cm^3)	Moisture *2*3 (wt%)	G Value*2*3 (Molecule/100 eV)	Compressive strength*2*3 (MPa)

□ Location

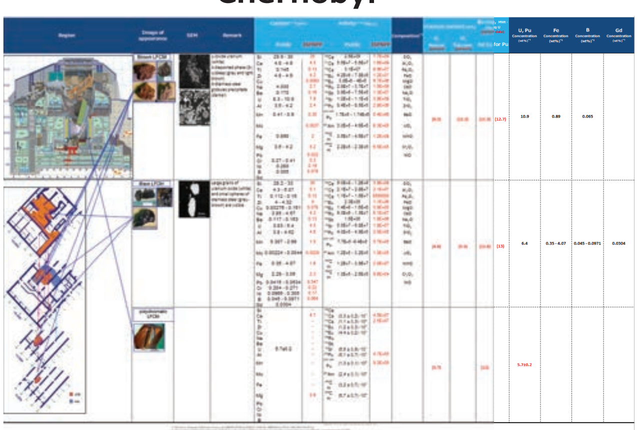
Location	Image of appearance (Shape)*1	Remark	Composition*1	U, Pu Concentration (wt%)	SUS Concentration (wt%)	B_4C Concentration (wt%)	Gd Concentration (wt%)	Cl Concentration (wt%)	U enrichment (wt%)	Size	Porosity (vol%)	Bulk density (g/cm^3)	Moisture (wt%)	G Value (Molecule/100 eV)	Compressive strength (MPa)
No.6 	● Crust (MCCI Products) 	Material of cooled and solidified relatively early on the upper surface of the molten pool at the MCCI event	Main compound: $(U,Zr)O_2$-C, $(Zr,U)O_2$-T, Al-Ca-Si-O	40	30	2	0.002	0.002	1.87	0.1-1(m)	30		5	0.5	22
	● Molten corium pool / Rock (MCCI Products) 	Molten corium slowly cooled and clumped at MCCI	Main compound: $(U,Zr)O_2$-C, $(Zr,U)O_2$-T, Al-Ca-Si-O	40	30	2	0.002	0.002	1.87	above dozens(cm)	30		5	0.5	22
	● Metal region (MCCI Products) 	Metal part deposited at each part of MCCI	Main compound: Fe	40	30	2	0.002	0.001	1.87	several(cm)	30		4	0.5	22
	● Powder / Pebble (No Photos)		Main compound: (U,Zr)O2-C, (Zr,U)O2-T	40	30	2	0.002	0.003	1.87	several(μm) to several(cm)	88		14	0.5	230

Task 1: Macroscopic Properties

- **Past severe accident experience**

-In collaboration with NRC/DOE/EPRI & ISP-NPP, debris information of TMI-2 and Chernobyl where data of characteristics exists in past severe accident was compiled in Characteristic table

Chernobyl



TMI-2

Task 1 Summary

1) Figure of debris' location

- Debris endstate figure is useful communication tool
- Integrating information from post-accident examinations, plant-specific analyses, plant instrumentation, and severe accident knowledge

2) Characteristic table

- Debris information of TMI-2 and Chernobyl where data of characteristics exists in past severe accident was compiled
- Debris properties in each location at 1F will be estimated

Decommissioning especially, removal of debris as well as severe accident research

Task 2
Analysis needs

Task 3
Intl. R&D framework

10

Structure of Task 2

Task 2-1 : Needs for fuel debris analysis

Discussion points;

- Criticality Control
- Containment
- Cooling function
- Radiation exposure
- SA research

Task 2-2 : Study on major issues for future debris sample analysis, transportation, storage and disposal

Task 2-3: Experimental/analytical techniques and required function in Hot-testing facilities

11

Structure of Task 2

Task 1
Debris characterization

Other viewpoints

Task 2-1
Needs for Debris analysis

Methodology for safety
- Transport, Handling, Storage etc.

Design/License
- Sampling, Transport, Analysis etc.

Handling debris in facilities for analysis
- Techniques & Demands
- Transport, Analysis, Storage etc.

Task 2-2
Methodology for safety major issues of handling debris for analysis

Task 2-3
Techniques & demands

Task 3
Discussion about contents of international cooperation and R&D.

12

Task 2-1: Analytical table

Analytical table

Adopting up key issue related to discussion points, required information (Debris characteristics) are presented.

Organizing analysis items/scoring priorities for required debris characteristics

Analytical Table No. (Characteristic table position)	Access time (including priority for analysis)	Location of Sample: Region	Location of Sample: Zone	Issues to be solved with high priority in Japan	What/ How Obtained (Features of Sample): What (Target Object) (Form)	What/ How Obtained (Features of Sample): How	Why (Objective/ Motivation)	Expected Benefit/Use	Methods/Tools Needed to Collect Information or Data
1	Sampling	Drywell (inside Pedestal)							
			D/W floor	Sample (Described from Key issue)	Particle, Slurry (in suspension in water samples)	Debris charact. fuel debris retrieval, assess Analysis items	Purpose (Described from Key issue) Assess how sediments will affect fuel debris retrieval operations, decide on their removal or other option and treatment	Outcomes (Described from Key issue) Optimisation of fuel debris retrieval operations, assess challenges and risks related to the presence of sediments	Robotic means with camera and water jet able to investigate sediment layer in depth

Task 2-1: Analytical table

E.g., Criticality control

Location of sample			Significance of property for CC in each phase. High: 5 to Low: 1				
			Decommissioning				SA
Region	Zone	Why (Objective / Motivation) Material properties etc.	For Waiting period for retrieval	For Retrieval process	For Transport, storage,	For Conditioning, disposal	For SA research
RPV+ PCV	Initial location, lower vessel head, below-vessel, floor of PCV inside /outside pedestal	Map of fuel location	5	4	1	1	4
PCV	below-lower vessel head, floor of PCV inside / outside pedestal	Dilution of fuel in various materials (such as lower structural materials below pressure vessels) & evaporation of Boron oxide	3	3	4	4	3

14

Task 2-1 Summary

1) Criticality control
- All analysis items are high priority in all steps
- Mass ratios of U + Pu, 157Gd / (U+Pu), Ag, In, Cd (AIC) in fuel are highest score among analysis items

2) Radiation exposure
- Scores before "Transport storage" are higher
- Analysis items "Quick Look Video", "Airborne Particle, Aerosol in cutting" are important

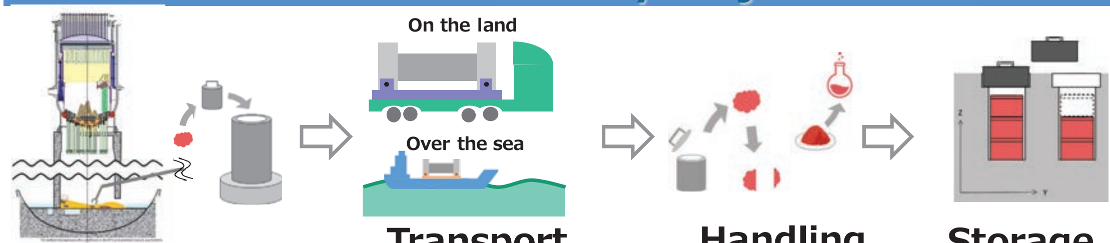
3) Containment function
- Retrieval process has main issues
- Dust in cutting debris is concerned mainly

4) Cooling function
- Heating value by calorimeter & Particle size distribution by SEM are important analysis items in Retrieval & Transport storage

Analytical table not only identifies knowledge gaps and priorities, but will also lead to understanding debris conditions and stability as well as accident progression scenario.

Task 3 Intl. R&D framework

Task 2-2: Methodology for safety major issues



(in Japan)	**Transport** (Domestic/Foreign)	**Handling** (in Hot Lab)	**Storage**
・**Radiation shielding calculation** ・**Criticality safety calculation** ・Domestic legal issues ・Safeguards, etc.	・Radiation shielding calculation ・Criticality safety calculation ・**Legal issues** ・Safeguards, etc. ・**Secure transportation** (Port availability, consider land transportation means)	・**Hot laboratory at Okuma , Tokai and Oarai area** ・**Hot laboratory at each organization** ・Radiation shielding calculation ・Criticality safety calculation ・Safeguards, etc.	・**Hot laboratory at Okuma, Tokai and Oarai area** ・**Hot laboratory at each organization** ・Radiation shielding calculation ・Criticality safety calculation, etc.

Major issues & methodologies at each step;
"Transport", "Handling" and "Storage", are considered

16

Task 2-2: Major issues and their methodologies table

Contents	Issues	Oganization, Facility	Knowledge & methodology (from TMI-2, Chernobyl and the latest findings)	Suggestion / Conclusion
Transport	- Hydrogen management			
	- Criticality safety			
	- Heat removal			
	- Containment			
	- Radiation shielding			
Storage	- Criticality safety		<Criticality Safety> Criticality safety in strage condision is evaluated as folloing condition. This storage condition is used for the hot cell of JAEA facility as an example. ①Calculation code ・SCALE6.1 (KENO-V.a) (Slide No.2) →Generation : 1000 →History : 1000 →Cross sectional libraly : v7-238 … (Pu238, Pu239, Pu240, Pu241, Pu242) = (1, 60, 24, 10, 5)[wt%] (Slide No.3) ・U fuel weight ratio to MOX fuel(%) : 0 to 100 (Pure MOX fuel is most conservative as the fissile material on this evaluation. (Slide No.5))	<Criticality Safety> ・The same calculation model by using spent fuel can be applied for storage of fuel debris. …hat …are …Pu- Content : highest value

Example

17

Task 2-3: Information of Hot-testing facilities

- **Summarizing of analytical techniques in Hot-testing facilities**

-Currently includes capabilities from the following:

- Canadian Nuclear Laboratories (CNL) – Canada
- US Department of Energy (DOE) – USA:
  - Pacific Northwest Nuclear Laboratory (PNNL); Oak Ridge National Laboratory (ORNL); Savannah River National Laboratory (SRNL); Idaho National Laboratory (INL); Los Alamos National Laboratory (LANL); Argonne National Laboratory (ANL).
- Paul Scherrer Institute (PSI) – Switzerland
- Commissariat à l'Energie Atomique et aux énergies alternatives (CEA) – France
- Korea Atomic Energy Research Institute (KAERI) – Korea
- Studsvik - Sweden
- JRC Karlsruhe (JRC) – European Union
- Japan Atomic Energy Agency (JAEA) – Japan

- **Ready to integrate capabilities from other labs!**

18

Task 2-3: Hot testing analysis capabilities table

- **The capabilities are divided into 8 sections:**

1. Hot Cell Facilities General Description
2. Material Handling
3. Sample Preparation
4. Non-destructive Testing
5. Destructive Testing (Mechanical Analysis)
6. Chemical Analysis
7. Microscopy
8. Materials and Surface Science Analysis

- **Discuss additions/improvements to sections**

19

Task 3: Planning of future international R&D framework

- **Based on developments of two tasks, preliminary proposals for future international R&D framework of 1F sample analysis are to be made**

  - **OA continue to discuss with stakeholders for 1F decommissioning about fundamental image of international research cooperation with using 1F fuel debris.**
  - **Proposed future international R&D frameworks will be discussed at 5^{th} meeting on February, 2020**

20

Cooperation with relevant NEA projects (1)

- **PreADES/TCOFF, Property tables for re-criticality evaluation**

Example of Property Tables

- Unit-3
- Pedestal
- Retrieval

Fact*
Expected

Fuel Debris Type		Characteristics	Evaluation Item for Criticality Control					
			Fissile	^{157}Gd	B	Fe	MA	FP
Fuel Debris Type	Sediments	A few µm size particle containing Uranium	Negligible, UO_2 or fcc-$(U,Zr)O_2$, fcc-$(U,Zr)O_2$+α-Zr(O)	Negligible	Negligible	Major component of oxide	Negligible, accompanied with U & Pu	Small amount of Cs, Fe, Si and Mg included
	Oxidic melt	$(U,Zr)O_2$ containing some elements	Relatively homogeneous	Relatively uniform distribution at less than 100µm in case melt pool formation inside RPV	Negligible	Potentially included up to solubility of Fe in oxidic melt	Homogeneous with U & Pu	Coexisting with $(U,Zr)O_2$ preferentialy excluding Rh,Tc and Mo in case melt pool formation inside RPV
	Metallic melt	Spread out whole in the pedestal, Less MCCI	Very low amount	Negligible	ZrB_2 or $(Fe,Cr,Ni)_2B$ dispersed in melt, partially oxidized	Mixture of SS and Zry containing less amount of actinides	Negligible	Possibly soluble for Rh, Tc and Mo in case melt pool formation inside RPV
	Partially melt or intact materials	Large amount of structural materials	Heterogeneous	Heterogeneous	Negligible	Major component (ex. platform	Heterogeneous	Heterogeneous

- Key nuclides for criticality are evaluated by PreADES
- TCOFF intends to input physical, material, and mechanical properties into PreADES's tables

Cooperation with relevant NEA projects (2)

- **Joint task force for formation process & characteristics of debris based on 1F analysis results**

  - Collaborative discussion on 1F-samples analysis data, in terms of formation mechanism of U-bearing particles & its influence to debris characterisation.
  - PreADES started discussion on establishment of Joint task force.
  - OA will propose Joint task force structure to PreADES members together with ARC-F & TCOFF by next PreADES meeting.

Cooperation with relevant NEA projects (3)

- **Collaboration with OECD-NEA project: ARC-F**

  - It has been agreed that PreADES and ARC-F exchange projects results & information each other.
  - Information presentations for PreADES and ARC-F to exchange technical information between both projects in semi-annual meetings so far.

Summary

- Based on agreed schedule of PreADES, 4th meeting took place on July in Tokyo Japan. Task 1 is almost completed and Task 2 will be completed soon.
- Collaboration among PreADES, ARC-F, and TCOFF is important and starts to be implemented.
- Next meeting : February 10-14, 2020 in Paris France; the meeting schedule has been coordinated with ARC-F.

24

C.2. Topic Area Presentations

C.2.1. Topic Area 1 - Component/System Performance

REACTOR SAFETY TECHNOLOGY
Experts Panel Forensics Meeting

Topic 1 - Component/System Examinations

J. Gabor, Jensen Hughes
K. Robb, ORNL

November 18-19, 2019
ANL Offices
Washington, DC

Topics

- Key questions
- Current status – November 2019
- Major observations from 2019 activities
  1. First deposit investigations at U2 pedestal region
  2. U2 below vessel structures mostly intact
  3. Investigation of U1 shield plugs
  4. Debris removal from spent fuel pool areas
  5. Plan to begin injection flow reduction
  6. NURETH-18 presentation by Mizokami

2

Key Questions

- What visual damage has been observed in component and structures with RPV, PCV and RB?
- What plant data supports damage assessment?
- What insights are gained from damage assessment (e.g. peak temperatures, pressures and radiation levels)
- Can insights be used to enhance reactor safety and SA guidance.
- Are analysis improvements needed?

3

U.S. Efforts in Support of Examinations at Fukushima Daiichi – 2019 Evaluations

ANL-19/08

August 27, 2019

Table 3-1. Area 1 information needs from the reactor building

Item	What/How Obtained[a]	Use[b]	Data Available[c]
RB-1	Photos/videos of condition of RCIC valve and pump before drain down and after disassembly (1F2 and 1F3)	AE, AM	NA
RB-2	Photos/videos of HPCI System after disassembly (1F1, 1F2, and 1F3)	AM	NA
RB-3a	Photos/videos of damaged walls and structures (1F1)	AE, AM, DD	A
RB-3b	Photos/videos of damaged walls and structures (1F3)	AE, AM, DD	A
RB-3c	Photos/videos of damaged walls and structures (1F4	AE, AM, DD	A
RB-4	Photos/videos of damaged walls and components and radionuclide surveys (1F2)	AE, AM, DD	A
RB-5	Radionuclide surveys (1F1, 1F2, and 1F3)	AE, AM, DD	A
RB-6	Radionuclide surveys and sampling of ventilation ducts (1F4)	AE, AM, DD	A
RB-7	Isotopic evaluations of obtained concrete samples (1F2)	AE, AM, DD	A
RB-8	Photos/ videos and inspection of seismic susceptible or radiation degraded components and structures (e.g., bellows, penetrations, welds, structures, supports, etc. in 1F1, 1F2, 1F3, and 1F4)	AE, AM, DD	A
RB-9	DW concrete shield radionuclide surveys (1F1, 1F2, and 1F3 - after debris removed)	AE, AM, DD	A
	Photos/videos and dose surveys around mechanical seals and hatches and electrical penetration seals (as a means to classify whether joints were in compression or tension)	AE, AM, DD	A
RB-10	Photos/videos of 1F1 (vacuum breaker), 1F1, 1F2, and 1F3 PCV leakage points (bellows and other penetrations)	AE, AM, DD	A
RB-11	Photos/videos and available information on 1F1, 1F2, and 1F3 containment hardpipe venting pathway, standby gas treatment system and associated reactor building ventilation system	AE, AM, DD	A
RB-13	Photos/videos of 1F1, 1F2, and 1F3 recirculation lines and pumps	AM, DD	A
RB-14	Deposits or particles sampled inside reactor building (1F1, 1F2, 1F3); e.g., white deposits from HPCI room using FE-SEM, XRD, etc.	AE, AM, DD	NA
RB-15	Examinations of 1F1 RCW surge tank; water level and additional dose measurements.	AE, AM, DD	A

a. See list of acronyms.
b. Use: AE – Accident evaluation (code modeling updates), AM- Accident management and prevention, DD – Decontamination and Decommissioning, and PM – Plant maintenance (see Appendix C for more information).
c. A: some information available [Green]; NA: no information available [Orange].

4

U.S. Efforts in Support of Examinations at Fukushima Daiichi – 2019 Evaluations

ANL-19/08

August 27, 2019

Table 3-2. Area 1 information needs from the PCV

Item	What/How Obtained[a]	Use[b]	Data Available[c]
PC-1	Photos/ videos of drywell head, head seals, and sealing surfaces (1F1, 1F2, and 1F3). Procedures used to tension and torque the bolts used to close the drywell head bolts.[d]	AE, AM, DD	NA
PC-2	Photos/videos and radionuclide surveys/ sampling of IC (1F1).	AE, AM, DD	NA
PC-3[e]	a) Photos/videos of relocated debris and crust, debris and crust extraction, hot cell exams, and possible subsequent testing (1F1 - 1F3)	AE, AM, DD	A
	b) PCV liner examinations (photos/videos and metallurgical exams); (1F1-1F3)	AE, AM, DD	A
	c) Photos/video, RN surveys, and sampling of pedestal wall and floor (1F1-1F3)	AE, AM, DD	A
	d) Concrete erosion profile; photos/videos and sample removal and examination (1F1-1F3)	AE, AM, DD	NA
	e). Photos/videos of RPV lower head and of structures and penetrations beneath the vessel to determine damage and corium hang-up (1F1-1F3)	AE, AM, DD	A
PC-4	Examinations and operability assessments of 1F1, 1F2, and 1F3 ex-vessel sensors and sensor support structure	AE, AM	A
PC-5	Photos/videos of 1F1, 1F2, and 1F3 main steam lines and ADS lines to end of SRV tailpipes, including instrument lines	AE, AM, DD	NA
PC-6	Visual inspections of 1F1, 1F2, and 1F3 SRVs including standpipes (interior valve mechanisms)	AE, AM, DD	NA
PC-7	Ex-vessel inspections and operability assessments of 1F1, 1F2, and 1F3 in-vessel sensors and sensor support structures[f]	AE, AM, DD	A
PC-8	Examinations and operability assessments of 1F1, 1F2, and 1F3 ex-vessel sensors and sensor support structure	AE, AM, DD	A
PC-9	Photos/videos of 1F1, 1F2, and 1F3 PC (SC and DW) coatings	PM	A
PC-10	1F1, 1F2, and 1F3 RN surveys in PCV	AE, AM, DD	A
PC-11	Photos/videos of 1F1, 1F2, and 1F3 primary system recirculation pump seal and any potential discharge to containment	AE, AM, DD	NA
PC-12	Photos/videos of 1F1, 1F2, and 1F3 TIP tubes and SRV/IRM tubes outside the RPV	AE, AM, DD, PM	NA
PC-13	Photos/videos of 1F1, 1F2, and 1F3 insulation around piping and the RPV.	AM	NA
PC-14	Samples of conduit cabling, and paint from 1F1, 1F2, and 1F3 for RN surveys.	AE, AM	A
PC-15	Samples of water from 1F1, 1F2, and 1F3 for RN surveys.	AE, AM, DD	A
PC-16	Photos/videos of melted, galvanized, or oxidized 1F1, 1F2, and 1F3 structures.	AE, AM	A

a. See list of acronyms.
b. Use: AE – Accident evaluation (code modeling updates), AM- Accident management and prevention, DD – Decontamination and Decommissioning, and PM – Plant maintenance (see Appendix C for more information).
c. A: some information available [Green]; NA: no information available [Orange].
d. Available information is limited to the shield plug.
e. As described in Section 5.2.2, in-vessel components have been discovered ex-vessel for two of the affected units (i.e., fuel assembly top tie plate for 1F2 and Control Rod (CR) guide tube and CRD index tube for 1F3) indicating that the vessels have failed. In addition, sample evaluations from 1F1 have detected the presence of uranium.
f. Inspections and evaluations [e.g., continuity checks, calibration evaluations, etc] of sensors [differential pressure (dP) cells, water level gauges, TIPs, TCs, CAMs, etc.] and sensor support structures, cables, removed TIPs, etc.; Requires knowledge of sensor operating envelop.

5

U.S. Efforts in Support of Examinations at Fukushima Daiichi – 2019 Evaluations

ANL-19/08

August 27, 2019

Table 3-3. Area 1 information needs from the RPV

Item	What/How Obtained[a]	Use[b]	Data Available[c]
RPV-1	1F1, 1F2, and 1F3 dryer integrity and location evaluations (photos/videos with displacement measurements, sample removal and exams for fission product deposition, peak temperature evaluations). If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.	AE, AM, DD	NA
	Photos/videos, probe inspections, and sample exams of 1F1, 1F2, and 1F3 MSLs; interior examinations of MSLs at external locations. If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.	AE, AM, DD	NA
	Photos/videos and metallurgical examinations of upper internals and upper channel guides. If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.	AE, AM, DD	NA
RPV-2	Photos/videos of 1F1, 1F2, and 1F3 core spray slip fit nozzle connection, sparger and nozzles. If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.	AE, AM, DD	NA
	Photos/videos of 1F1, 1F2, and 1F3 feedwater sparger nozzle and injection points. If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.	AE, AM, DD, PM	NA
RPV-3	1F1, 1F2, and 1F3 steam separators' integrity and location (photos/videos with displacement measurements, sample removal and exams for FP deposition, peak temperature evaluations). . If significant distortion observed, then metallurgical exams of samples would be of interest during removal for D&D.	AE, AM, DD	NA
RPV-4	1F1, 1F2, and 1F3 shroud inspection (between shroud and RPV wall); Photos/videos of interest. If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.	AE, AM, DD	NA
	1F1, 1F2, and 1F3 shroud head integrity and location (photos/videos). If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.	AE, AM, DD	NA
	Photos/videos of 1F1, 1F2, and 1F3 shroud inspection (from core region). If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.	AE, AM, DD	NA
	Photos/videos of 1F1, 1F2, and 1F3 core plate and associated structures.	AE, AM, DD	NA
RPV-5	Remote mapping of 1F1, 1F2, and 1F3 core through shroud wall from annular gap region (muon tomography and other methods, if needed).	AE, AM, DD	A
	Mapping of end state of core and structural material (visual, sampling, hot cell exams, etc.).	AE, AM, DD	NA

a. See list of acronyms.
b. Use: AE – Accident evaluation (code modeling updates), AM- Accident management and prevention, DD – Decontamination and Decommissioning, and PM – Plant maintenance (see Appendix C for more information).
c. A: some information available [Green]; NA: no information available [Orange].

6

U.S. Efforts in Support of Examinations at Fukushima Daiichi – 2019 Evaluations

ANL-19/08

August 27, 2019

Table 3-4. Results from component and system examinations[a]

Area	1F1	1F2	1F3
X-100B PCV penetration[b]	Possible melted shielding material [12] No damage observed on outside [131]	TBD	TBD
X-51 PCV penetration[c]	TBD	No damage observed; pressurized water could not penetrate blockage in standby liquid cooling system line [132, 133]	TBD
X-53 HPCI steam supply penetration (1F2/1F3)[d]	High dose rate measured [17]	No damage observed [134]	No damage observed [135]
X-6 PCV penetration (CRD hatch)	TBD	Melted material [136, 137]	No damage observed from inside [138]
Equipment hatch	TBD	TBD	Water puddle [139, 140] unknown source
Personnel hatch and nearby penetrations	No damage observed [141]	TBD	TBD
HPCI pipe penetration[e]	No damage observed, but high dose rates measured; traces of flow and white sediment observed [17, 141,142]		
TIP room	No leakage observed from PCV through TIP guide penetrations. Relatively high dose rates measured near other primary system instrumentation penetrations (X-31, X-32, X-33) [17,143]	Dose surveys do not indicate leakage from PCV through TIP guides. High dose levels in samples of materials from TIP indexer [144]	
Wetwell (WW) vacuum breaker line	Leakage on expansion joint of one line (X-5E) [145]	TBD	TBD
DW/WW vent bellows	Water leakage attributed to vacuum line above [145]	No leakage observed [146]	
DW sand cushion drain pipe	Leakage [147]	No leakage observed [146]	TBD
SC water level	Almost full [53]	Middle [53]	Full [53]
DW Water Level	~2 m[53]	~0.2 m[53]	~6 m[53]

7

U.S. Efforts in Support of Examinations at Fukushima Daiichi – 2019 Evaluations

ANL-19/08

August 27, 2019

Table 3-4. Results from component and system examinations[a]

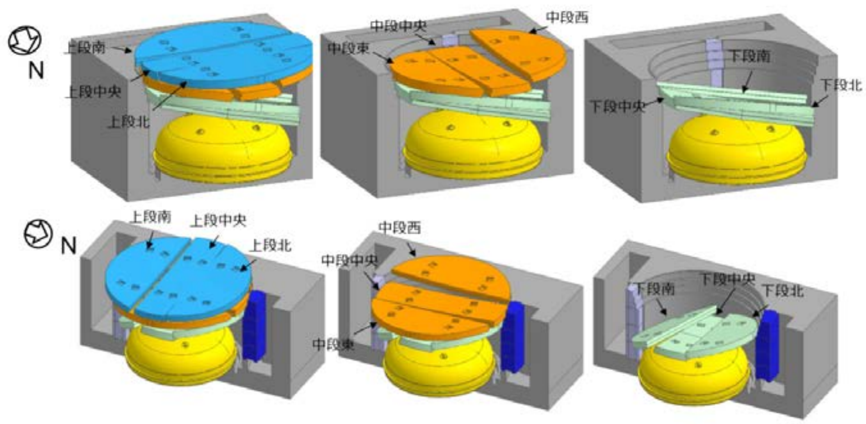
Area	1F1	1F2	1F3
Torus room	Partially flooded [148, 149] Rusted handrails/equipment [12] TBD	Partially flooded [150] Non-rusted handrails/ equipment [12,151] Some room penetrations tested, no leakage observed [153]	Partially flooded [150] Non-rusted handrails/ equipment [12,152] TBD
MSIV room	Limited view obtained [17]	Water leakage cannot be observed [154]	Leakage in Line D near bellows [155]
DW head and shield plugs	Reactor well shield plug displaced [156]	Possible leakage [157]	Leakage likely due to radiation measurements at head and presence of H_2 burn [53,157]
RCIC or other low SC piping	TBD	Suspected leak location, not confirmed [12]	TBD
RPV lower head	Ex-vessel debris images, dose surveys, and sample examinations indicate failure [19, 53,159]	Ex-vessel debris and images confirm failure [158]	Ex-vessel debris images confirm failure [158]

a. Nomenclature: [Clear]: TBD (To be determined); no information available; [Red]: available information indicates damage or leakage; [Orange]: available information suggests possible damage; [Green]: available information indicates no damage.
b. X-100B is vacant for 1F1, allowing this penetration to be used for DW investigations.
c. X-51 is an instrument pipe penetration for measuring differential pressure in 1F2/1F3. The penetration is joined to the Standby Liquid Cooling (SLC) pump injection line in the drywell. This penetration is designated as X-27 in 1F1.
d. X-53 is vacant for 1F2 and 1F3, allowing these penetrations to be used for DW investigations.
e. X-53 is the HPCI steam supply penetration and X-54 is the HPCI steam instrument pipe penetration for 1F1. X-11 is the HPCI steam supply penetration for 1F2 and 1F3.

8

Investigation of Unit 1 Shield Plugs
Physical surveys

- Expanded surveys
  - 3-D scanning



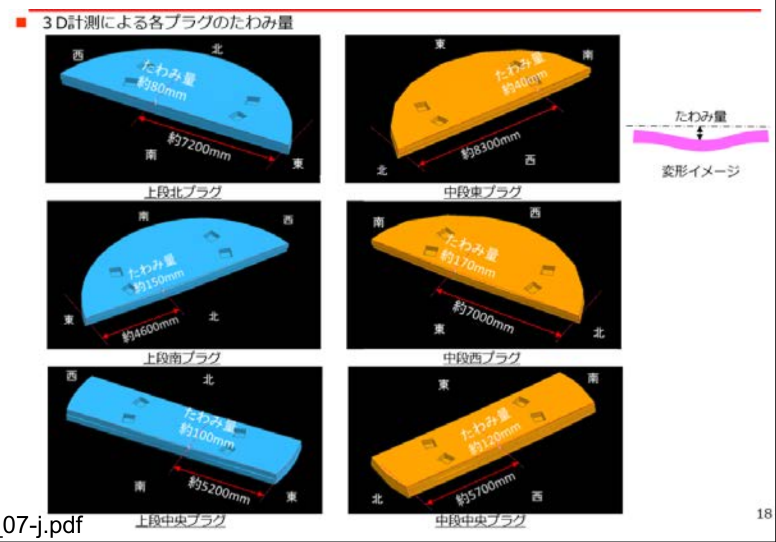
3D計測を基に作成したイメージモデル

From: d190926_07-j.pdf

9

Investigation of Unit 1 Shield Plugs
Physical surveys

- 40-170 mm deflections measured



From: d190926_07-j.pdf

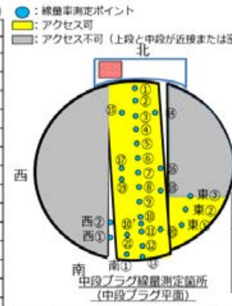
18

Investigation of Unit 1 Shield Plugs
Radiological surveys

- Between Upper and Middle plug
- Max was1970 mSv/h which is lower than 2017 survey, consistent with Cs-137 decay

>1500mSV/h | >1000mSV/h （単位：mSv/h）

測定箇所	低所 下向き	低所 上向き	中所 下向き	中所 上向き	高所 下向き	高所 上向き
①	850	700	–	–	–	–
②	1390	1010	–	–	–	–
③	1640	1250	–	–	–	–
④	1290	1330	–	–	–	–
⑤	1560	1380	1530	1260	–	–
⑥	1560	1510	1550	1270	–	–
⑦	1720	1240	1560	1360	–	–
⑧	1570	1200	1260	1120	–	–
⑨	760	730	920	700	–	–
⑩	840	820	800	800	–	–
⑩'	1080	860	1000	760	–	–
⑪	1250	920	1010	790	940	820
⑫	1400	900	880	930	800	700
⑬	1090	700	840	690	600	460
⑭	1630	1210	–	–	–	–
⑮	1370	1000	–	–	–	–
⑯	1970	1330	1390	1170	–	–
⑰	1550	1200	1280	1040	–	–
⑱	1520	1140	1220	1020	–	–
⑲	1520	1070	1130	950	–	–
⑳	1350	860	870	860	840	700
㉑	1540	940	980	730	720	620



（単位：mm）

各測定位置の高さ		ロボット姿勢 低所	ロボット姿勢 中所	ロボット姿勢 高所
線量計の向き	上向き	240	470	690
線量計の向き	下向き	20	250	470

（単位：mSv/h）

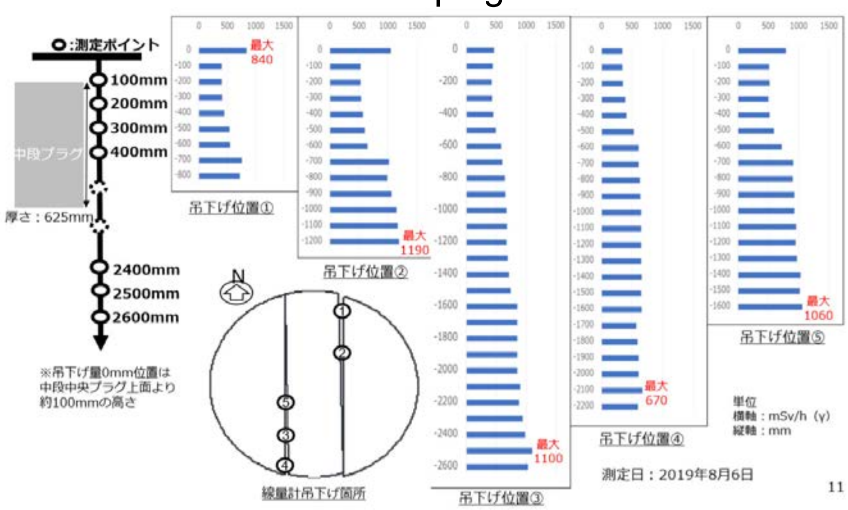
測定箇所	低所 下面	低所 上面	低所 壁面	中所 下面	中所 上面	中所 壁面	高所 下面	高所 上面	高所 壁面
西①	640	630	–	–	–	–	–	–	–
西②	690	660	–	–	–	–	–	–	–
東①	1350	930	–	900	950	–	–	–	–
東②	850	830	–	920	780	–	–	–	–
東③	960	770	–	730	690	–	–	–	–
南①	1240	920	920	850	710	700	650	690	660

15

From: d190926_07-j.pdf

Investigation of Unit 1 Shield Plugs
Radiological surveys

- Between Middle and Lower plug



11

From: d190829_07-j.pdf

12

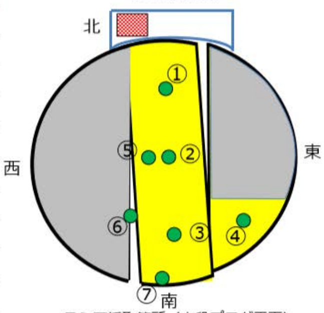
Investigation of Unit 1 Shield Plugs
Radiological surveys

- Smear results from Middle plug

分析結果 (Bq/cm²)

No	測定箇所	γ線放出核種				α線放出核種※1
		Cs-134	Cs-137	Co-60	Sb-125	
①	上段プラグ北側下面	7.0E+3	1.0E+5	6.4E+1	4.8E+3	6.4E-1
	中段プラグ北側上面	4.7E+3	6.9E+4	1.6E+1	8.4E+2	6.4E-1
②	上段プラグ中央下面	1.1E+4	1.6E+5	5.5E+1	4.4E+3	6.4E-1
	中段プラグ中央上面	－※2	2E+6	< 8E+1	－※2	1.1E+0
③	上段プラグ南側下面	6.2E+3	9.2E+4	6.3E+1	5.7E+3	6.4E-1
	中段プラグ南側上面	5.9E+3	8.7E+4	<2.6E+1	7.2E+2	<5.7E-1
④	上段プラグ東側下面	1.3E+3	1.9E+4	2.7E+1	1.8E+3	<5.7E-1
	中段プラグ東側上面	1.3E+3	1.9E+4	4.8E+0	1.9E+2	8.5E-1
⑤	上段プラグ中央下面	1.5E+4	2.2E+5	8.7E+1	6.7E+3	<5.7E-1
	中段プラグ中央上面	3.4E+3	5.3E+4	<1.1E+1	<3.2E+2	2.7E+0
⑥	中段プラグ西側側面	－※2	3E+6	< 1E+2	－※2	6.6E+0
⑦	南側ウエル壁	2.7E+3	3.9E+4	<1.0E+1	9.2E+2	1.3E+0

● : スミア採取ポイント
: アクセス可
: アクセス不可（上段と中段が近接または密着しているため）



スミア採取箇所（中段プラグ平面）

※1：ZnSｼﾝﾁﾚｰｼｮﾝｻｰﾍﾞｲﾒｰﾀによる定量結果
※2：Ge半導体スペクトロメータでは、線量が高すぎて過小評価（ﾃﾞｯﾄﾞﾀｲﾑ高）となることから、別のスペクトル測定器（CZT）で計測。標準線源を所持している核種のみ定量した。

From: d190926_07-j.pdf

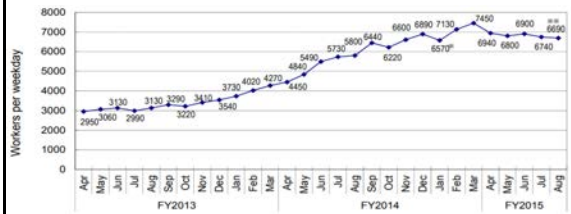
6

Other

- Began in August to dismantle shared stack of Unit 1&2 from the top-down
  - They are surveying and smearing removed sections
  - *d190725_07-j.pdf, d190926_07-j.pdf, hd03-02-03-001-d190926_01-e.pdf*
- Continued effort to clear unit 1 floor debris and collapsed roof.
- Continued surveys of unit 1 SFP
  - Gates were confirmed to be leak tight and not deformed

14

Other – Workforce



From: d151001_01-e.pdf

From: d171026_01-e.pdf

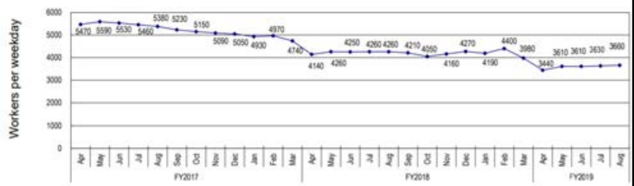


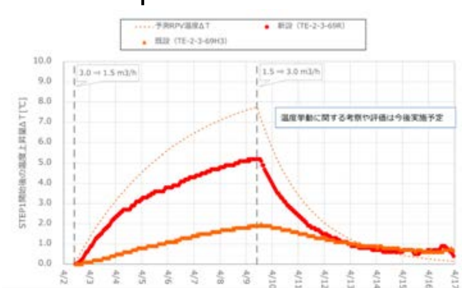
Figure 7: Changes in the average number of workers per weekday for each month since FY2017 (actual values)

From: hd03-02-03-001-d190926_01-e.pdf

15

Suspension of Water Injection

- Purpose: Optimize emergency response procedures
- April 2019: Unit 2 test
  - STEP1 – reduce flow from 3.0 to 1.5 m^3/hr
  - Bottom head temperature increase from 20.2 to 25.4 °C



http://www.tepco.co.jp/decommission/information/committee/roadmap_progress/pdf/2019/d190425_10-j.pdf

16

Suspension of Water Injection

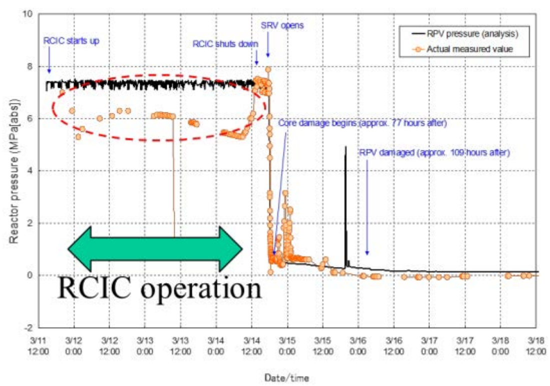
- October 14-17, 2019: Unit 1 test
  - Injection temporarily suspended
  - Bottom head temperature increase was 0.2 °C
  - PCV temperature increase was 0.6 °C
  - Both as expected
  - Similar test at Unit 3 planned for March 2020

17

NURETH-18 Presentation

Shinya Mizokami – Tokyo Electric Power Company Holdings, Inc

- Transient Behavior with Uncontrolled RCIC operation in Unit 2
  - RCIC restarted 2 minutes before loss of DC
  - RCIC flow exceeds decay heat after 15 minutes
  - Initial MAAP analysis did not capture RPV pressure response

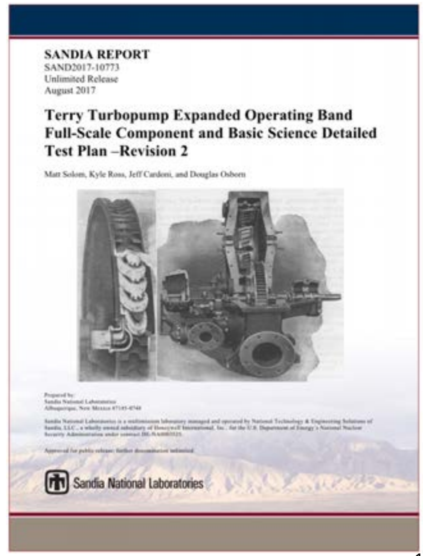


18

NURETH-18 Presentation

Shinya Mizokami – Tokyo Electric Power Company Holdings, Inc

- Conclusions from RCIC Analysis
  - *…there is a possibility that this uncontrolled operation, representing a sort of passive system, can become a measure to enhance the nuclear safety.*
  - *…US-Japan collaboration to understand the RCIC behavior under such extreme condition and limit of its functioning is now on-going*



SANDIA REPORT
SAND2017-10773
Unlimited Release
August 2017

Terry Turbopump Expanded Operating Band Full-Scale Component and Basic Science Detailed Test Plan –Revision 2

Matt Solom, Kyle Ross, Jeff Cardoni, and Douglas Osborn

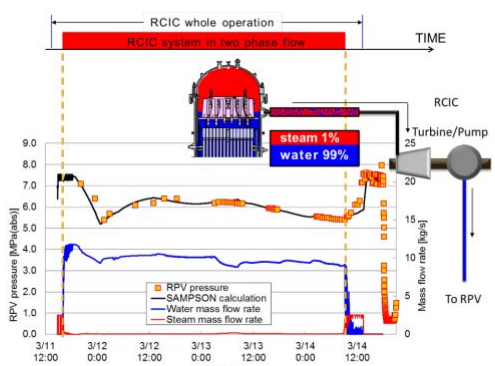
Sandia National Laboratories

19

NURETH-18 Presentation

Shinya Mizokami – Tokyo Electric Power Company Holdings, Inc

- Surplus water resulted in level increase to main steam line
- RCIC turbine driven by 2-phase mixture
- 2-phase discharge resulted in reduction in RPV pressure below SRV set point
- With corrected boundary conditions, SAMPSON code predicts RPV response



20

NURETH-18 Presentation

Shinya Mizokami – Tokyo Electric Power Company Holdings, Inc

- Vessel Failure Status Unit 1
  - BSAF results from 2015 show total amount of debris released varies from 45% to above 100% of total inventory
  - Good agreement among codes for this simple scenario is an indication that governing phenomena is well understood
  - Muon detector analysis confirms large amount of discharge

21

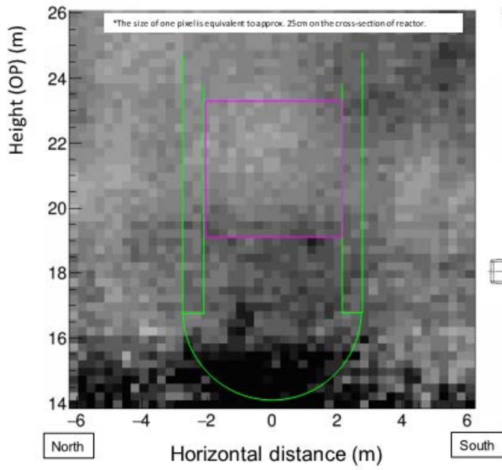
NURETH-18 Presentation

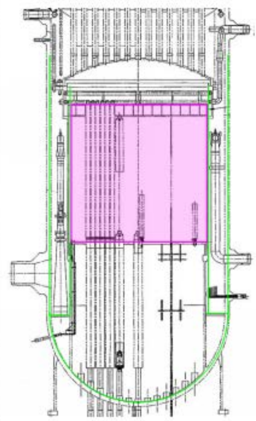
Shinya Mizokami – Tokyo Electric Power Company Holdings, Inc

- Vessel Failure Status Unit 2
  - BSAF from 2015 shows a wide spread of results due to uncertainties in core degradation, melt generation and relocation
  - Several codes predicted in-vessel retention with total core degradation ranging from 20% to 70%
  - Three calculations predicted vessel failure and 2 of 3 predicted MCCI in pedestal region
  - Muon detector analysis confirms large amount of debris in lower head

22

NURETH-18 Presentation
Shinya Mizokami – Tokyo Electric Power Company Holdings, Inc





Structures at the lower part of the reactor pressure vessel

23

NURETH-18 Presentation
Shinya Mizokami – Tokyo Electric Power Company Holdings, Inc

- Unit 2 PCV Deposits
  - Modeling challenges when system geometry changes
  - Original code design based on TMI-2 behavior
  - PCV investigations have yielded significant insights on vessel failure

The conclusion at this time is that there is not enough knowledge about RPV bottom failure

24

NURETH-18 Presentation

Shinya Mizokami – Tokyo Electric Power Company Holdings, Inc

- Overall Conclusions
  - *In some cases, SA code was confirmed well validated, such as the prediction of large core degradation in unit 1.*
  - *In some others cases, there is large inconsistency between observation and calculation.*
  - *In some of them we have realized that the state of knowledge is still unknown (*e.g. RCIC operation in 2-phase flow).

To improve the SA code ability is directly connected to the enhancement of nuclear safety.

25

C.2.2. Topic Area 2 - Radiation Surveys and Sampling

Topic area leads did not provide any presentations.

C.2.3. Topic Area 3 - Debris Endstate

WE START WITH YES.

Argonne NATIONAL LABORATORY

TOPIC 3 - CORE DEBRIS LOCATION EVALUATIONS

MITCH FARMER
Nuclear Science & Engineering Division
Argonne National Laboratory

Fukushima Forensics Meeting, November 18-19, 2019
Argonne Offices, 955 L'Enfant Plaza, North, SW, Suite 6000, Washington, DC 20024-2168

TOPICS FOR DISCUSSION

- Insights and comments based on findings from Fukushima related to MCCI and debris coolability
  - Summary of insights from SAWA/SAWM study to support discussions
  - 1F1
  - 1F2 (including recent TEPCO findings)
  - 1F3
- Revisions to information requests
- Update on status of the ROSAU project

2 Argonne

SAWA/SAWM STUDY (BASED ON PEACH BOTTOM)

- For MAAP5 and MELCOR pour conditions, MELTSPREAD3 used to calculate post-spread core debris distribution in containment.
  - Melt pour temperature & water depth parameterized to examine sensitivities.
- Multi-nodal CORQUENCH4 model then used to evaluate extent of ablation, time to debris quench, NC gas production, and water spillover into torus for various flooding scenarios:
  - Location of water injection (i.e. core vs. drywell sprays)
  - Timing of water addition (i.e. 0-8 hours after RPV failure)
- Assumed water addition:
  - SAWA at 500 gpm (5 hours)
  - SAWM at 100 gpm for balance of calculation
- All cases run to 72 hours.

Pour Conditions

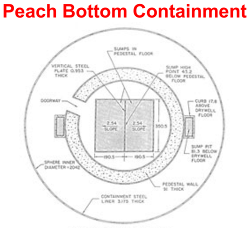
Case Designator	MELCOR	MAAP
Onset of Pour (hours)	12.95	14.51
Water level/temp. on cavity floor at vessel failure (cm/K)	58/336	'Dry'
Pour Duration (sec)	2223	53
Melt Pour Characteristics	• Rapid oxide pour over ~300 seconds • Slow metal pour over the duration	Uniform melt composition during pour
Melt Pour Temperature (K)	1758-1770	934-2240
Melt Solid Fraction	0.696	0.689
Total Pour Mass (MT)	327	298

3 Argonne

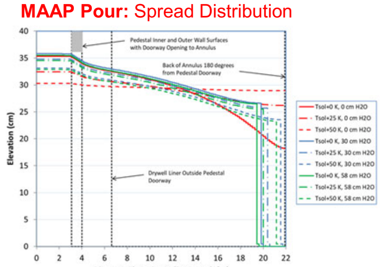
MELTSPREAD3 PREDICTIONS OF SPREADING BEHAVIOR: MAAP5 POUR CONDITIONS

- Extent of spreading reduced with presence and depth of water initially on cavity floor.
- Increase in melt temperature increases spreading extent.
- For water depths up to 1 m (including sump), effect of jet breakup on spreading is small.
  - ~ 5 % of pour mass rendered into particle bed.

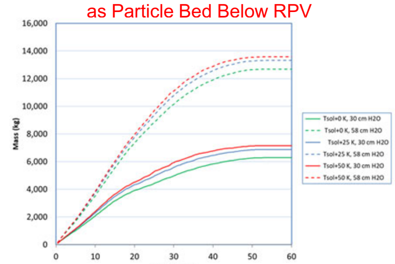
Peach Bottom Containment



MAAP Pour: Spread Distribution



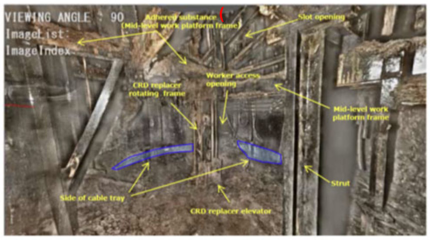
MAAP Pour: Mass of Material Rendered as Particle Bed Below RPV



Reference: M. T. Farmer, "A Case Study on Severe Accident Water Management for a Mark I Containment," ANL-18/21, September 2018. 4 Argonne

MELTSPREAD3 PREDICTIONS OF SPREADING BEHAVIOR: MELCOR POUR CONDITIONS

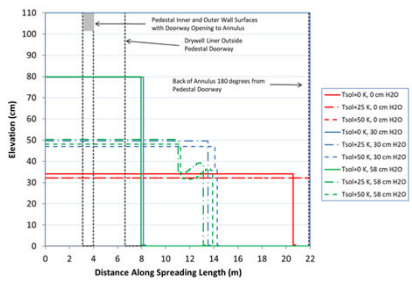
Fukushima Unit 2 In-Pedestal Core Debris



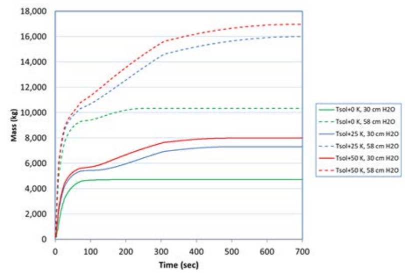
Post-Spread Metal-Oxide Distribution

Node	Location	Constituent in debris at location (wt%)	
		UO_2	SS (Fe, Cr, Ni)
1	Pedestal sumps	22.9	68.6
2	Pedestal floor (ring around sump)	55.6	16.2
3	Pedestal doorway opening	62.7	6.0
4	Floor area outside doorway	67.3	3.5
5	Drywell floor adjacent to doorway	70.0	2.5

MELCOR Pour: Spread Distribution



MELCOR Pour: Mass of Material Rendered as Particle Bed Below RPV



5

MAJOR FINDINGS RELATED TO WATER MANAGEMENT

- For all cases considered (MAAP5 vs. MELCOR pours, dry cavity vs. flooded cavity at RPV failure, 0-8 hour water injection delay, core vs. drywell water injection), the core debris was eventually quenched and stabilized within the 72 hour calculated time interval.
- For MAAP cases in which all post-spread melt depths were less than the downcomer height (58 cm), the time to debris quench was relatively insensitive to the location of water addition.
- However, for the low temp MELCOR pours where initial debris height is greater than the dowcomer height, the time to debris quench as well as the extents of cavity ablation and NC gas production were all substantially increased for the drywell injection case for short injection delays of ~ 5 hours or less.
  - Reason: deep accumulations in pedestal formed a dam, thus allowing water to spill over into the torus which has a lower inlet elevation.
- However, due to concrete densification upon melting (slump) during MCCI, the differences become much smaller after ~ 5 hours since the debris upper surface elevation is eventually reduced below the downcomer inlet height.
- The above results are sensitive to plant concrete type as well as the height of the downcomer inlet.

6

Argonne

DISCUSSION ON 1F1

- Based on limited robotics examinations inside the PCV as well as muon tomography, it is thought most of the core inventory has exited the RPV and is in the pedestal/drywell regions.
- The presence of significant accumulations of material in the drywell outside the pedestal doorway (~0.8-1.0 m) has been identified.
- At the X-100B location, ~130 degrees from the pedestal doorway, material ~30 cm deep has been found.
  - Covered by loose sediment, and it is not currently known how far down the loose sediment extends, and whether the sediment covers other material (e.g., core debris).
- Clear evidence that PCV liner has failed.
- Presence of core debris in pedestal region is consistent with MELCOR/MAAP/MS/CQ studies.
- Current thinking is that water was not actually injected into the RPV until ~ 12 days after the accident (due to valve misalignment).

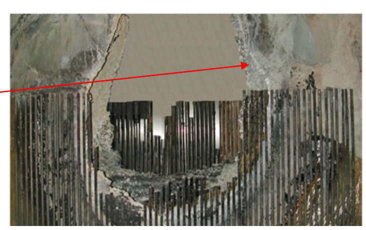
7

Argonne

DISCUSSION ON 1F1 (CONTD.)

- Argonne tests have shown that, for dry MCCI with siliceous concrete, a prodigious amount of Silica aerosol is formed.
- Thus, sentiment layer over the core debris at X100B location would be consistent with settling (or wash down?) of that aerosol to cover the debris.
- However, chemical analysis on this sediment reported by TEPCO at the last meeting is not consistent with this theory.
  - Analysis of white sample from HPCI room may provide additional insights.
- Also, based on limited observations inside the PCV, the extent of damage does not seem to be consistent with a completely dry scenario.
  - See below, as well as information on Chernobyl cavity ablation*.
- Thus, for 1F1, questions remain (in the author's opinion) on how much water actually made it to the RPV.

Test CCI-2 showing concrete ablation above melt due to radiation heat transfer

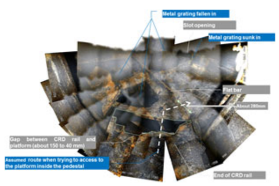


*see https://www.oecd-nea.org/nsd/pubs/2017/7392-soar-molten-corium.pdf

DISCUSSION ON 1F2

- Both Muon tomography and robotic exams have revealed valuable data on debris locations for 1F2.
- Initial access through the X-6 penetration revealed the presence of significant core debris retention on the CRD access platform.
- Additional entries revealed debris accumulation in the pedestal region that ranges from 40 to 70 cm thick.
- Despite the extent of this material, there did not appear to be significant damage to structures within the pedestal region, at least above the upper surface of the debris. May be attributable to:
  1. Significant water present on the pedestal floor when the vessel failed, and/or
  2. The debris in the pedestal region is predominately metallic, thereby containing a low fission product (decay heat) content.

Material holdup on below vessel structure



Core debris on pedestal floor

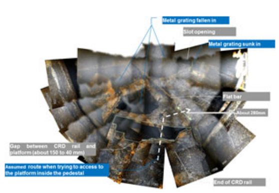
9

DISCUSSION ON 1F2 (CONTD.)

Linking the observations to the water SAWA/SAWM study results:

1. The debris depths and relatively flat debris profile are consistent with the MELCOR melt pour scenario.
2. The possibility of a high metal content in the pedestal region is also consistent with the MELCOR oxide-metal pour sequence.
3. The code results, as well as a conservation of mass argument made by the author during the 2018 meeting (see those viewgraphs) indicate the possibility of core debris in the drywell region also.
   - Based on water management study, liner would remain intact if water was present and sustained (as well as the results of the Mark I shell vulnerability assessment studies).

Material holdup on below vessel structure



Core debris on pedestal floor

10

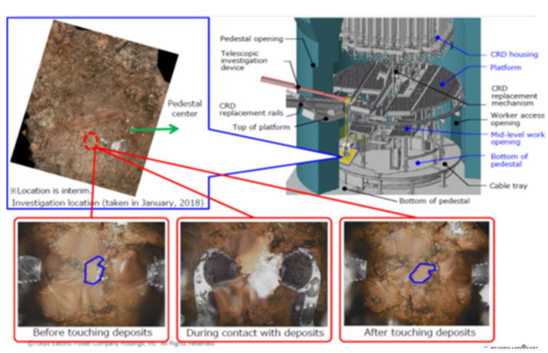
DISCUSSION ON 1F2 (CONTD.)

- As noted earlier, TEPCO reports debris depths in the pedestal region in the range of 40 to 70 cm, which is well above the water height of 30 cm in the drywell.
- Meanwhile, the ex-vessel core debris was quenched and stabilized and remains cooled via injection through the core.
- Thus, this finding indicates that water injection through the core and subsequent water flow over/through the debris is able to cool the material.
  - Supports the idea that injection through the core is desirable.
- Finally, the absence of an observable river-type flow of water over the core debris as it goes from the pedestal to the drywell indicates that water is able to penetrate into (and thus cool) the core debris.
  - This provides clear evidence that the water ingression cooling mechanism (currently modeled in CQ as well as MAAP and MELCOR) is viable.

DISCUSSION ON 1F2 (CONTD.)

- Recent robotic examinations on core debris located in the pedestal region completed to determine material mobility.
- A total of six locations were tested. The results indicated that loose (i.e. movable) material existed at five of the six locations tested.
  - Important information that will help in planning for debris removal.
- The mobility of the material is also consistent with reactor material tests that have consistently indicated formation of loose debris when the material is cooled with water.

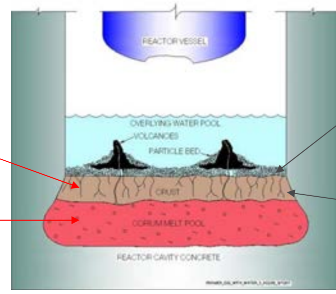
Robotic tool testing for mobility of core debris located in the 1F2 pedestal region

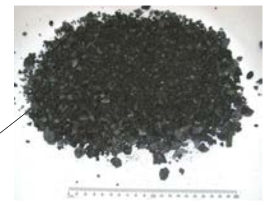


EXPECTED MORPHOLOGY FOR WET CAVITY CONDITIONS BASED ON TEST DATA

Photos shown are for tests with siliceous concrete

Loose material regions: easy to remove





~ 50 cm

Hard material: difficult to cut and/or break

13

Argonne

DISCUSSION ON 1F3

The results of Muon tomography as well as robotics examinations have provided valuable data on the debris distribution in 1F3. The results indicate that:

1. The CRD platform has been dislodged from the rails and a portion of it is buried under core debris.
2. The depth of the deposits is greatest in the center of the pedestal, and falls off as the pedestal wall is approached.
   – Trend consistent with lower head failure near the centerline, as opposed to 1F2 for which the data suggest that the lower head failed near the periphery
3. From the renderings, the debris is quite deep; i.e., in the range of 2-3 meters.

14

Argonne

DISCUSSION ON 1F3 (CONTD.)

- Mass of material in pedestal region can be estimated based on a simple argument.
- Assuming PCV dimensions similar to 1F2, then the sump volume is ~6.9 m^3, and the floor area within the pedestal is ~16 m^2.
- Further assuming a debris density of ~ 7 kg/l, and an average debris depth of 2 m over the pedestal floor, then the mass of core debris in the pedestal would be:
  - 224 MT assuming that the sump plates keep core debris out of the sumps, or
  - 270 MT if the sump plates failed and core debris is in the sumps.
- Based on the depth of material (> 2 m), the average loading on the sump cover plates if they remained intact would be ~137 kPa (~ 2900 lb/ft^2) which is significant.
- Thus, the chances the plates failed is pretty high, and the mass in the pedestal is likely closer to 270 MT. This is a significant fraction of the total core mass.
- Mass estimate would be lower if there is significant porosity in core debris.

15 Argonne

DISCUSSION ON 1F3 (CONTD.)

- A porous debris scenario could develop if the melt was fragmented by interaction with a deep water pool on pedestal floor during relocation, thereby forming a porous debris bed.
  - Would require a water depth on the pedestal floor at the time of vessel failure of several meters.
  - If plausible, then this very deep accumulation could have been rendered 'coolable' if sufficient makeup water was provided.
  - Scenario could be assessed with MELTSPREAD3 given the jet fragmentation model, if desired.
- Given the existing water ingression correlation developed based on Argonne testing, then this deep accumulation may not be coolable based on water ingression cooling of an initially molten core melt pool.
  - Observation based on current validation database that is limited to low metal content, PWR-type core melts.
  - This specific data gap is being addressed as part of the ROSAU test program.

16 Argonne

DISCUSSION ON 1F3 (CONTD.)

- Additional examinations by TEPCO will hopefully provide data on morphology of core debris in 1F3 pedestal.
- This information will be very beneficial to support coolability assessments, coolability modeling, and overall plant safety evaluations.
- Finally, given the geometry of the core debris, the fact that the material could be cooled continues to support the concept that injection through the RPV is the preferred injection pathway.

17 Argonne NATIONAL LABORATORY

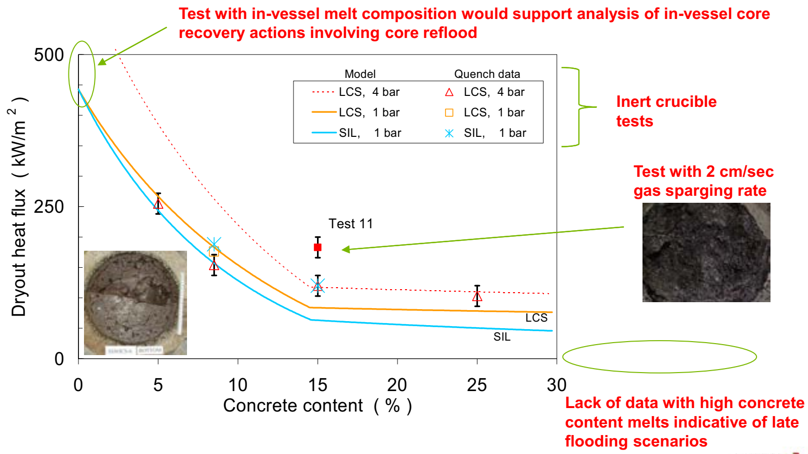
REVISIONS TO INFORMATION REQUESTS

- Effectively, none.
- However, if possible please pay particular attention (and characterize) the debris morphology for 1F3.
  - The deep (minimum of 2 m) accumulation of material for this case was apparently cooled, and this may (?) not be expected based on the current simplistic vision of water ingression cooling of a once molten melt pool.
  - If significant debris fragmentation occurred, this would resemble in some respects the accident management strategy for Swedish BWRs, and be important to characterize.
- Of course, this request has to be factored into the overarching need for TEPCO to D&D these reactors.
  - Thus, this request would only be made if it did not hinder progress to meet the above goal, and/or increase dose to workers performing the work.

18 Argonne NATIONAL LABORATORY

EXISTING WATER INGRESSION DATABASE

- Principal question is whether total decay heat in the debris divided by the pedestal area (yielding the effective decay heat flux in the debris) is less than the water ingression 'dryout' heat flux.
- If not, water ingression cooling will not be effective in terminating the accident.



Argonne

REDUCTION OF SEVERE ACCIDENT UNCERTAINTIES (ROSAU) PROGRAM STATUS

- Program launched as an international OECD project in September 2019.
  - NRC and EPRI are the current participating US organizations
- Program objective is to address two knowledge gaps in LWR severe accident progression identified following the reactor accidents at Fukushima Daiichi; i.e.,

1. Coolability of high metal content (BWR-type) core debris, and
2. The effect of water on core debris spreading following vessel failure.

- As part of the steps leading up to the program, developmental testing was carried out to develop exothermic chemical mixtures that can produce BWR-like in-vessel core debris melt
  - Highly successful; the developed mixtures do not rely on the use of U metal, which simplifies operations and allows large scale tests to be conducted.
  - Will minimize potential scaling distortions associated with variations in melt composition expected at reactor scale.
- We are on track to conduct 5 water ingression experiments as well as 6 large scale core debris spreading experiments to address the above knowledge gaps.
- Program includes a parallel model development/validation program to build the test results into enhanced codes that will serve as the legacy of the test program.

Argonne

C.2.4. Topic Area 4 - Combustible Gas Effects

Fukushima Forensics: Combustible Gas Effects
The explosions at Fukushima Daiichi unit 3 and unit 4 and implications on the evaluation of 1F3 accident

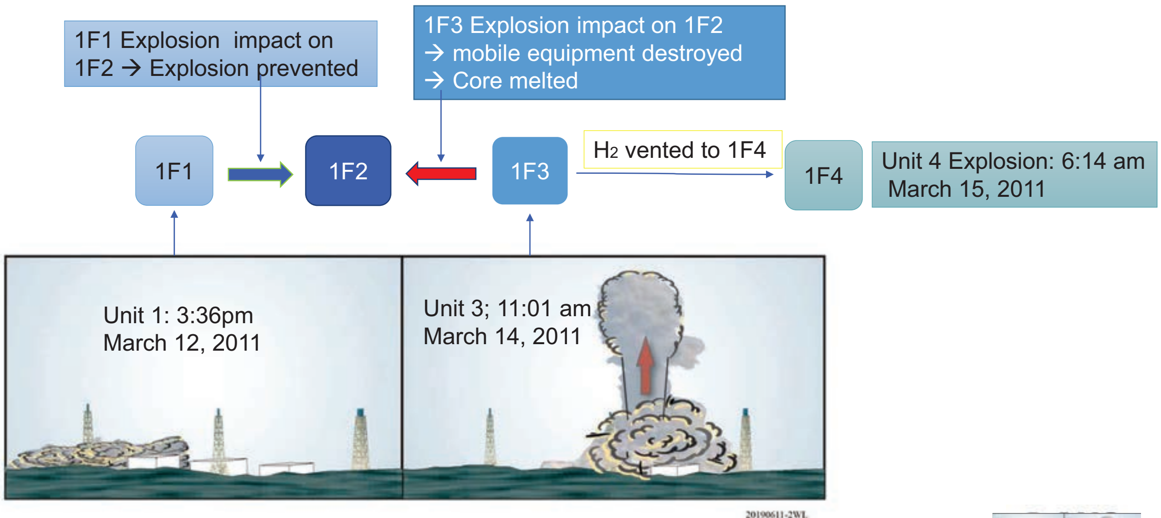
Wison Luangdilok
Fauske & Associates LLC
H2Technology LLC

DOE Reactor Safety Technology Expert Panel Forensics Meeting
Argonne National Laboratory Offices
Washington, DC
November 18-19, 2019

Motivation

- Hydrogen explosions are important accident data in addition to RPV pressure, RPV water level, and PCV pressure.
- These data include explosion time and the amount of hydrogen burned in the explosion.
- These data have Implications on
  - 1F3 accident progression
  - Reactor vessel failure time
  - Timing of MCCI attack
- Severe accident code analysis of the 1F3 should consider benchmarking the analysis results against these data.

Fukushima Daiichi Accidents and Hydrogen Explosions



3 H2TECHNOLOGY LLC

The 1F3 Explosion was more powerful than others

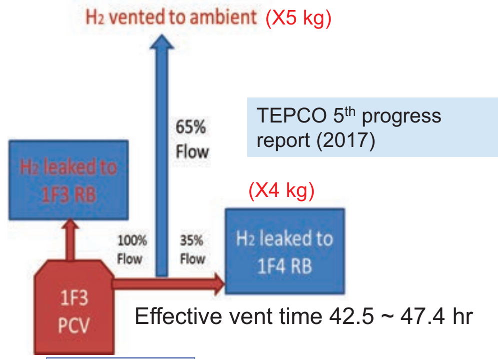
- It moved large heavy objects high into the sky.
- Big pieces of concrete or equipment were thrown into SFP.
- The explosion destroyed concrete surfaces and generated a large amount of dust that was pulled into the sky by the rising hot burned gases.

Images of reactor building explosions were removed

4 H2TECHNOLOGY LLC

Hydrogen Balance between Generation and Explosion

Total H2 generation ≈ H2 burned at 1F3 & 1F4
+ H2 vented to ambient+H2 in 1F3 PCV
X1+X2 (code analysis) ≈ X3+X4+X5 (explosion data)
+ H2 in PCV



Effective vent time 42.5 ~ 47.4 hr

Accident initiated → Severe core damage after RPV water level below BAF → Reactor lower head failure at time t1 → 1F3 Explosion (X3 kg) t=68.3 hr

In-vessel H2 generation (X1 kg) due to HT steam oxidation of B4C, SS, Zry and their eutectics

Molten core debris in contact with concrete

Ex-vessel H2 generation (X2 kg) due to MCCI:

Large-Scale Testing of an Explosion of Methane Gas from a leaked LNG Pipeline

Wind speed : 4-7 m/s

Temperature > 2200°C

Images of the pipeline explosion test were removed

Fireball diameter ~400 m

A 50-m long rupture on the 3.56-cm thick pipeline was a result of the explosion.

276,000 kg of CH4 burned

An explosion in a mushroom-shaped fireball occurred in 5 s

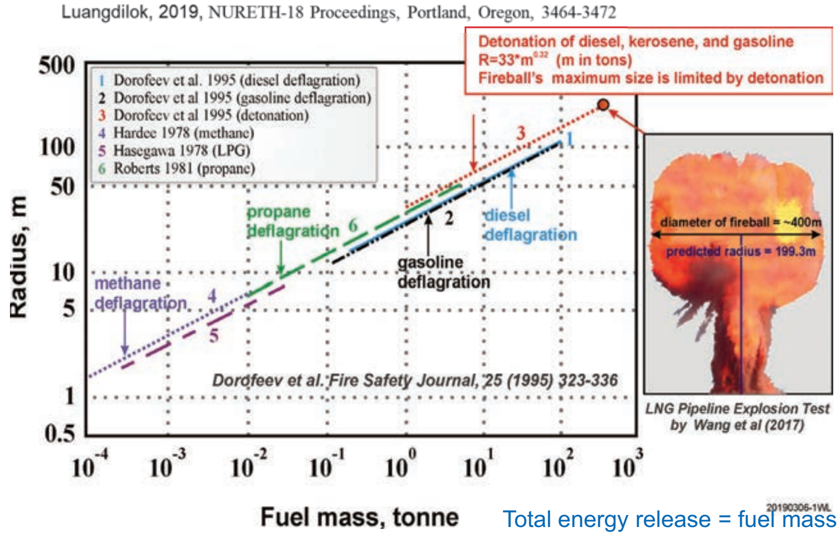
The explosion in the test was a detonation.

5-20 s: Stable and fully grown fireball
20-30 s: Fireball weakened and burnout
30-45 s: Height decreased and completely burned out.

Wang et al., J of Loss Prev. in the Proc. Industries, 46 (2017)13-22

Fireball Radius from Explosions as a function of total energy release

Luangdilok, 2019, NURETH-18 Proceedings, Portland, Oregon, 3464-3472



Fuel	Heat of Combustion, MJ/kg
Gasoline - conventional	43.44
Diesel - conventional	42.78
Kerosene	43.1
Average -gasoline, diesel, Kerosene	43.11
Propane	46.35
Methane	50
Hydrogen	120

Total energy release = fuel mass (kg) x heat of combustion (MJ/kg)

7 H2TECHNOLOGY LLC

Prediction of the Maximum Fireball Size from the LNG Pipeline Explosion using the Dorofeev Correlation

- For the same fireball size, the deflagration mode requires more mass of fuel to burn than the detonation mode.
- The maximum fireball size is limited by the detonation-mode fireball: R=33*(m)^0.32
- For the LNG pipeline explosion test, m =276 ton: R=33*(276)^0.32 = 199.3 m

Fireball diameter ~400 m

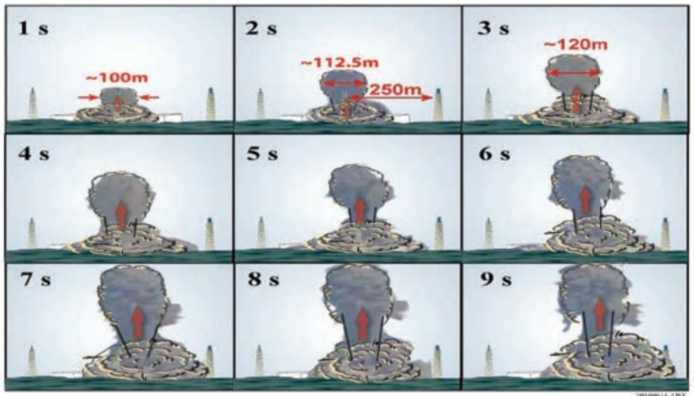
Predicted radius = 199.3m

Wang et al., J of Loss Prev. in the Proc. Industries, 46 (2017)13-22

8 H2TECHNOLOGY LLC

Size of the 1F3 Explosion Fireball

Luangdilok, 2019, NURETH-18 Proceedings, Portland, Oregon, 3464-3472



The Generalized Correlation

Luangdilok, 2019, NURETH-18 Proceedings, Portland, Oregon, 3464-3472

The fireball radius is expressed in terms of the combustion heat released rather than fuel mass.

$$R = 33\, {M_{hc}}^{0.32} \quad \Longrightarrow \quad R = 33\left(\frac{q_{H2}(1-\chi_{H2})}{q_{hc}(1-\chi_{hc})}\right)^{0.32}(M_{H2})^{0.32}$$

$$M_{H2} = \frac{(1-\chi_{hc})}{(1-\chi_{H2})}\frac{q_{hc}}{q_{H2}}\left(\frac{R}{33}\right)^{\frac{1}{0.32}}$$

$\chi_{hc} = \dfrac{3.6\times10^{6}}{q_{hc}}\left(\dfrac{R}{33}\right)^{-0.156}$ Radiative heat loss fraction for hydrocarbon

$\chi_{H2} = \chi_{hc} - 0.03$ * Radiative heat loss fraction for hydrogen

q = Heat of combustion of fuel (MJ/kg)

M = Total burned mass M (metric ton)

R = Fireball radius

*Molina, et al., 2007. Radiative fraction and optical thickness in large-scale hydrogen-jet fires. Proc. of the Combustion Inst. 31, 2565–2572.

Application to 1F3

- Fireball radius R = 53.125 m
- The mass of hydrogen burned (in metric ton) is given by

$$M_{H2} = \frac{1-\chi_{hc}}{1-\chi_{H2}} \frac{q_{hc}}{q_{H2}} \left(\frac{R}{33}\right)^{\frac{1}{0.32}}$$

χ_{hc} = 0.0775

χ_{H2} = 0.0475

q_{hc}= 43.11 MJ/kg,

q_{H2}=120 MJ/kg,

- Mass of hydrogen burned at 1F3 explosion = 1540 ±250 kg

Total Minimum Required H2 Generation

Source of H2	Amount of H2	Method or assumptions
H2 burned at 1F3	1540 ±250 kg	Estimated from the fireball size.
H2 in 1F3 RB (X3) +PCV	~1448 kg (75% H2 limit) +637 kg (PCV) = 2085 kg	Assumed high-end 75%H2 concentrations in RB 5F/4F at 323K/310K, 30%H2 in DW at 700K, 40%H2 in WW at 425K.
H2 in 1F4 RB (X4)	207 kg	Assumed near low-end concentrations in RB: 10%H2 in RB 5F/4F at 300K.
H2 vented through the 1F3/1F4 common stack (X5)	=(65/35)x H2 leaked to 1F4 = 383 kg	35% of vent flow from 1F3 leaked to 1F4
Estimate of total H2 generation by time of 1F3 explosion	2675 kg	Total sum of X3+X4+X5 +H2 (in 1F3 PCV)

Potential Sources and Amount of Hydrogen Generation in the 1F3 Accident

Core Component	Potential source (kg)	Potential H2 generation (100% oxidation) (kg)
Zr in fuel cladding	29000	1272
Zr in channel box	18000	789
Fe in control blade	12800	641
B4C in control blade	960	243
Total (kg)	60760	2945

BASF Phase 2 Results* for 1F3 Analyses

Participating organization	SA code used in the analysis	Calculated in-vessel H2 generation (kg)	Calculated ex-vessel H2 generation (kg)	Total calculated H2 generation (kg) up to 1F3 explosion vs expected value from ballpark estimate	Calculated RPV failure time (hr)	RPV failure mode
VTT	MELCOR	1220	1200	2420 vs 2675	43.3	penetration
SNL	MELCOR	1010	700	1710 vs 2675	58	user specified
JAEA	THALES/KICHE	790	875	1665 vs 2675	46.5	vessel melt
IAE	SAMPSON	790	500	1290 vs 2675	55.2	creep
PSI	MELCOR	1180	0	1180 vs 2675	73.1	penetration
IRSN	ASTEC	1150	0	1150 vs 2675	55.4	creep
NRA	MELCOR	910	100	1010 vs 2675	49.4	penetration
CRIEPI	MAAP5	600	0	600 vs 2675	102	penetration

*Results extracted from Lind, et al., 2019. Overview and Outcome of the OECD/NEA Benchmark Study of the Accident at the Fukushima Daiichi NPS (BSAF), Phase 2 – Results of Severe Accident Analyses for Unit 3, NURETH-18 Proceedings, Portland, Oregon, 1133-1146.

BASF Phase 2 Hydrogen Generation Results* for 1F3 MELCOR Analyses vs. Expectation

Participating organization	SA code used in the analysis	Calculated in-vessel H2 generation (kg)	Calculated ex-vessel H2 generation (kg)	Total calculated H2 generation up to 1F3 explosion (kg)	Total minimum expected H2 generation up to 1F3 explosion (kg)	Calculated RPV failure time (hr)	RPV failure mode
VTT	MELCOR	1220	1200	2420	2675	43.3	penetration
SNL	MELCOR	1010	700	1710	2675	58	user specified
PSI	MELCOR	1180	0	1180	2675	73.1	penetration
NRA	MELCOR	910	100	1010	2675	49.4	penetration

*Results extracted from Lind, et al., 2019. Overview and Outcome of the OECD/NEA Benchmark Study of the Accident at the Fukushima Daiichi NPS (BSAF), Phase 2 – Results of Severe Accident Analyses for Unit 3, NURETH-18 Proceedings, Portland, Oregon, 1133-1146

15

H2TECHNOLOGY LLC

BASF Phase 1 Hydrogen Generation Results*

Participating organization	SA code used in the analysis	Calculated in-vessel H2 generation (kg)	Calculated ex-vessel H2 generation (kg)	Total calculated H2 generation (kg)	Total minimum expected H2 generation (kg)	Calculated RPV failure time (hr)
SNL	MELCOR 2.1-5864	1950	250	2200	2675	63
EPRI	MAAP5.01	360	1300	1660	2675	60
IAE	SAMPSON	1310	0	1310	2675	n/a
IRSN	ASTEC	1300	0	1300	2675	n/a

*Results extracted from "Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF Project), Phase I Summary Report March 2015, Nuclear Regulation NEA/CSNI/R(2015)18.

16

H2TECHNOLOGY LLC

Result Implications

- The 1F3 explosion data demands
  - that the in-vessel hydrogen generation, time of vessel failure and ex-vessel hydrogen generation must be well coordinated in the code analysis to produce at the minimum the combined mass of ~2425 kg of H2 or H2 equivalent by the time of the 1F3 explosion.
- What was the source of combustible gases feeding the 1F3 mushroom-shaped fireball?
  - The result suggests the possibility that a large amount of unmixed hydrogen or hydrogen-equivalent gases had been accumulating at extremely rich concentrations in the reactor building including the 5th floor and the 4th floor prior to the explosion.

Some Discussion of the Missing Model in SA Codes

- A comparison of the estimate of the total amount of hydrogen that must be generated during the 1F3 accident with the BASF phase 2 study shows an encouraging result.
- One of the analyses by VTT using MELCOR comes very close to this estimate (2420 kg vs. 2675 kg).
- However, at this time there is no information regarding the underlying assumptions and any particular oxidation models used in this analysis.
- It is expected that in order to generate this large amount of hydrogen (while most other analyses generate much less), the key source of potential hydrogen that is unique to the BWR design must be included.

Some Discussion of the Missing Model in SA Codes

- Key source of potential hydrogen generation recently identified by Steinbrück (2014)* and re-emphasized by Kurata et al. (2018)**
- The oxidation of B_4C containing melt such as B_4C-SS-Zry melt due to eutectic interaction of B_4C and stainless steel (SS) in the control blade and zircaloy (Zry) in the channel box.
- The oxidation of the eutectic melt of these core components has been confirmed by experiments at KIT to be significantly higher than that of pure B_4C, pure Zry, and pure steel whose oxidation kinetics are well established

Core Component	Potential source (kg)	Potential H2 generation (100% oxidation) (kg)
Zr in fuel cladding	29000	1272
Zr in channel box	18000	789
Fe in control blade	12800	641
B4C in control blade	960	243
Total (kg)	60760	2945

* Steinbrück, J. Nucl Mat. 400 (2010) 138-150

** Kurata et al. J. Nucl. Mat 500 (2018) 119-140

19 H2TECHNOLOGY LLC

Some Discussion of the Missing Model in SA Codes

- B4C reacts with the stainless steel cladding, forming a mixture that melts from 1200 C onwards and reacts with the Zircaloy of the channel box.
- The reaction between the B4C/SS melt and Zry is strongly affected by the thickness of the oxide layer formed on the Zry channel box surface prior to contact.
- There appear to be threshold conditions of steam flow and temperature ramp rates that determine whether only the degradation of the control blade occurs or both degradations of the control blade and the Zry-channel box occur.
- When the oxide layer is sufficiently thin, the B4C/SS-melt can rapidly attack Zry, causing the degradation of both the control blade and the channel box: JAEA test (low steam flow), XR2-1 test (pre-oxidized Zr, inert atmosphere)
- When the oxide layer is sufficiently thick, the oxide layer prevents the liquefaction of Zry, and therefore only the control blade melts down in the particular control blade channel: JAEA test (high steam flow), CORA-16
- At ~1250°C, interactions rapidly produce complex (low-viscosity) melts.

20 H2TECHNOLOGY LLC

Concluding Remarks

- B4C/SS melt is formed at temperature as low as 1174°C.
- JAEA control blade tests have identified the conditions for the B4C/SS melt to either form a flow blockage at the bottom of the fuel bundle or to drain out from the bottom of the bundle like a liquid.
- The conditions are related to the ability of B4C/SS melt to attack the channel box. The channel box can be protected from the attack if the Zry surface is sufficiently oxidized.
- The conditions will have significant ramifications for subsequent core melt progression.
- Research is needed to identify these protective conditions; for example, the condition might be the thickness of the protective oxide layer that prevents the attack.
- Research is also needed to quantify the oxidation kinetics of the eutectic melt of B4C/SS/Zry for modeling of the hydrogen production from such materials.

C.2.5. Topic Area 5 - Operations and Maintenance

C.2.5.1. BWROG EPC Update

Operations and Maintenance - BWROG EPC Update

Bill Williamson (TVA – Browns Ferry)

Phil Ellison (GEH)

Ken Klass (Talen - Susquehanna)

DOE Forensics Meeting
November 18-19, 2019
Washington, DC

BWR OWNERS' GROUP

BWR Expertise – Proven Solutions

Copyright 2019, BWR Owners' Group, All Rights Reserved

Topics

- Status Report on Implementing the EPG/SAG procedure changes to the BWR fleet based on lessons learned from investigations at Daiichi
- Implications for Operations, Maintenance, Severe Accident Mitigation, and Accident Analysis
- Computer Based Training (CBT) - Using INPO's NANTeL System
- Instrumentation Practical Insights (CBT)

Copyright 2019, BWR Owners' Group, All Rights Reserved

November 18-19, 2019 2

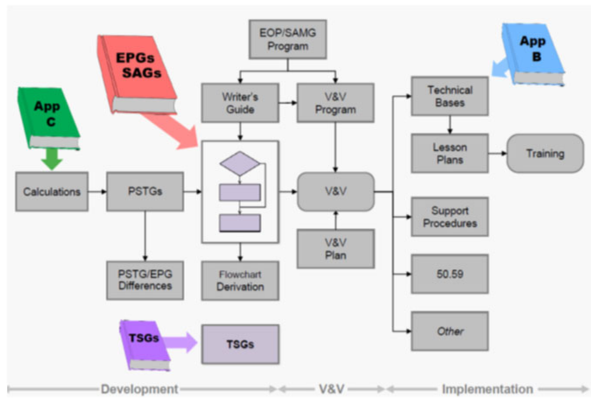
Implementation Status – EPG/SAG Rev 4 and TSG Rev 1

Implementation Schedule:

- Early sites late 2020
- Most sites plan for Spring 2021 (~June)
- Last sites are Fall 2021 and Spring of 2022

BWROG is developing NANTeL CBT for SAG Training - expected Fall 2020

- based on PWROG's success with CBT



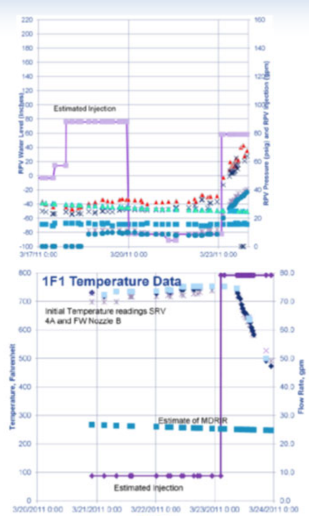
Copyright 2019, BWR Owners' Group, All Rights Reserved

Implications for Operations, Maintenance, Severe Accident Mitigation and Accident Analysis

U.S. DOE Lab and EPRI insights for SAWA/SAWM and the insights associated with the control of the Spent Fuel Pool have been the most beneficial

EPG/SAGs and TSG will be revised as new insights are identified:

- EPG/SAG changes are expected to occur through an issue process which documents the insights/lessons learned and the proposed changes
- A significant revision in the next 5 years is not expected (e.g. no significant update before 2025)

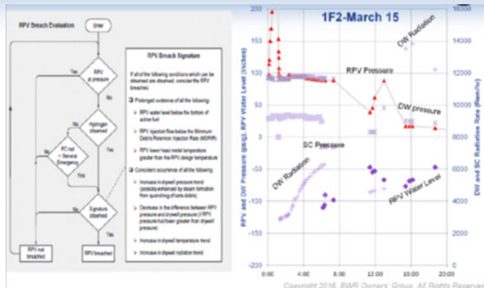


Copyright 2019, BWR Owners' Group, All Rights Reserved

Computer Based Training (CBT) for Severe Accident Management Training

BWR SAMG CBT available on NANTeL to the US Fleet - September 2020:

- The PWROG has shared some of their insights / lessons learned from implementing SAMG CBT.



Severe Accident information developed from national / international programs along with "***added value***" insights obtained from the U.S. DOE National Laboratories is being shared with leadership in the commercial industry:

- Decision Makers and Evaluators
- Licensed & Non-Licensed Implementers

Copyright 2019, BWR Owners' Group, All Rights Reserved

Instrumentation Practical Insights (CBT)

Understanding and the correct modeling of severe accident instrumentation response is vital:

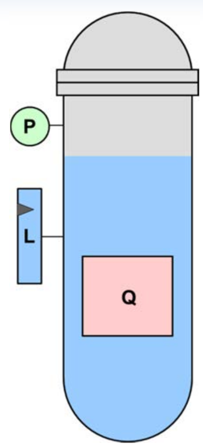
- *Lessons learned from Daiichi (1F1-1F3) and Daiini*

Decisions need to be made considering the plant's status; the plant status requires an understanding of the expected influence of the plant's control parameters:

- *Lessons learned from TMI-2, and to lesser degree, Chernobyl and 1F1-1F3*

TSG Toolkit calculations are used to and help validate instrumentation indications:

- *Calculations developed based on Lessons Learned*



Copyright 2019, BWR Owners' Group, All Rights Reserved

C.2.5.2. PWROG Procedures Update

Westinghouse Non-Proprietary Class 3

© 2020 Westinghouse Electric Company LLC. All Rights Reserved.

PWROG
PWR Owners Group

Global Expertise • One Voice

Kyle Shearer – PWROG Procedures Update

Reactor Safety Technology Experts Panel Forensics Meeting / November 2019

*** This record was final approved on 1/30/2020 9:14:58 PM. (This statement was added by the PRIME system upon its validation)

Westinghouse Non-Proprietary Class 3

© 2020 Westinghouse Electric Company LLC. All Rights Reserved.

Topics

- Severe Accident Management Guideline Maintenance Program
- Risk Beneficial Procedure Changes Program
- Long Term Containment Venting Study

*** This record was final approved on 1/30/2020 9:14:58 PM. (This statement was added by the PRIME system upon its validation)

Westinghouse Non-Proprietary Class 3
© 2020 Westinghouse Electric Company LLC. All Rights Reserved.

PWROG
PWR Owners Group

SAMG Maintenance Program

- SAMG Maintenance Program receives, evaluates, categorizes, and dispositions SAMG feedback to support docketed NRC commitment of site implementation of PWROG SAMG
- All members may provide feedback via SDW Requests
- SAMG Maintenance Core Group prioritizes feedback, endorses consensus responses for PSC approval

Reactor Safety Technology Experts Panel Forensics Meeting

*** This record was final approved on 1/30/2020 9:14:58 PM. (This statement was added by the PRIME system upon its validation)

Westinghouse Non-Proprietary Class 3
© 2020 Westinghouse Electric Company LLC. All Rights Reserved.

PWROG
PWR Owners Group

SAMG Maintenance Program

- SAMG Maintenance Program covers:
  - PWROG SAMG (PWROG-15015-P)
  - International SAMG (PWROG-16059-P)
  - SECURE Software (PWROG-17005-P & SP/CA modules)
  - Computer-Based Training (CBT) Material (PWROG-18032-P)

Reactor Safety Technology Experts Panel Forensics Meeting

*** This record was final approved on 1/30/2020 9:14:58 PM. (This statement was added by the PRIME system upon its validation)

Westinghouse Non-Proprietary Class 3

© 2020 Westinghouse Electric Company LLC. All Rights Reserved.

PWROG
PWR Owners Group

SDW Resolution Process

1. Core Group Accepts feedback for work
2. Resolution drafted by SAMG engineers
3. Core Group and SAMG engineers discuss and revise resolution as needed
4. Core Group Endorses consensus response
5. PSC Approves final resolution
6. Changes made to "Plus" Revision of generic SAMG materials as needed

*** This record was final approved on 1/30/2020 9:14:58 PM. (This statement was added by the PRIME system upon its validation)

Westinghouse Non-Proprietary Class 3

© 2020 Westinghouse Electric Company LLC. All Rights Reserved.

PWROG
PWR Owners Group

Risk Beneficial Procedure Changes

- Provide risk informed insights for EOPs and FSGs using PRA
  - Use of FLEX equipment in EOPs to provide additional defense in depth
  - Identify operator actions that have a high contribution to CDF and LERF
  - Identify best practices to improve interface between the site PRA staff and operations

Reactor Safety Technology Experts Panel Forensics Meeting

*** This record was final approved on 1/30/2020 9:14:58 PM. (This statement was added by the PRIME system upon its validation)

Westinghouse Non-Proprietary Class 3

© 2020 Westinghouse Electric Company LLC. All Rights Reserved.

PWROG
PWR Owners Group

Long Term Containment Venting

- Program to develop recommendations for strategies to mitigate concerns with flammable gases in long term
- International PWRs are fitted with PARs to control hydrogen levels during postulated severe accidents
- If the severe accident progression is predicted to lead to prolonged MCCI, hydrogen may continue to be produced after the PARs have depleted the initially available oxygen

*** This record was final approved on 1/30/2020 9:14:58 PM. (This statement was added by the PRIME system upon its validation)

Westinghouse Non-Proprietary Class 3

© 2020 Westinghouse Electric Company LLC. All Rights Reserved.

PWROG
PWR Owners Group

Long Term Containment Venting

- Program to investigate the following strategies:
  - Low Efficiency Recombination – Inject oxygen
  - High Efficiency Recombination – Inject non-condensables & oxygen
  - Containment Venting Using CFVS
  - Containment Venting Using Non-Filtered CVS

Reactor Safety Technology Experts Panel Forensics Meeting

*** This record was final approved on 1/30/2020 9:14:58 PM. (This statement was added by the PRIME system upon its validation)

© 2020 Westinghouse Electric Company LLC. All Rights Reserved.

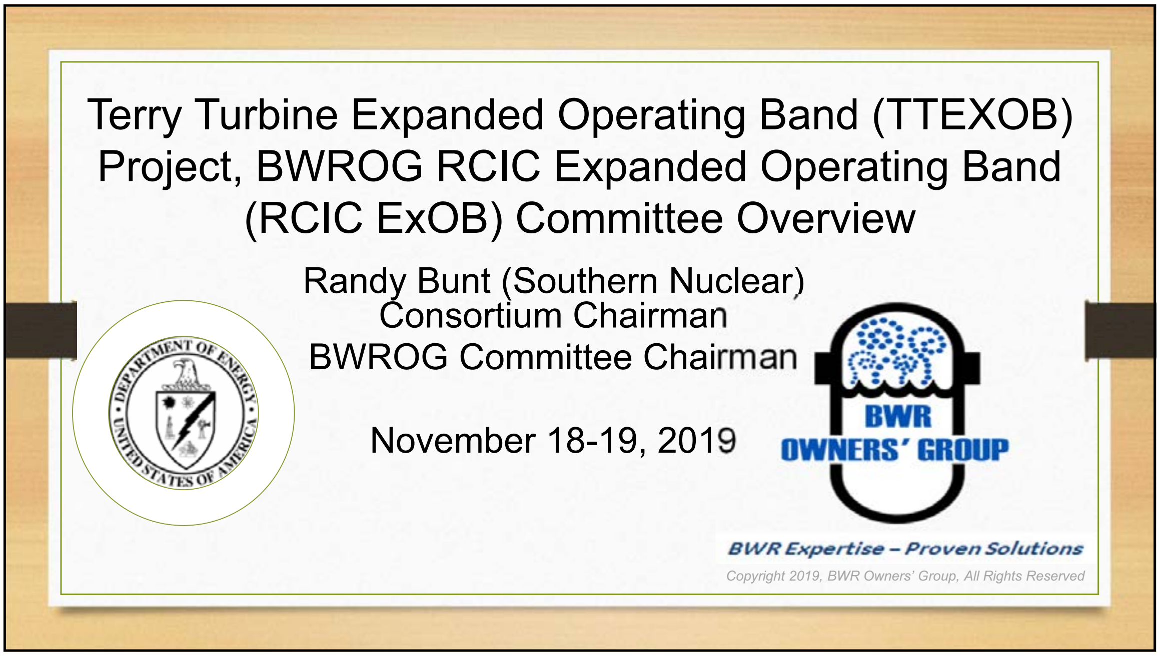
PWROG
PWR Owners Group

Conclusion

- The SAMG Maintenance Program, Risk Beneficial Procedure Changes Program, and Long Term Containment Venting Study are examples of how the PWROG is continuing to study and enhance accident management strategies

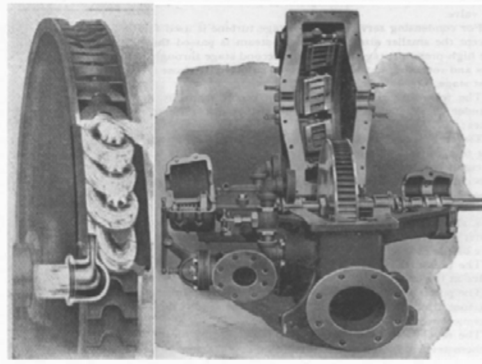
*** This record was final approved on 1/30/2020 9:14:58 PM. (This statement was added by the PRIME system upon its validation)

C.2.5.3. Terry™ Turbine Expanded Operating Band Project - BWROG



Terry Turbine Testing Overview

- The TTEXOB project uses a step-wise approach to expand and define the actual operating project elements (within the Summary Plan Milestones) to include plan development, first principle analytical modeling, prototype testing & modeling, small scale testing & modeling, and large scale testing & modeling. The plan is described within the Project Detailed Test Plans and the Project Summary Plan which provides the structure and basis for the Experimental Test Procedures, Goals, and Deliverables. The Project Charter provides the structure for the Consortium (Turbo-TAG), Pooled Inventory Management (Terry Turbine ExOB Equipment Committee) and BWROG (RCIC ExOB Committee) groups' interaction.



3

Terry Turbine Testing Overview (cont)

- The goal of the International Consortium is to provide long-term oversight of the TTEXOB. The TTEXOB Project goal is to expand and define the actual operating limitations (margins) of the Terry turbine systems (i.e. RCIC/TDAFW) used in the nuclear industry.
- Membership for the project is based on the Turbo-TAG and as identified in the Program Plan (SAND2017-5562). Additional details on *Project* structure and participation are included in the Summary Project Plan (SAND2017-1725). The US Nuclear Industry, US DOE, and IAE (Japan), are the major stakeholders of the TTEXOB Committee and, as such, have leadership roles in the Turbo-TAG as well. Additional members of the *Consortium* would be identified and approved by the Turbo-TAG. The TTEXOB Project Manager will control the membership list and make changes as directed by the Turbo-TAG.
- The overall experimental program support (e.g. equipment, personnel, technical output) will be equitably shared between the major funding stakeholders (Japan, US DOE, and US Industry), but will vary based on milestone content. Cost sharing will vary based on directives of the major funding stakeholders.

4

Value of Extended Performance

- Reduce and Deter Costs
  - Provide improved transition to portable FLEX equipment
  - Deferring the use of raw water can save plants many dollars
- Reduce Risk of Operations
  - Update emergency operating procedures (EOPS)
  - Establish technical basis for operational changes that prevent progression to core damage and reduce core damage frequency
- Simplify Plant Operations
  - Add flexibility to respond to event conditions identified in the Fukushima accidents
  - Increased time available for implementation of FLEX

5

Milestone Summary

Milestone 1-2: Principles & Phenomenology
- Scoping and limited modeling efforts complete

Milestone 3: Full-Scale Separate Effects Component Experiments
- Testing at Texas A&M University (TAMU)

Milestone 4: Terry Turbopump Basic Science Experiments
- Testing at TAMU

Milestone 5: Integral Full-Scale Experiments for Long-Term Low Pressure Operations
- Test facility evaluation in progress

Milestone 6: Integrated Full-Scale or Small-Scale Experiments Replicating 1F2 Self-Regulating Feedback
- Scoping and Cost Estimate to be performed

Milestone 7: Collection of Milestone Information for Code Updates and Project Closeout
- Integral with milestone work to be completed after last approved milestone

6

Milestone 5 Low Pressure Steam Testing

Assumptions
4 Tests lasting 10 days for 24-hour support
40 days of 1-day testing
4 days of delays on 24/7 testing
10 days of delays on 1-day testing

- Establish a high pressure (300-800 psia) steady state to emulate normal operations
- Transition from high pressure to minimum operating pressure over a set period of time
- 100-150 psia minimum operating pressure,
- Turbine back pressure variations of 40-60 psia
- 10 days of continuous operation for each experimental run, (~1.5 times 7 days)
- Periodic oil testing, and
- Continuous vibrational monitoring.

Weeks**	Schedule
8	Procedures and General Layout
14	Setup for Testing (material purchase and calibration)
6	Setup for Testing (mechanical setup after receipt of equipment)
20	Testing
8	Final Report
** Note: Some efforts are expected to have overlay. It is assumed 6 weeks of overlap would occur for this work.	
50	Total

7

Milestone 5 Low Pressure Steam Testing

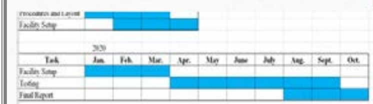
Table 2.1 Long-Term Low-Pressure Test Matrix

Inlet pressure	40 psi backpressure	60 psi backpressure
800-300 psi, saturated steam	2 hours at 3000 rpm	2 hours at 3000 rpm
300 to 150 psi, saturated steam	1-2 hour transient at 3000 rpm	1-2 hour transient at 3000 rpm
150 psi, saturated steam	10 days at 3000 rpm	10 days at 3000 rpm
100 psi, saturated steam	1 hour at 2000+ rpm	1 hour at 2000+ rpm

8

Milestone 5 Overall Timeline

- Milestone 5 Detailed Test Plan to Test Facility for quote
- Specify terms and conditions
- Generic contract for line items
- Finalize contract with Test Facility
- Contract issuance
- Initial payment of non-recurring program costs
- Finalize Milestone 5 Detailed Test Plan
- Finalize *Japan specified testing scope* (Appendix in detailed test plan)
- Payment of remainder of non-recurring program costs
- Test Facility acceptance of detailed test plan



9

Expected Major Activities / Three Year Plan

2019:

- Independent Program Peer Review
- Complete full-scale component testing
- Complete Basic Science Integral Testing
- Develop scaling algorithm for Z-1 to G-1 and Air-Water to Steam-Water

2020:

- Complete low pressure long-term full-scale testing
- Conduct RCIC self regulation (analysis & testing)
- Fleet-wide implementation of improvements

2021:

- Milestone 3, 4 & 5 Report
- Milestone 6 High Pressure Long-Term Small-scale Testing

10

C.2.5.4. Terry^{TM} Turbine Expanded Operating Band Project - SNL

Exceptional service in the national interest

Sandia National Laboratories

Using Terry Turbine Systems for Enhancing Plant Resilience

Nathan Andrews, Lindsay Gilkey, Matthew Solom

Collaborators: INL, TAMU, IEA, BWROG

Sandia National Laboratories is a multimission laboratory managed and operated by National Technology and Engineering Solutions of Sandia, LLC, a wholly owned subsidiary of Honeywell International, Inc., for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-NA0003525. 1

Project Overview

- **Primary Goal:** Understand real-world behavior of Terry turbopumps under BDBE conditions to advance predictive fidelity and applicability in emergency and accident prevention and mitigation.
- The purpose of this research is to further develop a dynamic and mechanistic system-level model of the RCIC/TDAFW turbine/pump system capable of predicting the system performance under BDBE conditions that include two-phase water ingestion into the Terry turbine at various potential reactor operating pressures, and to characterize its ability (or not) to maintain adequate water injection with sufficient pump head under degraded operating conditions.
- The scaled and full-scale Terry turbopump experiments and modeling will support an improved understanding of plant risk, improve plant operations, and provide the technical basis for improving the reliability of an essential plant system:
  - **Regulatory/Risk:** Test data can reduce operational risk and improve regulatory compliance
  - **System:** Improve reliability; essential system to mitigate risk dominated accidents
  - **Operations:** Improvement during an BDBE, mitigating the accident under varying conditions

Terry Turbopump Modeling

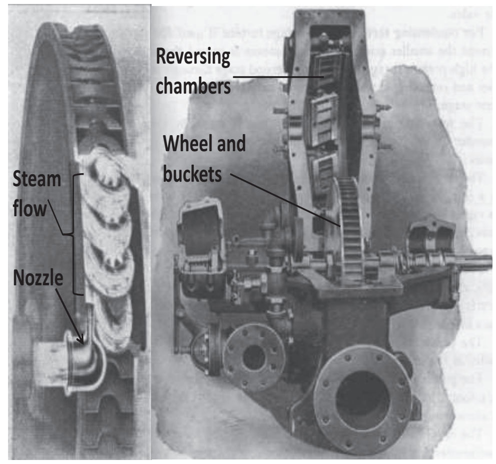
Terry turbine is a small, single-stage, impulse turbine.

Terry turbines were principally designed for waste-steam applications with the following key attributes:

1. The turbine and casing are not pressurized out of necessity: it may be at low or even atmospheric pressure;
2. Rapid startup (less than 60 s) is of primary importance;
3. Reliability, resilience under off-nominal conditions, and low maintenance are of primary importance;
4. Efficiency is of secondary importance.

In contrast to more typical turbines (large, complex, high-pressure, high efficiency).

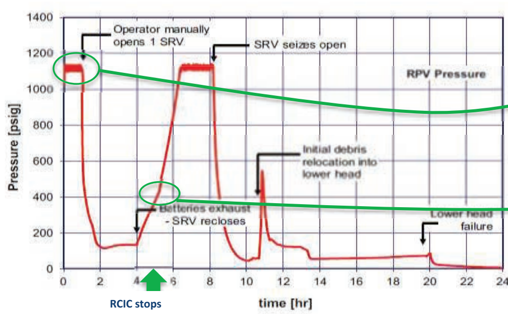
Additionally: Nozzles are detached from turbine and stationary.



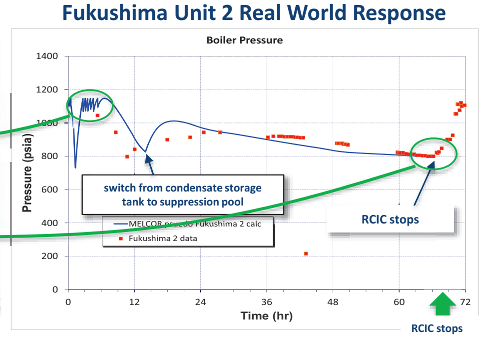
Modeling of SBO before and after Fukushima

Reactor core in isolation cooling (RCIC) system performance during beyond design basis event conditions was poorly known.

Pre-Fukushima Understanding (NRC SOARCA)



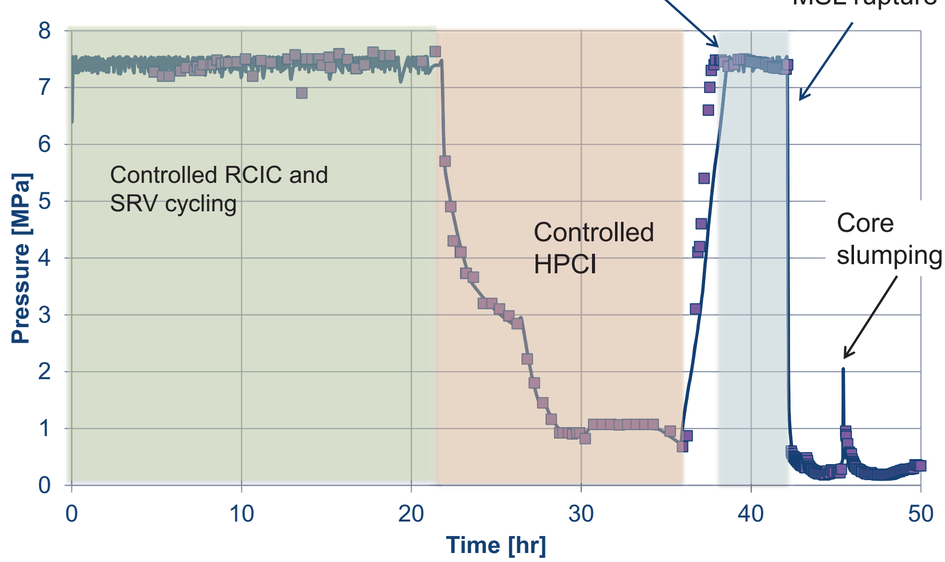
Fukushima Unit 2 Real World Response



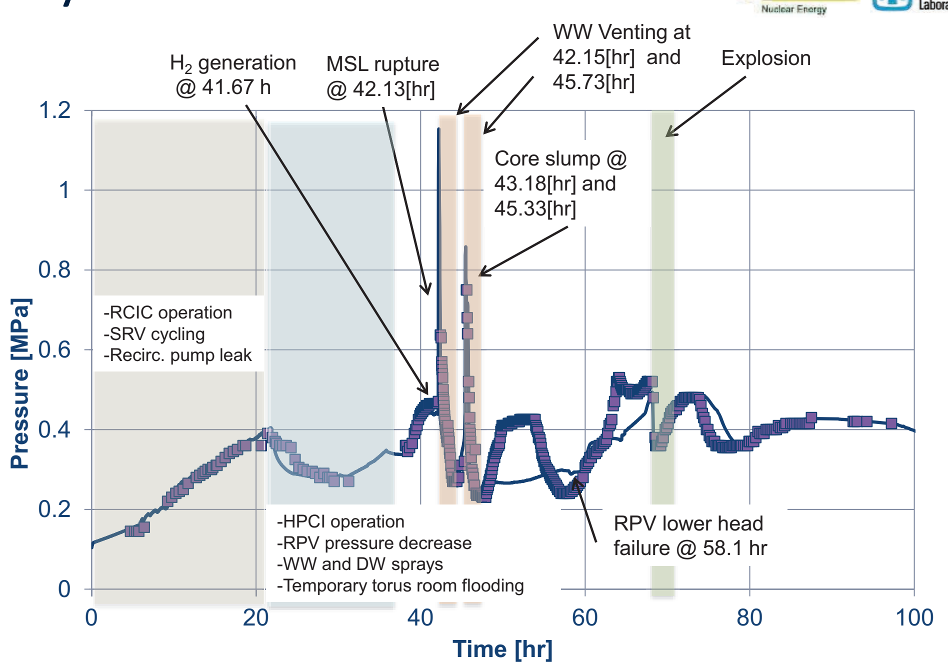
- Turbine-driven RCIC injection maintains desired water level in reactor pressure vessel (RPV)
- Battery depleted @ 4 hours
  - *SRV closes and RCIC runs full on*
  - *RPV overfills, MSL floods, water enters RCIC turbine, and RCIC assumed to fail*
- Core meltdown at 10 hours

- Turbine-driven RCIC injection maintains desired water level in RPV at start of event
- Batteries fail @ 45 minutes from tsunami flooding
  - *RPV overfills, MSL floods, water enters RCIC turbine, but RCIC turbine <u>does not fail</u>*
  - *RCIC self-regulates RPV water level in cyclic mode*
- Core damage avoided for nearly 3 days

Reactor Pressure Vessel Pressure – 1F3

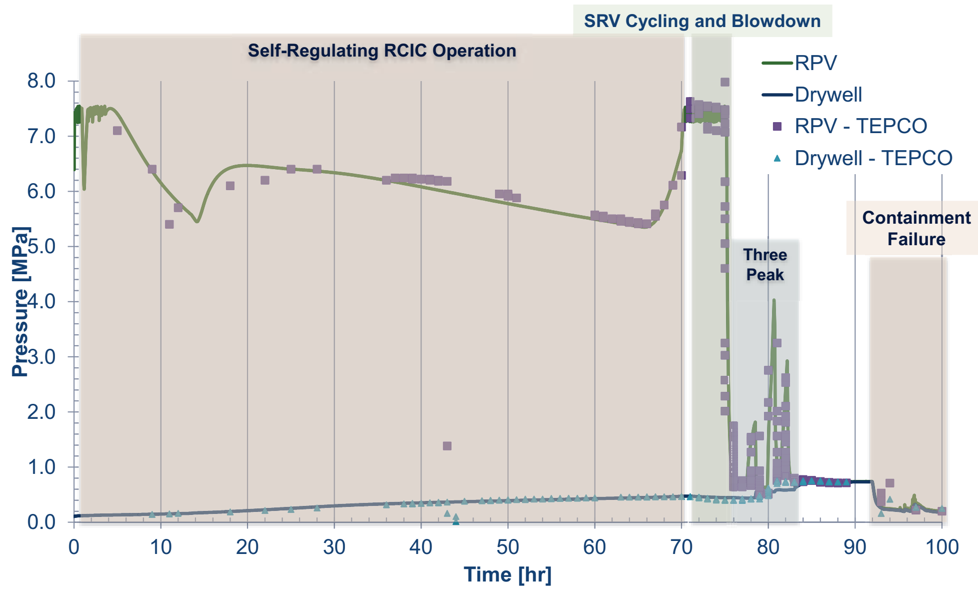


Drywell Pressure – 1F3



6

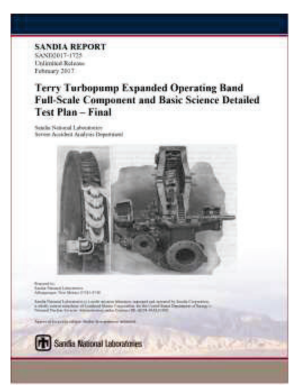
System Pressures -1F2



7

Value Proposition of Extended Operating Band

- <u>Reduce and Defer Costs</u>
- Extends the intervals between preventive maintenance periods
- Provides improved transition to portable FLEX equipment
  - Deferring the use of ultimate FLEX measures using raw water at one BWR plant saves ~$450M
- <u>Reduce Risk of Operations</u>
- Update emergency operating procedures (EOPs)
- Establish technical basis for operational changes that prevent progression to core damage and reduce core damage frequency
- <u>Simplify Plant Operations</u>
- Add flexibility to respond to event conditions identified in the Fukushima accidents
- Increased time available for implementation of FLEX


SANDIA REPORT

Terry Turbopump Expanded Operating Band Full-Scale Component and Basic Science Detailed Test Plan – Final

Sandia National Laboratories

8

Milestone-based Approach

- Milestone 1 – Plan development
- Milestone 2 – First principle analytic modeling
  - CFD and initial MELCOR/SAMPSON models
- Milestone 3 – Prototype and component testing
  - Governor valve, trip/throttle valve, lube oil, bearing degradation
- Milestone 4 – Small-scale and shakedown testing
  - Air/water and steam/water ZS-1 testing, air/water GS-2 testing
- Milestone 5 – Large-scale testing
  - Steam/water GS-2 testing
- Milestone 6 – Self-regulation testing
  - Currently postponed
- Milestone 7 – Project closeout

9

EXPERIMENTATION PROGRAM

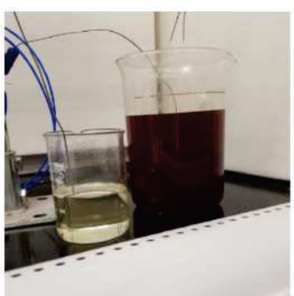
10

Key Experimental Activities

- Lube-oil
  - Degradation evaluation at elevated temperatures
  - Degraded oil will be used in future GS-2 tests
- Trip-throttle valve
  - Flow coefficient evaluation
- Governor valve
  - Flow coefficient evaluation
- ZS-1 “small one”
  - Air/water shake-down tests
  - Steam/water tests to build a SA code model for use to be validated against GS-2 tests
- GS-2 “big one”
  - Air/water shake-down tests
  - Steam/water tests next FY at representative temperatures and pressures

11

Lube Oil Testing

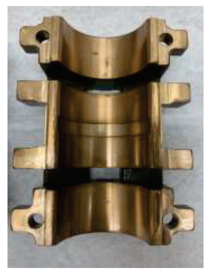


- Key tests:
  - 325 F for 72 hours
  - 250 F for 72 hours
  - 200 F with 10% water for 6 hours
- Oil changes at 325F were remarkably more significant than at 250F.
- Precipitates in the 325F test are measurable.
- As expected, viscosity changes drastically as the oil heats up from room temperature.
- There are noticeable differences between the viscosity-temperature spectrum of fresh and degraded oil.
  - Viscosity of degraded oil decreases faster than that of fresh oil.
  - The spectrum for degraded oil is a lot more smooth.

12

Bearing Testing

- ZS-1 "The Duke"
  - 18-inch wheel, 3600 rpm
- GS-2 from Clinton
  - To be swapped in future
- Conclusions:
  - Capable of maintaining speed for 1000 rpm and 3000 rpm tests
  - No turbine wear noticed at 6 hrs with fresh oil
  - Characteristics of speed and torque vs. time shown



Valve Testing

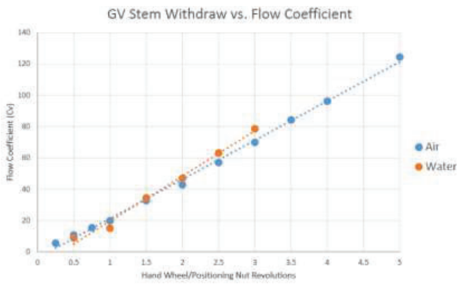
Governor Valve

Trip Throttle Valve

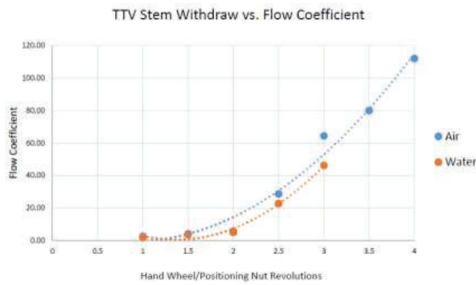
14

Valve Testing

Governor Valve



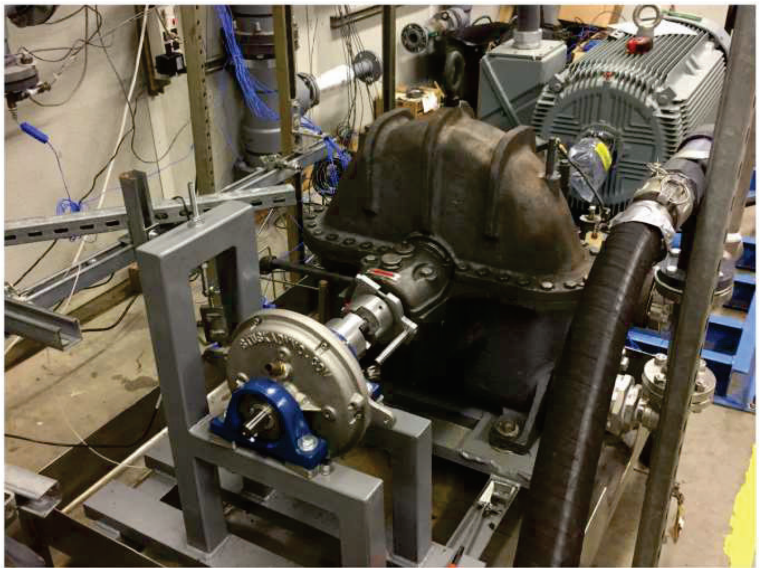
Trip Throttle Valve



- TTV showed bi-linear correlation between handwheel revolutions and flow coefficient
  - Inflection point around 2 revolutions
  - Flow coefficient tested using air was always higher than when using water
- GV showed single linear correlation between positioning nut revolutions and flow coefficient
  - Flow coefficient higher with air below 1 revolution
  - Flow coefficient higher with water above 1 revolution

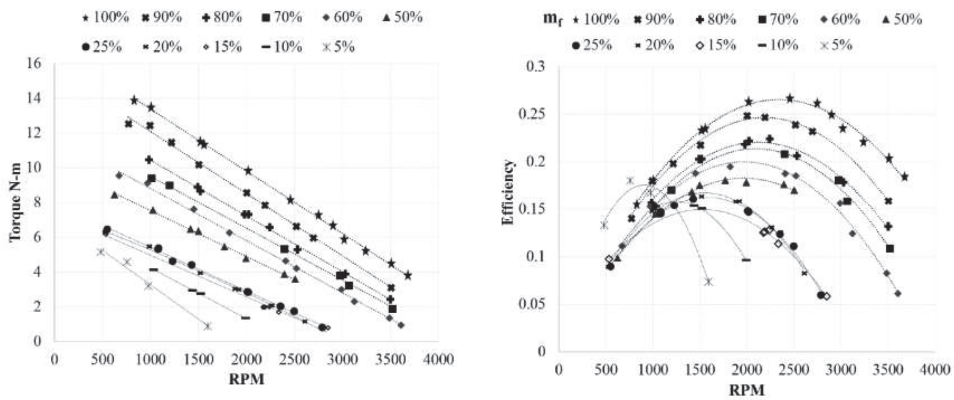
15

ZS-1 Testing



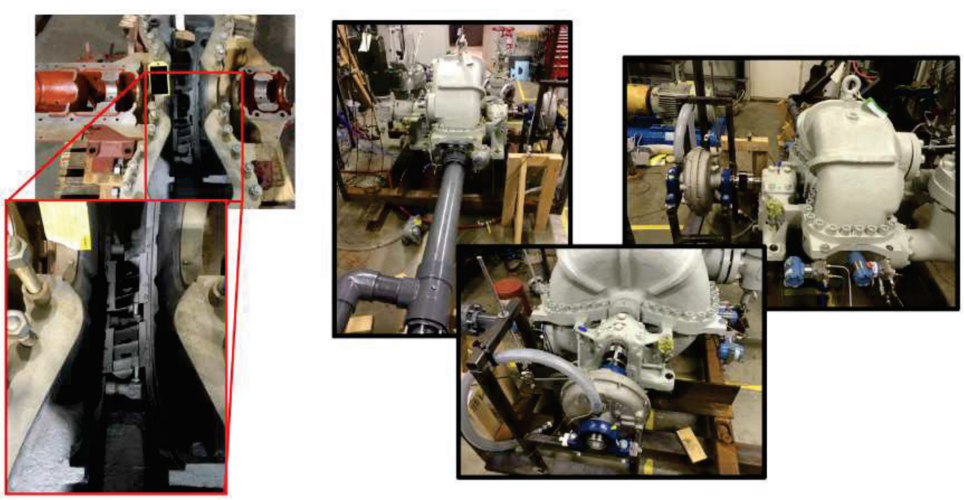
16

ZS-1 Results



17

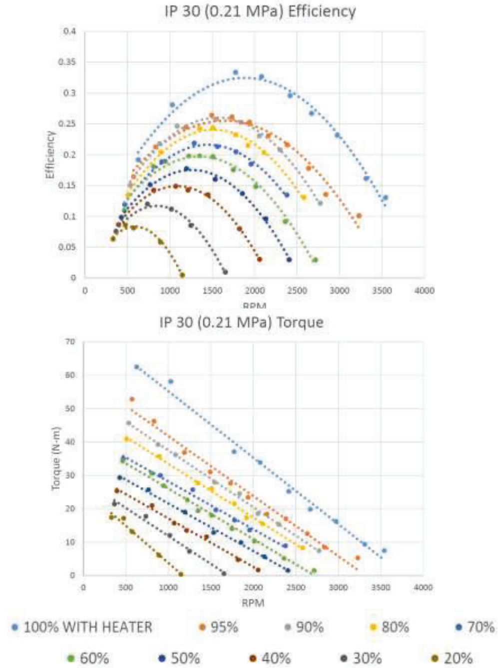
GS-2 Testing



18

GS-2 Testing Results

- Test Ranges
  - 20 to 70 psia (0.14 to 0.517 MPa)
  - 5% to 100% air mass fraction
  - Below 4000 RPM
- Dynamometer and needle control valve to regulate turbine speed
- Air heater implemented to warm exit temperature above freezing due to expansion cooling

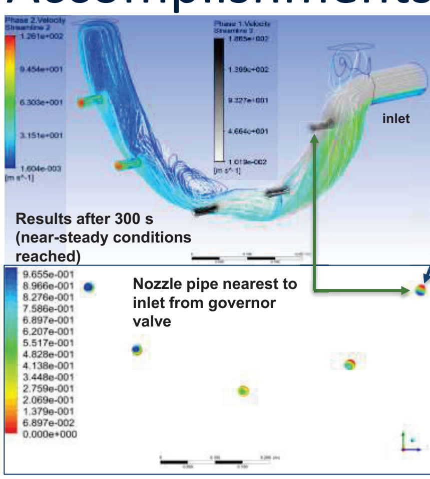


Terry Turbine Milestone 5: Long-Term Low-Pressure Experiments

- Intended to define the true operating limitations (margins) of Terry turbo-pump systems used in the nuclear industry.
- Based on the Fukushima experience, the hypothesis is that the Terry turbine has the capability to operate long-term (days) over:
  - Extended range of pressures (75-1205 psig; design range is 150 psig to lowest SRV set point),
  - Varied steam quality (100% to 0%; current assumption is 100%)
  - Increased lube oil temperature conditions (215-300 F; current limit is 160 F).
- Milestones 3-4 testing have addressed all these elements at reduced scale.
  - Milestone 5 testing is intended to verify long-term operations at low pressure with a full scale GS-2 turbine.
- Large steam supply (up to ~10 MW) needed for this type of testing; three facilities considered (two visited) to evaluate testing feasibility.
- Best candidate has been found. Cost-schedule estimates have been developed, and discussions are underway through the US Industry-Japan-DOE consortium on moving forward with this work in FY20.

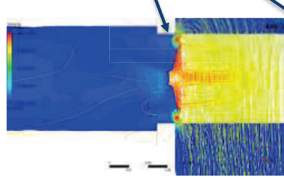
MODELING EFFORTS

Accomplishments (Modeling)

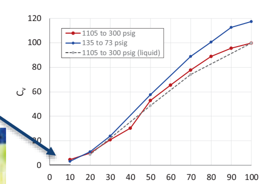


Liquid vol. fraction flowing to nozzles

- Sandia modeling using CFD and system level modeling
  - Pre-experimental modeling
  - Identified potential areas of experimental uncertainties
  - Provides justification of Milestone 3 & 4 experimental efforts



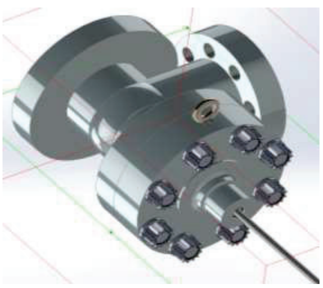
Close-up view of velocity streamlines for 0.1 open fraction, above valve mid-plane

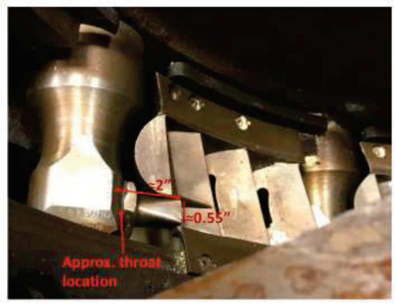


Governor valve C_v vs. valve position

Terry Turbopump CFD

- CFD analysis was performed detailed CAD models. Analysis was performed in Fluent.
- CFD analysis was on the governor valve and nozzles. Focus given to nozzle CFD, as it was used to inform the MELCOR modeling of the ZS-1 and GS-2.
- CFD was also used to identify areas of interest for experimentation.

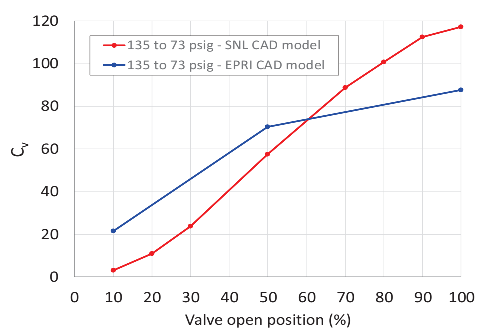


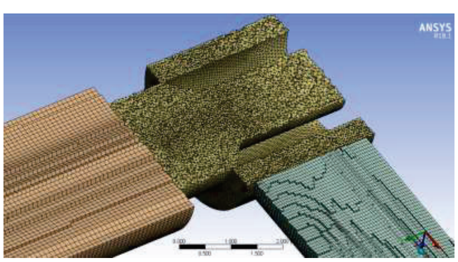


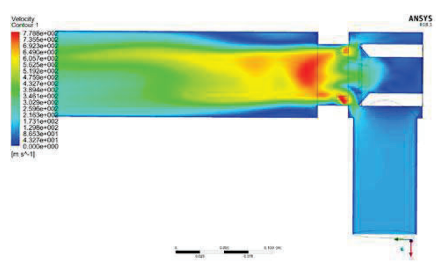
23

Terry Turbopump CFD: Governor Valve

- Flow losses
- Flow characteristics for use in MELCOR and other intergral codes



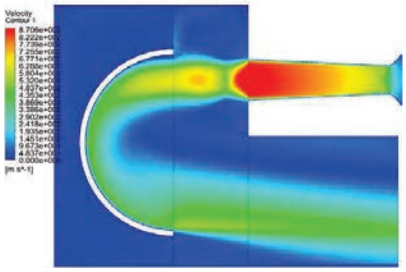




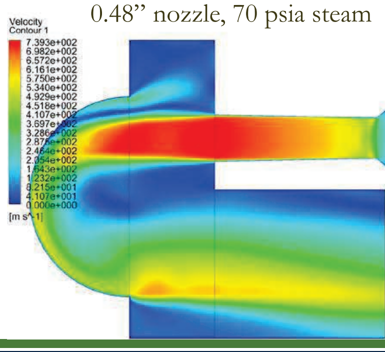
24

Terry Turbopump CFD: Nozzles

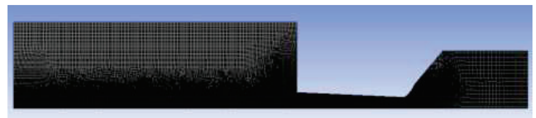
0.38” nozzle, 70 psia steam



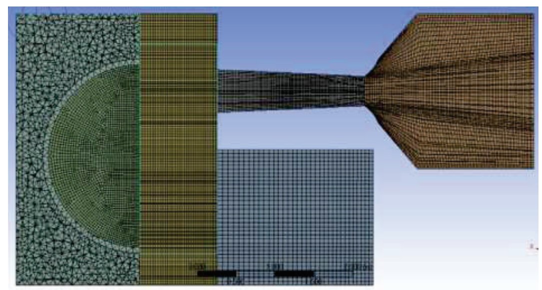
0.48” nozzle, 70 psia steam



2D Mesh



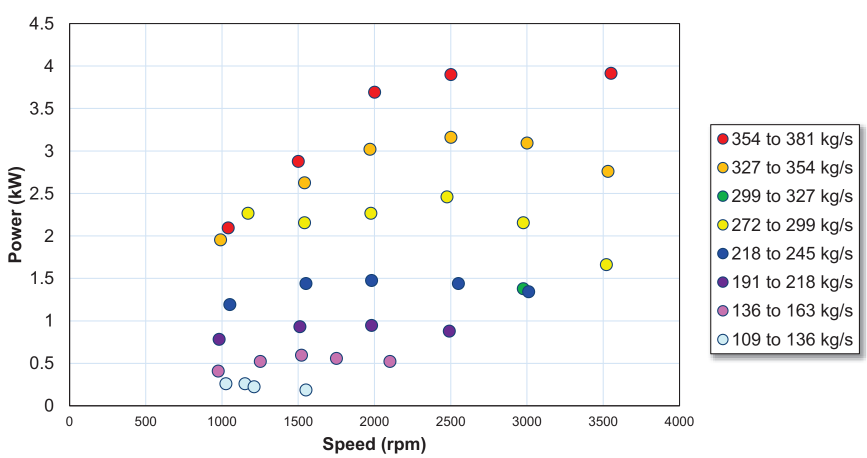
3D Mesh



25

ZS-1 Measured Turbine Power vs Airflow

Power vs Speed (colored by airflow)

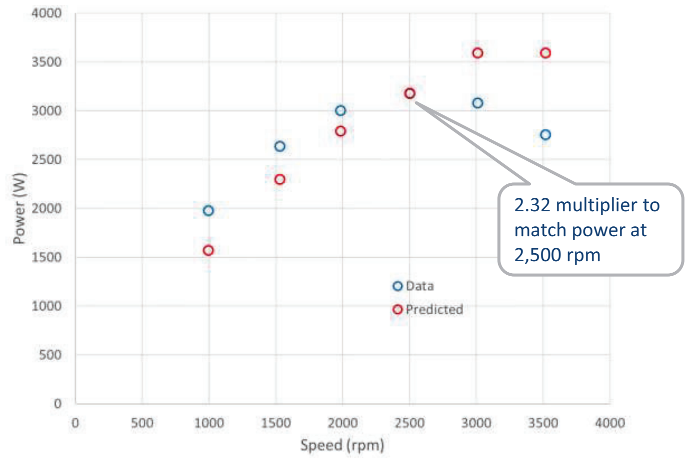


Modeling and experiments revealed same powers can be produced by different speeds / pressures at a given airflow.

Constant pressure trends identify a most efficient speed for the ZS-1 of 2,500 rpm.

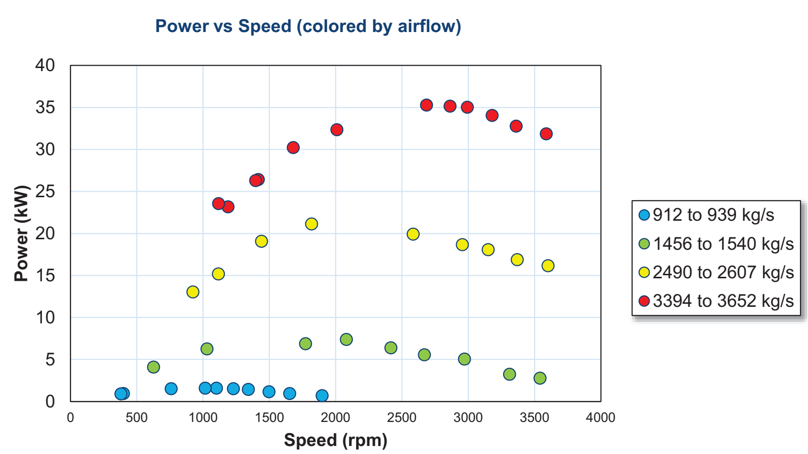
26

Measured and Predicted ZS-1 Power at Inlet Pressure of 90 psia and Differing Speeds



Multiplier on predicted torque of 2.32 used to match power at 2,500 rpm, 2.32 multiplier applied in all predictions.

GS-2 Turbine Power vs Airflow



Only single-phase (100% air) flow is considered here.

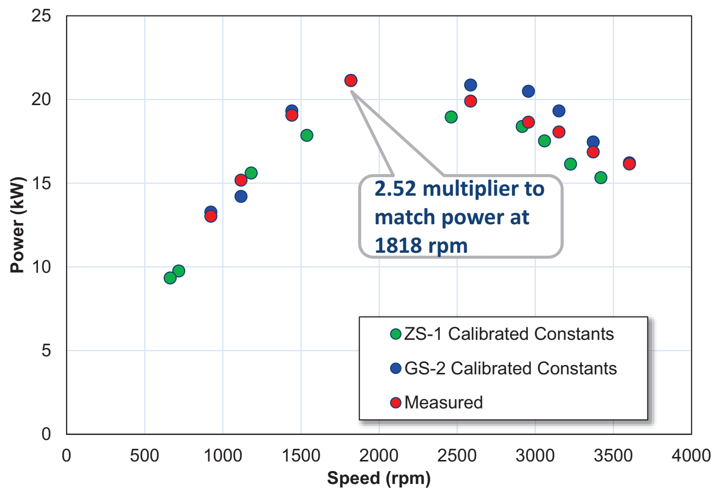
Testing is at higher airflow rates and power than for ZS-1.

Experiments reveal same powers can be produced by different speeds / pressures at a given airflow.

Constant pressure trends identify a most efficient speed for the GS-2 of between 1,000 to 2,700 rpm.

28

Measured and Predicted GS-2 Power at Inlet Pressure of 50 psia and Differing Speeds



Multiplier on predicted torque of 2.52 used to match power at 1818 rpm, 2.52 multiplier applied in all predictions.

How does multiplier on GS-2 compare to multiplier on ZS-1?

- Preliminary Results, Modeling Air Tests: 2.32 (ZS-1) vs 2.52 (GS-2)

Conclusions

- Improved understanding of Terry turbopumps can be obtained from a combined effort of modeling and full-scale experimental testing.
- CFD was able to inform experimentation and modeling.
- MELCOR models and experiments identify that the same power can be achievable for different airflows and speeds.
- Our MELCOR model is matching trends in net power and speed, however there are some issues we will be looking into resolving to improve predictions.

30

Future Activities

FY-19

- Complete Full-Scale Component Experiments (Milestone 3)
- Complete Terry Turbopump Basic Science Experiments (Milestone 4)
- Complete Milestone 3 & 4 modeling
- Initial Scoping and estimates of Milestone 5
  - Integral Full-Scale Experiments for Long-Term Low Pressure Operations

FY-20

- Finalize plan and start Milestone 5 experiments
- Modeling to support Milestone 5 experiments

FY-21

- Complete Milestone 5 experiments and modeling
- Complete Milestone 7 and closeout project

QUESTION OR COMMENTS?

32

C.3. Other Presentations

C.3.1. U.S. DOE

U.S. DEPARTMENT OF ENERGY | Nuclear Energy

US Efforts to Support Examinations at Fukushima Daiichi – Expert Panel Meeting

Joy Rempe
Technical Lead, Rempe and Associates, LLC

Damian Peko
DOE Program Manager, US Department of Energy

Mitch Farmer
ANL Program Manager, Argonne National Laboratory

ANL Offices, Washington, DC
November 18-19, 2019

Program Overview

U.S. DEPARTMENT OF ENERGY
Nuclear Energy

Forensics Efforts Offer US Perspective to Fukushima Daiichi Examination Activities

Objectives:

- Develop consensus US input for ***high priority examination tasks and supporting research*** that can be completed with ***minimal disruption of TEPCO D&D activities.***
- Evaluate obtained information to:
  - Gain a better understanding of events that occurred in each unit at Daiichi
  - Gain insights to reduce uncertainties in predicting phenomena and equipment performance during severe accidents
  - Provide insights beneficial to TEPCO Phase 2 Fuel Debris Retrieval Evaluations
  - Confirm/improve guidance for severe accident prevention, mitigation, and emergency planning
  - Update/refine original information requests.
- Facilitate implementation of Japan-led international research efforts to support D&D.

Graphic courtesy ANS

Motivations:

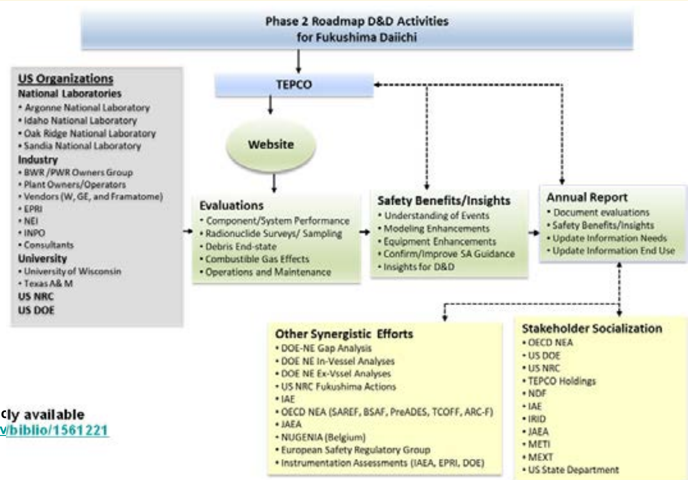
- Provides Japan access to US expertise in plant operations, severe accident modeling & testing, and defueling & cleanup.
- Provides US access to full-scale, prototypic data from multiple units with distinct accident signatures.

2

Program Overview

U.S. DEPARTMENT OF ENERGY
Nuclear Energy

U.S. Efforts Coordinated with Phase 2 Roadmap D&D Activities and Other Programs



- **FY2019 report publicly available** https://www.osti.gov/biblio/1561221

3

FY20 Activities

U.S. DEPARTMENT OF ENERGY
Nuclear Energy

Proposed Schedule for Action Items

- **Meeting Summary Letter Report**.
  - Contributor author release and graphic permissions releases- **November 30, 2019** – ***All***
  - Draft Letter Report to Contributors for Review - **December 15, 2019 – Rempe**
    - Meeting Summary
    - Updated Appendix C Information Requests
    - Slides from Meeting
  - Comments on Meeting Summary – **December 30, 2019 *– All***
  - Letter Report Issued – **January 31, 2020** - ***ANL***
- **Other Considerations (depending on funding availability):**
  - Information Briefs –
    - Unit 1 Inspections
    - Maintenance Practical Lesson Learned
    - Insights from International Activities (e.g., PreADES, ARC-F, etc.).

4

U.S. DEPARTMENT OF ENERGY
Nuclear Energy

Remaining Meeting Topics*

- **NDF**
  - **2019 Strategic Plan – Wakabayashi**
  - **Fundament Concepts for Debris Analysis – Nakano**
- **TEPCO –Update on TEPCO Activities- Mizokami**
- **JAEA – Update on PreADES – Nakayoshi (Rempe)**
- **US Applications / Needs from Forensics Information**
  - **MAAP: Henry**
  - **MELCOR- Luxat**
  - **Modeling Issues - Panel**
- **Topic Area Lead discussions:**
  - **Area 1 - Components/System Performance – Robb/Gabor/Andrews**
  - **Area 2 - Radionuclide Surveys/Sampling – Andrews/Luxat**
  - **Area 3 - Core Debris Location Evaluations – Farmer**
  - **Area 4 – Combustible Gas Effects - Luangdilok**
  - **Area 5 – Operations and Maintenance – Williamson//Klass/Shearer/Bunt**
- **Path Forward and Next Steps - All (led by Rempe)**

*Not listed in order, See Agenda for Presentation Schedule

Link to FY19 Report, Agenda, & Viewgraphs:
https://1drv.ms/u/s!ApliColj18LBgaUpEs8e-56sxcbtuA?e=E9P9Sf

5

C.3.2. U.S. NRC

International Collaborations in Severe Accident Research

Richard Lee
Office of Nuclear Regulatory Research

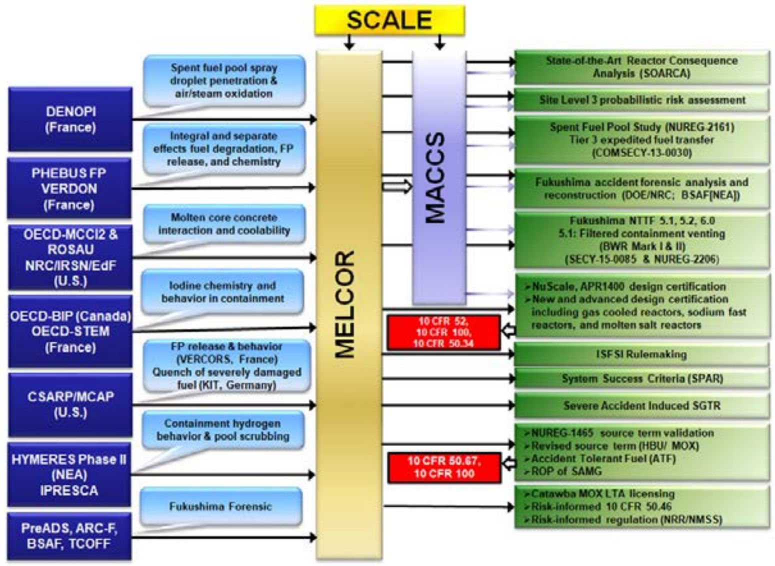
Reactor Safety Technology Expert Panel Forensics Meeting
Washington, DC
November 18, 2019

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NRC Severe Accident Collaborations

- NRC sponsors and collaborates in experimental research programs:
  - Develop an improved understanding of those phenomena that are important to reactor safety
  - Reduce residual uncertainties through a combination of experimental and analytical research activities
  - Develop and validate models in severe accident codes
- NRC supports international collaboration on severe accident and offsite consequence modeling through the Cooperative Severe Accident Research Program (CSARP)
  - 28 member nations focus on the analysis of severe accidents and offsite consequences using the MELCOR and MACCS codes
  - MELCOR and MACCS user group meetings to share experience with the NRC codes, identify code errors, perform code assessments, and identify areas for code improvements, experiments, and model development

Code Development, Regulatory Applications & International Collaboration



NRC Joint International Projects

- ***ARC-F (2019-2021)***: NEA/CSNI Analysis of Information from Reactor Building and Containment Vessel and Water Sampling in Fukushima Daiichi (ARC-F)
- ***BIP-3 (2016-2018)***: NEA/CSNI Behavior of Iodine Project (BIP) and follow-on BIP projects to provide separate effects and modelling studies of iodine behavior in a containment following a severe accident
- ***BSAF-1/2 (2012-2015, 2015-2018)***: NEA/CSNI Benchmark Study of the Accident at the Fukushima Daiichi (BSAF) project and the follow-on BASF-2 project to improve severe accident codes and to analyze the accident progression and current status of units 1 to 3 of the Fukushima Daiichi
- ***DENOPI (2018-2022)***: an IRSN project on separate effects experimental data on phenomena (cladding oxidation & spray effects) associated with a spent fuel pool loss-of-cooling and loss-of-coolant accidents.

NRC Joint International Projects

- ***ESTER (2020-2024)***: IRSN/CEA proposed experiments on Source Term for delayed Releases (ESTER)
- ***HYMERESII (2017-2020)***: NEA/CSNI Hydrogen Mitigation Experiments for Reactor Safety (HYMERES) project to improve the understanding of the hydrogen risk phenomenology in containment
- ***IPRESCA (2017-2020)***: EC NUGENIA/SARNET Joint international project, Integration of Pool Scrubbing Research to Enhance Source-term Calculations (IPRESCA), to promote a better integration of international research activities related to pool scrubbing

NRC Joint International Projects

- ***Phébus FP & Phébus-ISTP (1989-1996, 2005-2016)***: Joint international project provide integral and separate effects data on fission products release and behavior in the RCS and containment
- ***PreADES (2018-2020):*** NEA/CSNI Preparatory Study on Analysis of Fuel Debris (PreADES) project to provide a proposed list of information and data needs from damaged Fukushima Daiichi units
- ***QUENCH (1995-)***: Karlsruhe Institute of Technology (Germany) tests to examine and demonstrate the performance of fuel cladding under postulated accident conditions
- ***ROSAU (2020-2024)***: NEA/CSNI Reduction of Severe Accident Uncertainties (ROSAU) project to study molten-core concrete interaction and spreading and cooling

NRC Joint International Projects

- ***STEM2 (2016-2019)*****:** OECD/NEA Source Term Evaluation and Mitigation (STEM) project to improve the evaluation of source term and to reduce uncertainties on specific phenomena dealing with the chemistry of iodine and ruthenium
- ***TCOFF (2017-2019)*** : OECD/NEA Thermodynamic Characterization of Fuel Debris and Fission Products based on scenario analysis of severe accident progression at Fukushima Daiichi (TCOFF) project
- ***VERDON5 (2015-2017)***: Joint international project, VERDON 5 test and previous tests under the International Source Term Program (ISTP), to study fission product release from irradiated UO2 and MOX fuels in severe accident conditions

References

- NUREG/BR-0524, "Cooperative Severe Accident Research Program (CSARP)" (https://www.nrc.gov/reading-rm/doc-collections/nuregs/brochures/br0524/)
- SAREF: "Safety Research Opportunities Post-Fukushima - Initial Report of the Senior Expert Group," NEA/CSNI Report, February 2017 (https://www.oecd-nea.org/nsd/docs/2016/csni-r2016-19.pdf)
- BSAF-1: "Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant," NEA/CSNI/R(2015)18, March 2016 (http://www.oecd.org/officialdocuments/publicdisplaydocumentpdf/?cote=NEA/CSNI/R(2015)18&docLanguage=En)

C.3.3. MAAP Evaluations[*]

C.3.3.1. 1F2 Accident Progression

Fukushima Unit 2 Data Interpretations and Associated Implications

Chris Henry
Fauske & Associates, LLC (FAI)

Reactor Safety Technology Expert Panel Forensics Meeting
Washington, DC
November 18-19, 2019

© 2019 Electric Power Research Institute, Inc. All rights reserved.

Introduction

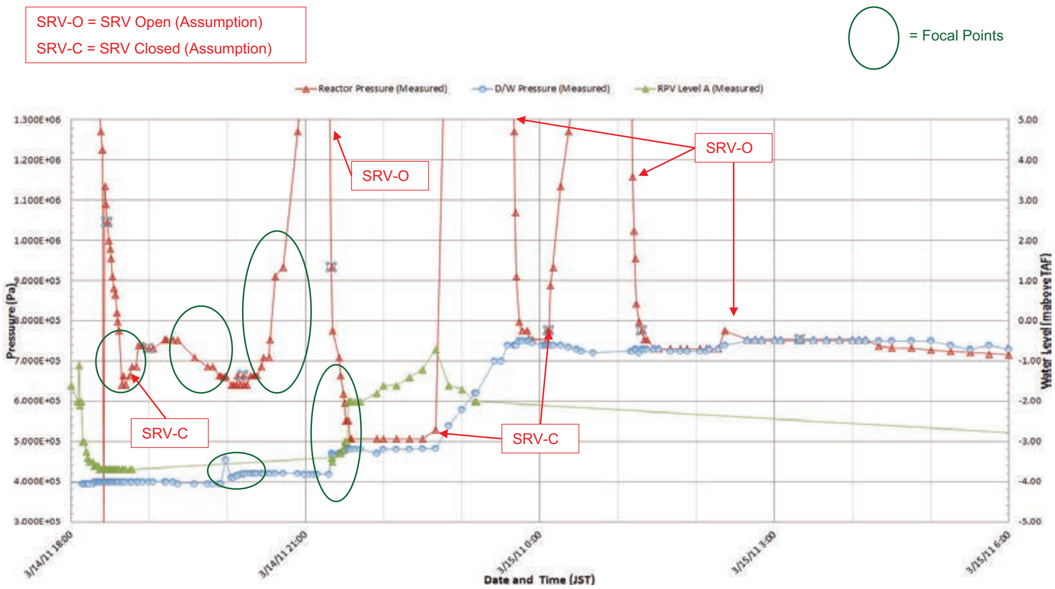
- Unit 2 (1F2) measured data for RPV water level, RPV pressure, and PCV D/W pressure are interpreted to determine major events and trends.
- The following items provide guidance in the noted interpretation:
  - Collateral forensic data (Example: Muon tomography) from 1F2, and also from 1F1 and 1F3.
  - TEPCO guidance from Project SMP-in-FACT investigation (10-12APR2013 meeting).
  - Existing industry technical bases for core and lower plenum, particularly TMI-2.
  - MAAP5 simulation of 1F2 for core, lower plenum, and primary containment (PCV).
- TEPCO guidance from the investigation of Unit 1 (1F1) RPV water level data was applied to 1F2 RPV water level interpretation, yielding significant findings.
- With synergies from all noted information sources, identification of major events and trends can be readily accomplished.
- Without the noted information sources, identification of major events and trends is extremely difficult.

2

© 2019 Electric Power Research Institute, Inc. All rights reserved.

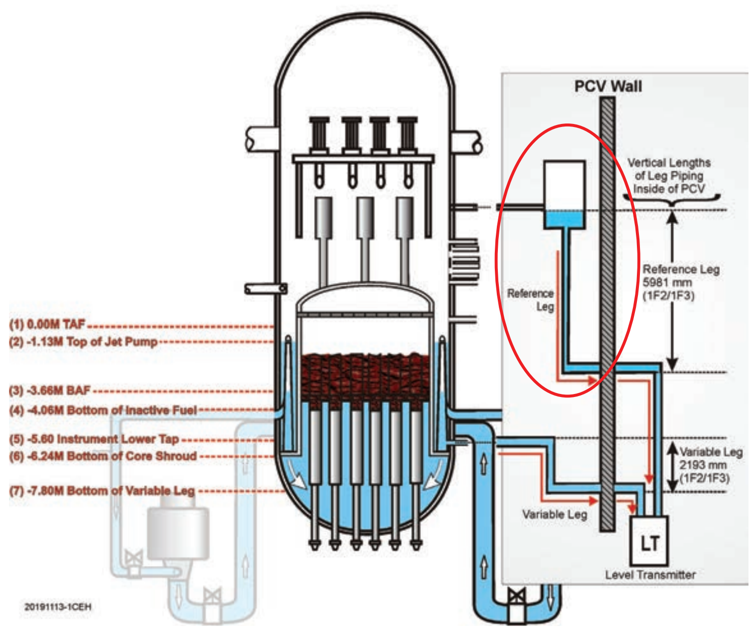
*. Data included in these presentations from TEPCO Holdings (14).

RPV Pressure , Water Level, PCV Pressure: Time Frame 3/14/2011 18:00 to 3/15/2011 06:00



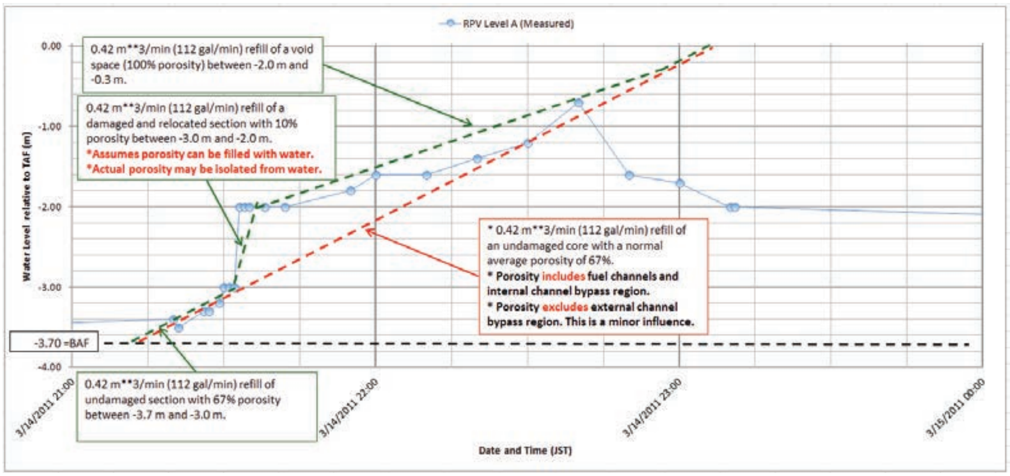
© 2019 Electric Power Research Institute, Inc. All rights reserved.
EPRI ELECTRIC POWER RESEARCH INSTITUTE

1F2 Reactor State: Time Frame 3/14/2011 20:00 to 21:20 RPV Water Level Near BAF



© 2019 Electric Power Research Institute, Inc. All rights reserved.
EPRI ELECTRIC POWER RESEARCH INSTITUTE

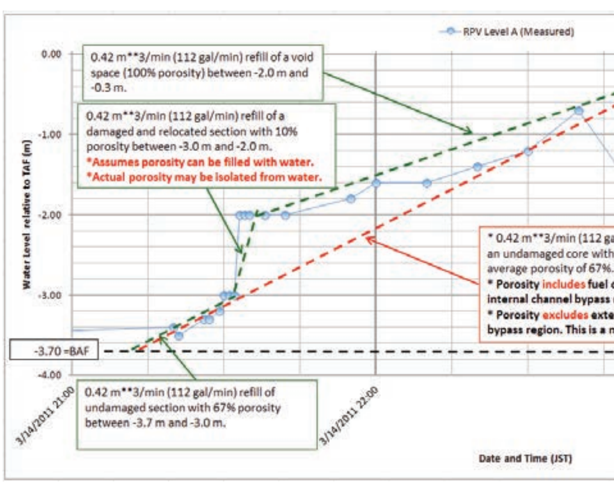
RPV Water Level:
Time Frame 3/14/2011 18:00 to 3/15/2011 09:00



© 2019 Electric Power Research Institute, Inc. All rights reserved.

ELECTRIC POWER RESEARCH INSTITUTE

RPV Water Level:
Time Frame 3/14/2011 18:00 to 3/15/2011 09:00



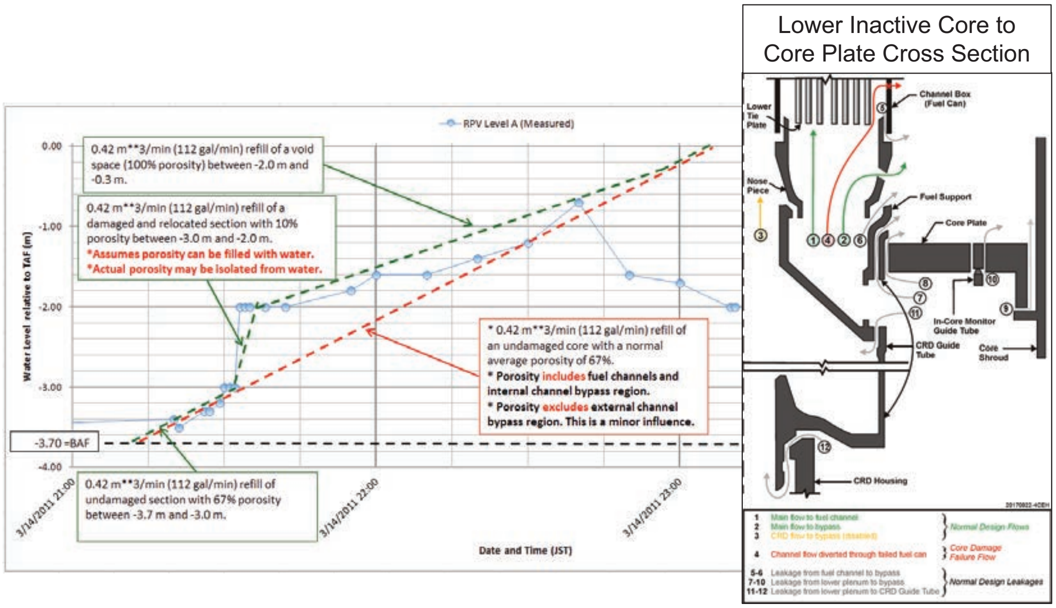
MAAP 5.04 Core Debris Mass Distribution at Reflood (3/14/2011 21:30)

	1	2	3	4	5	6		
29	0%	0%	0%	0%	0%	0%		0%
28	0%	0%	0%	0%	0%	0%		50%
27	0%	0%	0%	0%	0%	0%		75%
26	0%	0%	0%	0%	0%	0%		100%
25	0%	0%	0%	0%	0%	0%		125%
24	0%	0%	0%	0%	0%	0%		150%
23	0%	0%	0%	0%	0%	0%		175%
22	0%	0%	0%	0%	0%	0%		200%
21	0%	0%	0%	0%	0%	0%		250%
20	0%	0%	0%	0%	0%	0%		
19	0%	0%	0%	0%	0%	0%		
18	0%	0%	0%	0%	0%	0%		
17	0%	0%	0%	0%	0%	0%		
16	0%	9%	0%	0%	0%	0%		
15	210%	217%	146%	100%	25%	0%		
14	216%	216%	216%	217%	216%	0%		
13	215%	216%	216%	217%	217%	0%		
12	215%	215%	216%	217%	217%	64%		
11	216%	213%	215%	216%	206%	344%		
10	204%	193%	214%	212%	209%	345%		
9	230%	236%	199%	197%	210%	317%		
8	244%	265%	245%	236%	201%	352%		
7	235%	234%	234%	235%	234%	334%		
6	239%	237%	237%	228%	238%	351%		
5	146%	132%	135%	235%	273%	348%		
4	121%	121%	121%	146%	131%	229%		
3	325%	320%	321%	320%	320%	501%		
2	228%	230%	230%	228%	229%	199%		
1	112%	112%	114%	114%	114%	113%		

© 2019 Electric Power Research Institute, Inc. All rights reserved.

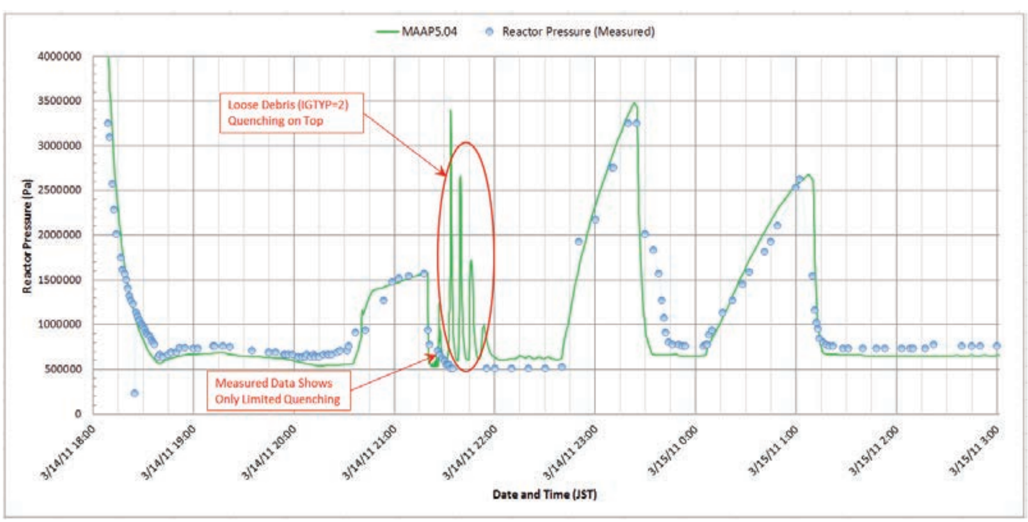
ELECTRIC POWER RESEARCH INSTITUTE

RPV Water Level: Time Frame 3/14/2011 18:00 to 3/15/2011 09:00



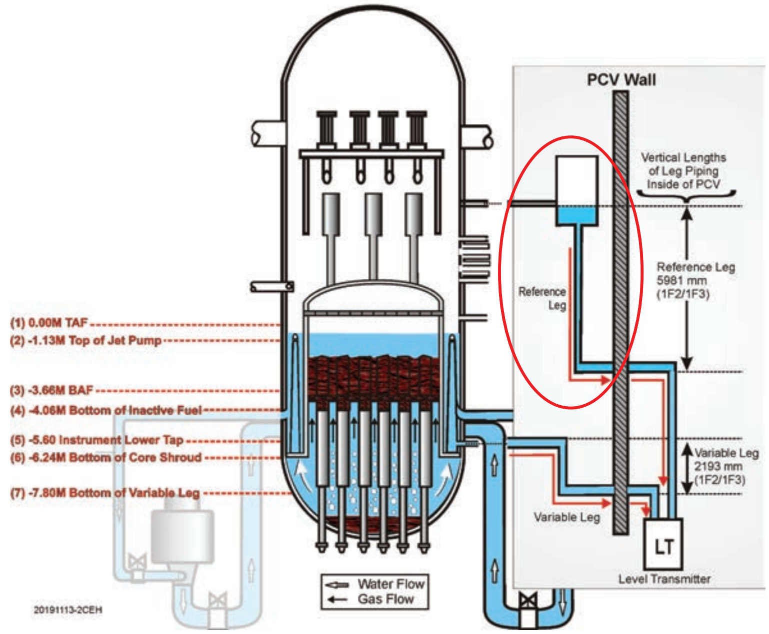
© 2019 Electric Power Research Institute, Inc. All rights reserved.

MAAP 5.04 RPV Pressure Comparison with Measured Data: Time Frame 3/14/2011 18:00 to 3/15/2011 03:00



© 2019 Electric Power Research Institute, Inc. All rights reserved.

1F2 Reactor State:
Time Frame 3/14/2011 22:40 Known Debris Relocation Event



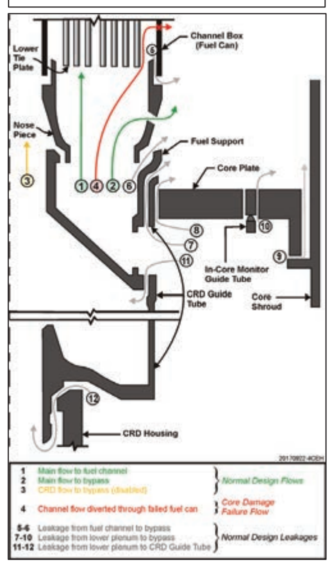
© 2019 Electric Power Research Institute, Inc. All rights reserved.

Core Debris Relocation Event:
Time Frame 3/14/2011 22:40

MAAP 5.04 Core Debris Mass Distribution at Reflood (3/14/2011 21:30)

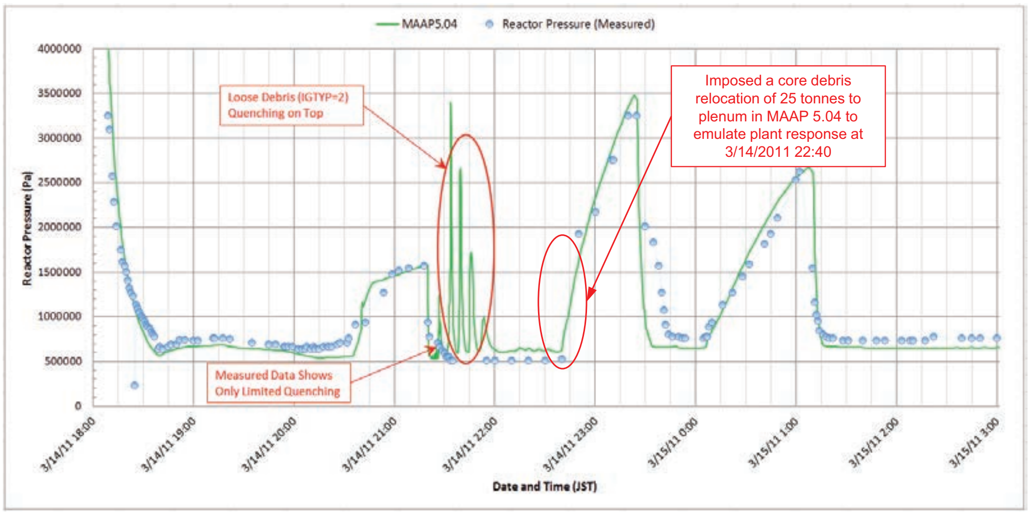
	1	2	3	4	5	6	
29	0%	0%	0%	0%	0%	0%	0%
28	0%	0%	0%	0%	0%	0%	50%
27	0%	0%	0%	0%	0%	0%	75%
26	0%	0%	0%	0%	0%	0%	100%
25	0%	0%	0%	0%	0%	0%	125%
24	0%	0%	0%	0%	0%	0%	150%
23	0%	0%	0%	0%	0%	0%	175%
22	0%	0%	0%	0%	0%	0%	200%
21	0%	0%	0%	0%	0%	0%	250%
20	0%	0%	0%	0%	0%	0%	
19	0%	0%	0%	0%	0%	0%	
18	0%	0%	0%	0%	0%	0%	
17	0%	0%	0%	0%	0%	0%	
16	0%	9%	0%	0%	0%	0%	
15	210%	217%	146%	100%	25%	0%	
14	216%	216%	216%	217%	216%	0%	
13	215%	216%	216%	217%	217%	0%	
12	215%	215%	216%	217%	217%	64%	
11	216%	213%	215%	216%	206%	344%	
10	204%	193%	214%	212%	209%	345%	
9	230%	236%	199%	197%	210%	317%	
8	244%	265%	245%	236%	201%	352%	
7	235%	234%	234%	235%	234%	334%	
6	239%	237%	237%	228%	238%	351%	
5	146%	132%	135%	235%	273%	348%	
4	121%	121%	121%	146%	131%	229%	
3	325%	320%	321%	320%	320%	501%	
2	228%	230%	230%	228%	229%	199%	
1	112%	112%	114%	114%	114%	113%	

Lower Inactive Core to Core Plate Cross Section



© 2019 Electric Power Research Institute, Inc. All rights reserved.

MAAP 5.04 RPV Pressure Comparison with Measured Data: Time Frame 3/14/2011 18:00 to 3/15/2011 03:00

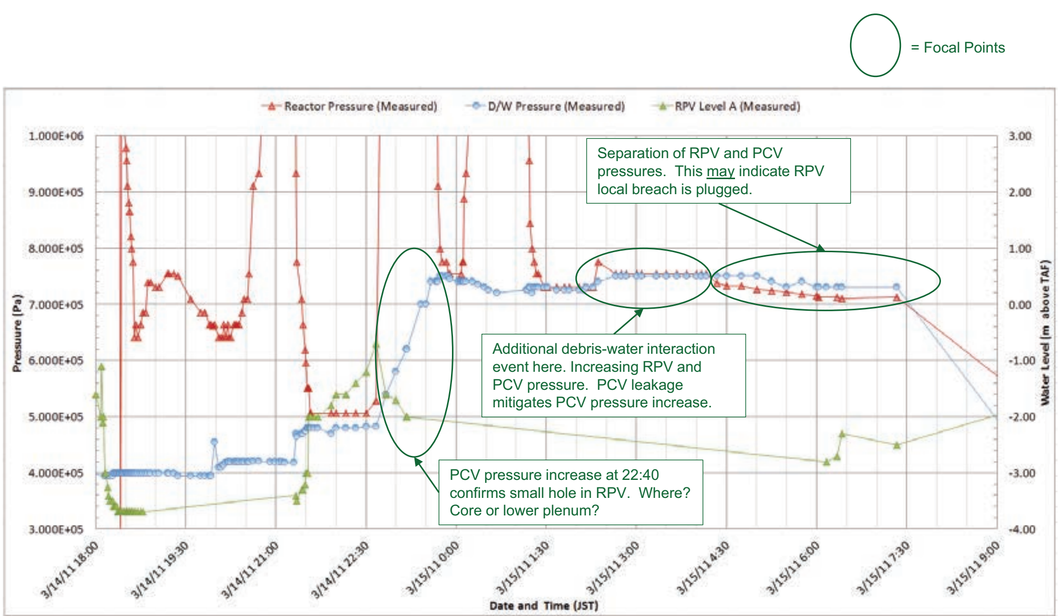


11

© 2019 Electric Power Research Institute, Inc. All rights reserved.

EPRI ELECTRIC POWER RESEARCH INSTITUTE

RPV Pressure , Water Level, PCV Pressure: Time Frame 3/14/2011 18:00 to 3/15/2011 09:00

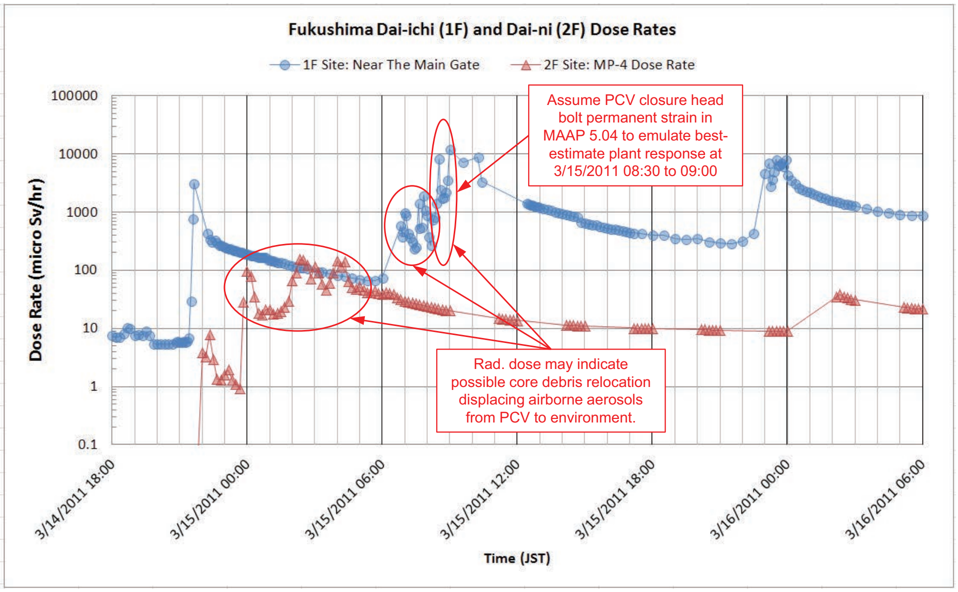


12

© 2019 Electric Power Research Institute, Inc. All rights reserved.

EPRI ELECTRIC POWER RESEARCH INSTITUTE

Dose Rate Measured Data:
Time Frame 3/14/2011 18:00 to 3/16/2011 06:00

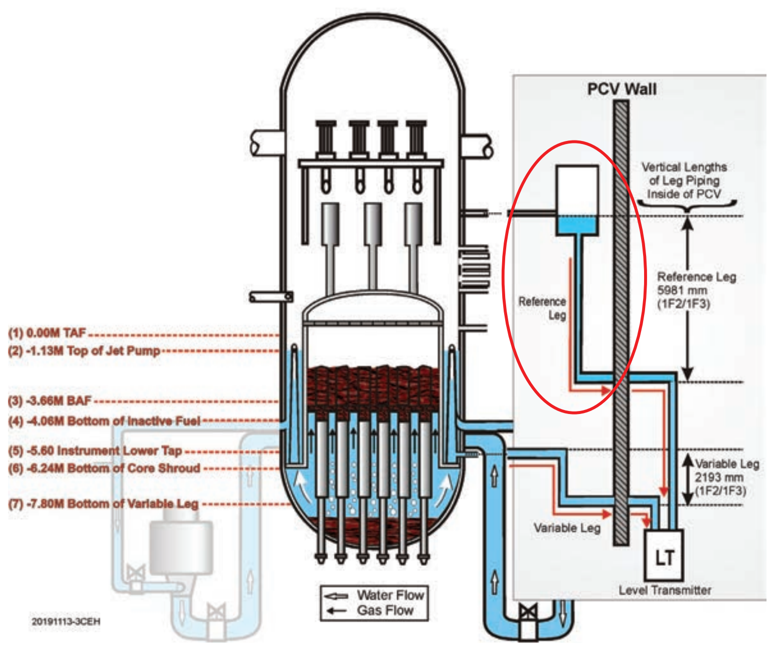


13

© 2019 Electric Power Research Institute, Inc. All rights reserved.

EPRI ELECTRIC POWER RESEARCH INSTITUTE

1F2 Reactor State:
Time Frame 3/15/2011 08:30 Assumed Debris Relocation Event

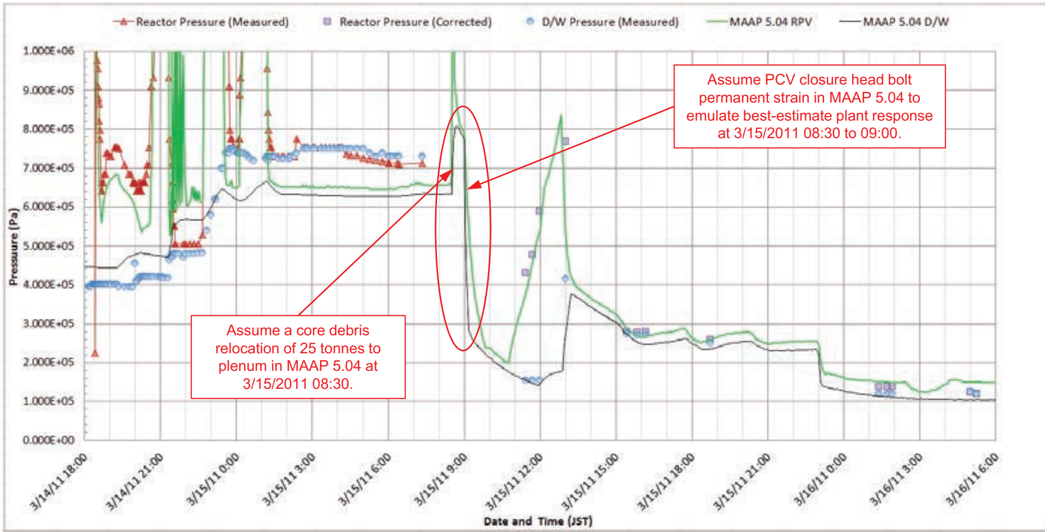


14

© 2019 Electric Power Research Institute, Inc. All rights reserved.

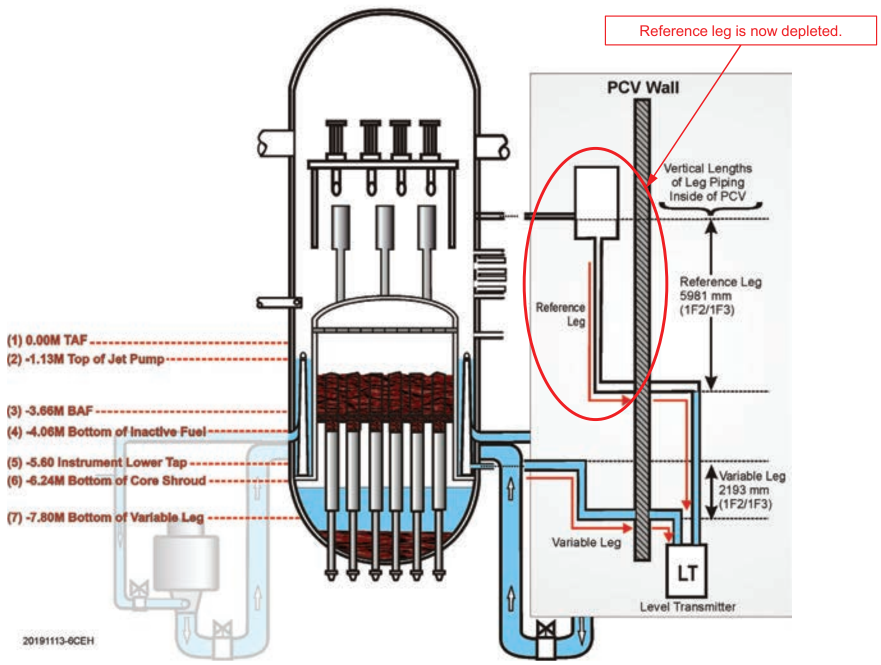
EPRI ELECTRIC POWER RESEARCH INSTITUTE

MAAP 5.04 RPV and PCV Pressure Comparison with Measured Data: Time Frame 3/14/2011 18:00 to 3/16/2011 06:00



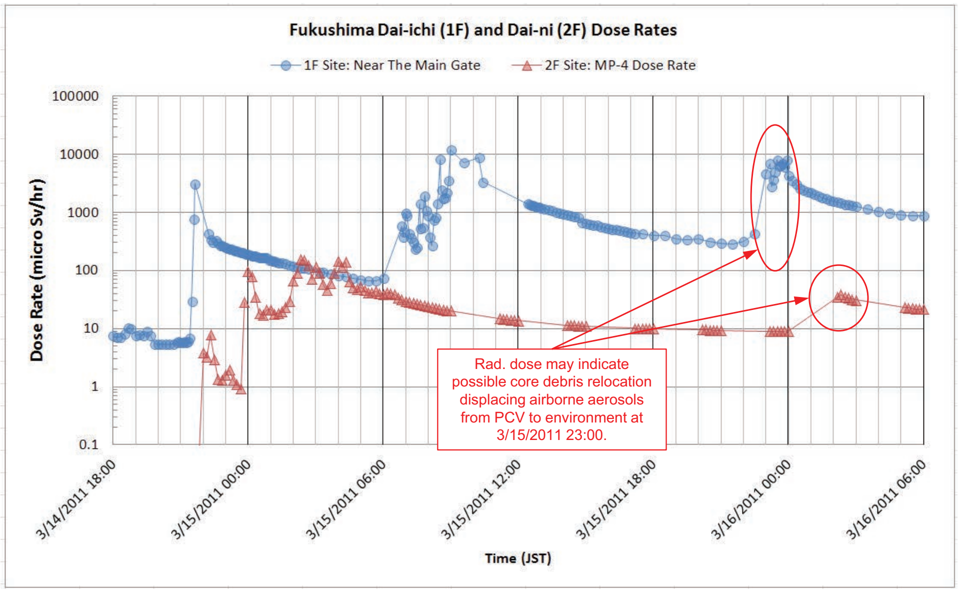
© 2019 Electric Power Research Institute, Inc. All rights reserved.

1F2 Reactor State: Time Frame 3/15/2011 11:00 to 13:00 RCS Re-pressurization Trend



© 2019 Electric Power Research Institute, Inc. All rights reserved.

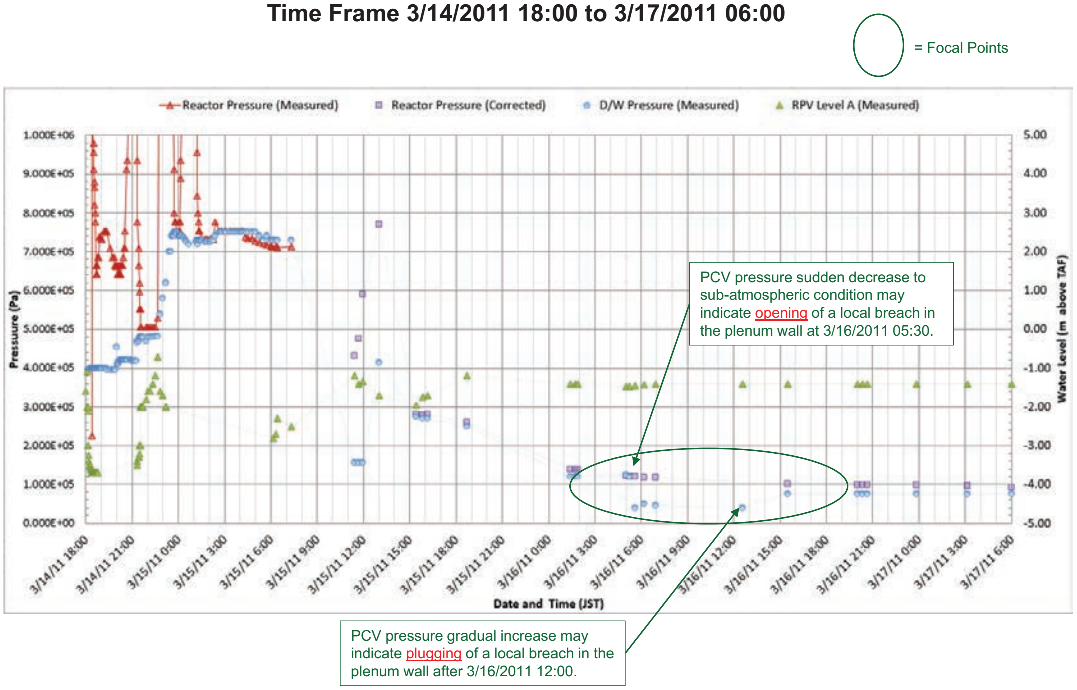
Dose Rate Measured Data:
Time Frame 3/14/2011 18:00 to 3/16/2011 06:00



© 2019 Electric Power Research Institute, Inc. All rights reserved.

EPRI | ELECTRIC POWER RESEARCH INSTITUTE

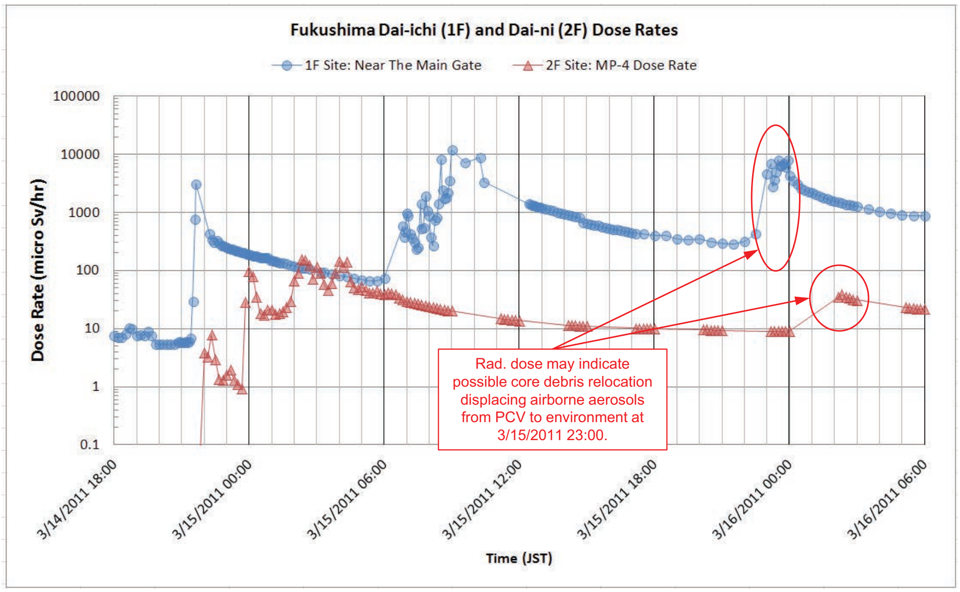
RPV and PCV Pressure and Water Level Measured Data:
Time Frame 3/14/2011 18:00 to 3/17/2011 06:00



© 2019 Electric Power Research Institute, Inc. All rights reserved.

EPRI | ELECTRIC POWER RESEARCH INSTITUTE

Dose Rate Measured Data:
Time Frame 3/14/2011 18:00 to 3/16/2011 06:00

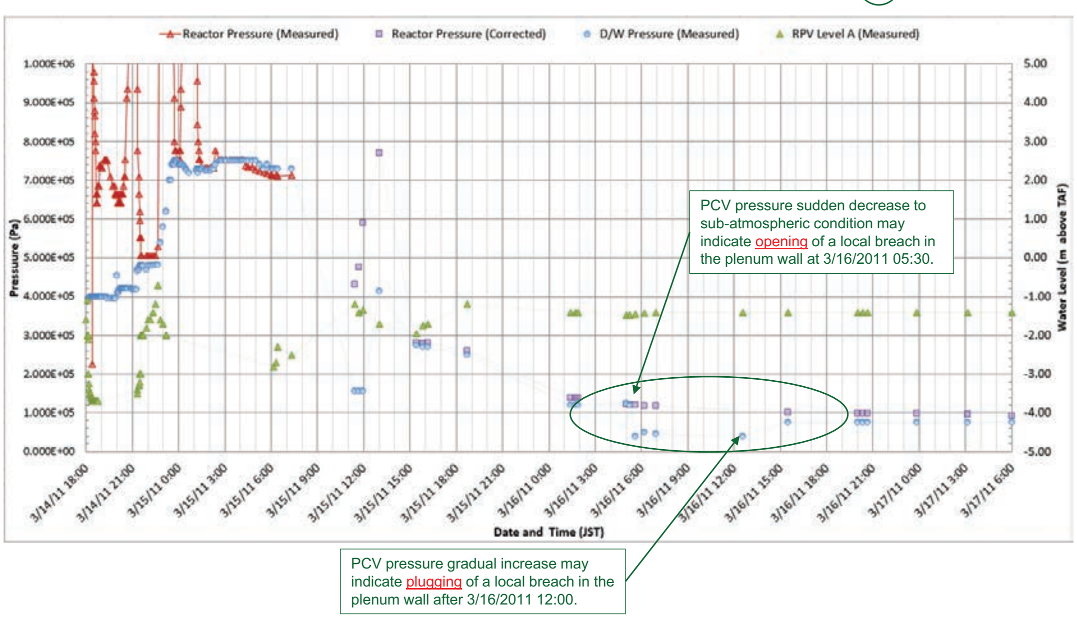


17

© 2019 Electric Power Research Institute, Inc. All rights reserved.

RPV and PCV Pressure and Water Level Measured Data:
Time Frame 3/14/2011 18:00 to 3/17/2011 06:00

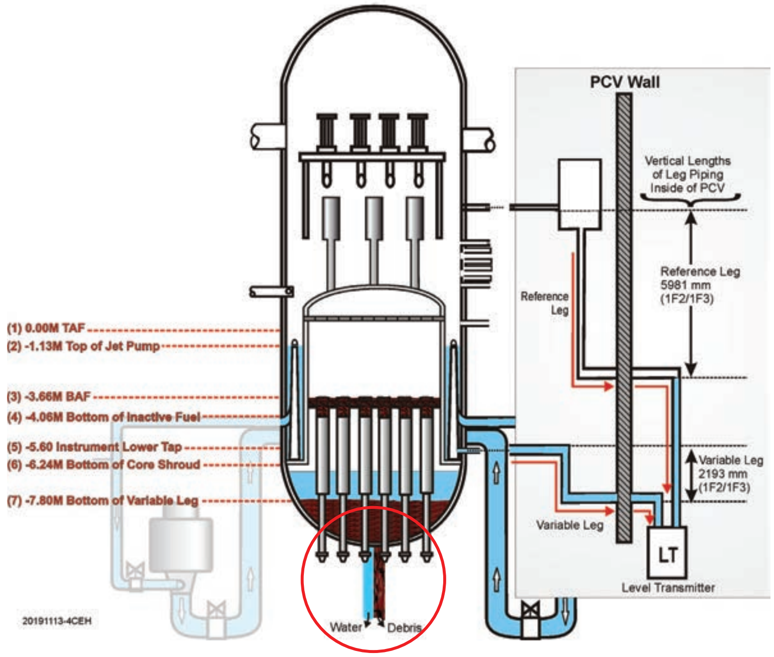
= Focal Points



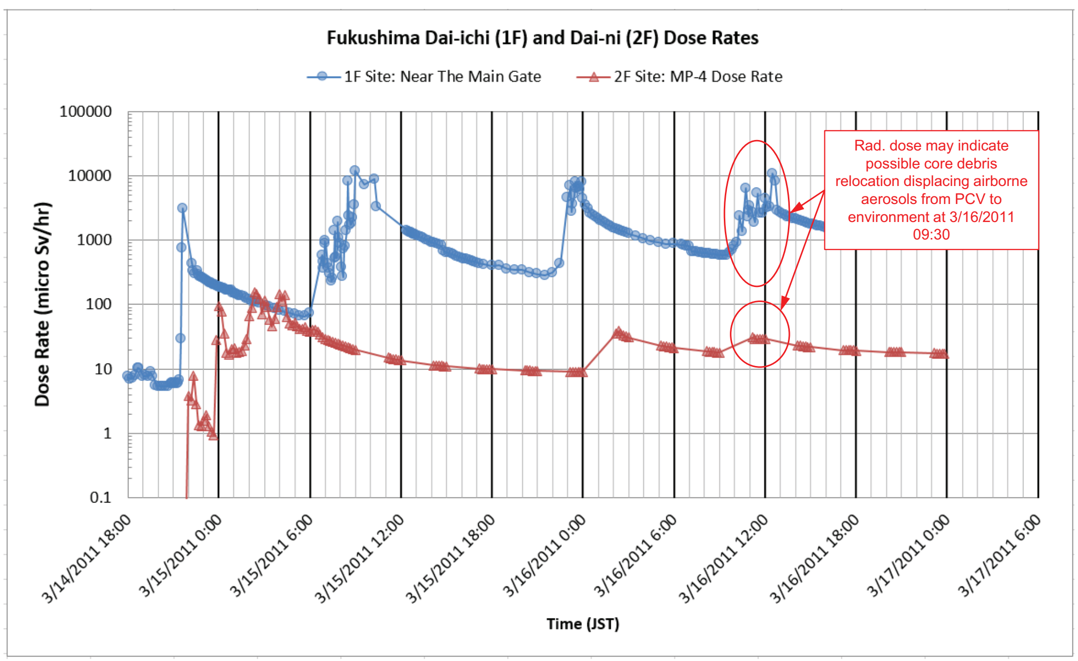
18

© 2019 Electric Power Research Institute, Inc. All rights reserved.

1F2 Reactor State:
Time Frame 3/16/2011 05:30 Potential Local Breach Opening in Plenum Wall



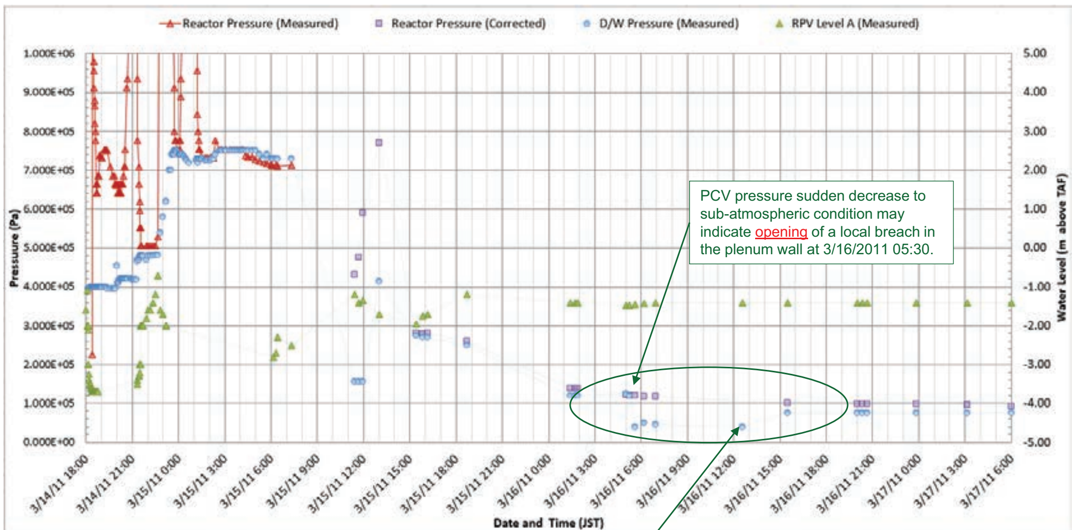
© 2019 Electric Power Research Institute, Inc. All rights reserved.
ELECTRIC POWER RESEARCH INSTITUTE



© 2019 Electric Power Research Institute, Inc. All rights reserved.
ELECTRIC POWER RESEARCH INSTITUTE

RPV and PCV Pressure and Water Level Measured Data: Time Frame 3/14/2011 18:00 to 3/17/2011 06:00

= Focal Points

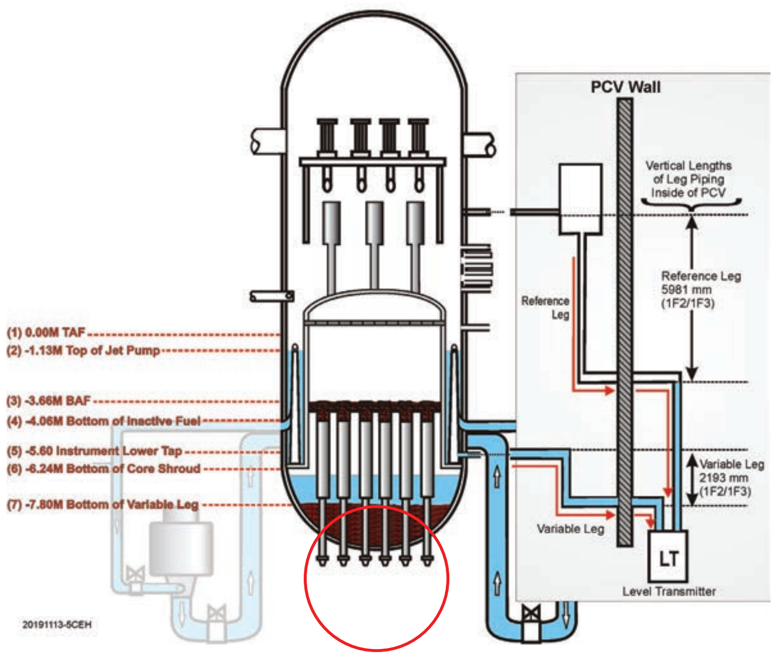


PCV pressure gradual increase may indicate plugging of a local breach in the plenum wall after 3/16/2011 12:00.

21

© 2019 Electric Power Research Institute, Inc. All rights reserved.

1F2 Reactor State: Time Frame 3/16/2011 15:00 Potential Local Breach Plugging in Plenum Wall



22

© 2019 Electric Power Research Institute, Inc. All rights reserved.

Conclusions

- With the guidance of the noted information sources, the following accident progression items are sufficiently understood:
  - In-core debris melt progression timing and extent, including re-flood at 3/14/2011 21:30.
  - Early-stage and late-stage RPV pressure performance.
  - Early-stage and late-stage PCV pressure performance.
  - Early-stage RPV water level performance, including RPV re-flood at 3/14/2011 21:30.
  - Late-stage local breach of lower plenum wall was at a later time, maybe 3/16/2011 05:30. However, the fundamentals are lacking (see below).
  - Successful long-term cooling of bulk debris and vessel wall in the lower plenum.
  - End-state debris configuration in the lower plenum and on the pedestal floor.

- Items with limited understanding, but fundamentals are lacking:
  - Early-stage local breach of RPV at 3/14/2011 22:40 debris relocation.
  - PCV sustained de-pressurization at 3/15/2011 08:30, presumably due to PCV closure head bolt plastic strain.
  - Late-stage RPV water level performance after PCV depress. at 3/15/2011 08:30.
  - Late-stage local breach of lower plenum wall, maybe at 3/16/2011 05:30, due to unsuccessful cooling of debris and wall.
  - Late-stage debris relocations from core to lower plenum at 3/15/2011 08:30 and beyond.
  - Discharge of one fuel assembly upper tie plate through the late-stage local breach.
  - Potential plugging of the late-stage local breach by solidified debris after 3/16/2011 12:00.

23

© 2019 Electric Power Research Institute, Inc. All rights reserved.

EPRI ELECTRIC POWER RESEARCH INSTITUTE

Together…Shaping the Future of Electricity

24

© 2019 Electric Power Research Institute, Inc. All rights reserved.

EPRI ELECTRIC POWER RESEARCH INSTITUTE

C.3.3.2. 1F2 RCIC Performance

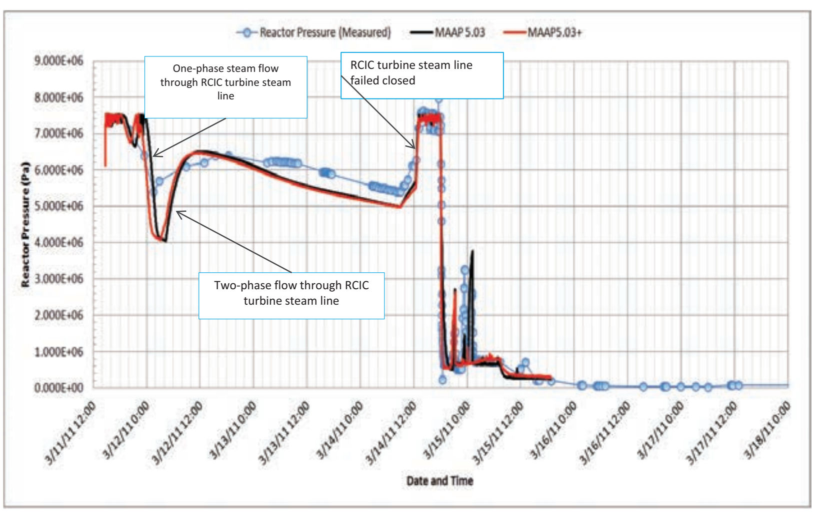
ELECTRIC POWER RESEARCH INSTITUTE

MAAP5 RCIC Model Benchmark Against Fukushima Unit 2 RCIC Performance

Chris Henry
Fauske & Associates
Reactor Safety Technology Expert Panel Forensics Meeting
Washington, DC
November 18-19, 2019

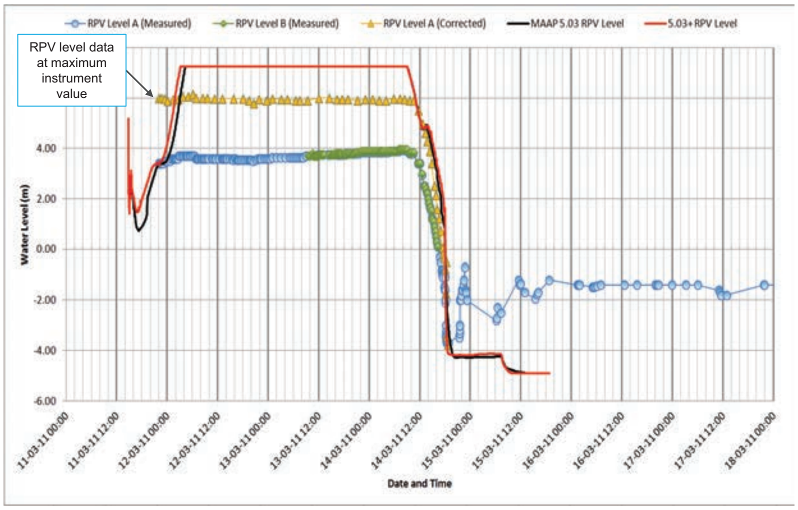
www.epri.com © 2019 Electric Power Research Institute, Inc. All rights reserved.

Comparison of 1F2 Results (27-AUG-2015): Measured Data, MAAP 5.03, and MAAP 5.03+



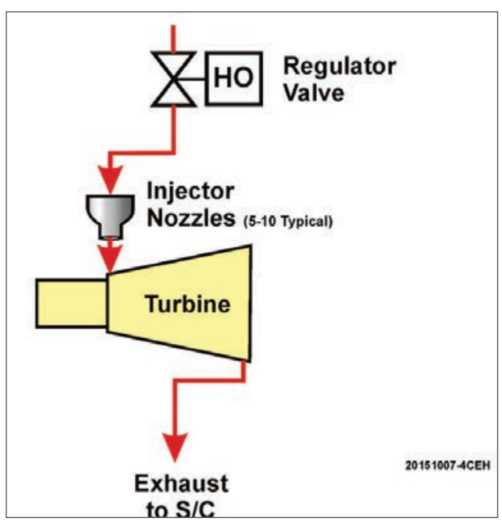
 © 2019 Electric Power Research Institute, Inc. All rights reserved.

Comparison of 1F2 Results (27-AUG-2015): Measured Data, MAAP 5.03, and MAAP 5.03+



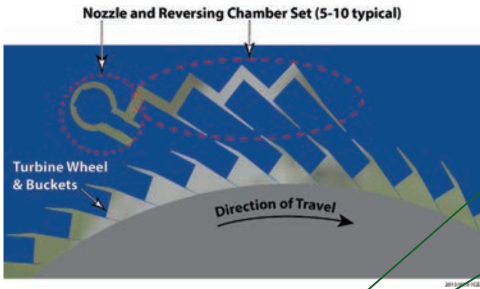
 © 2019 Electric Power Research Institute, Inc. All rights reserved.

RCIC Turbine Schematic



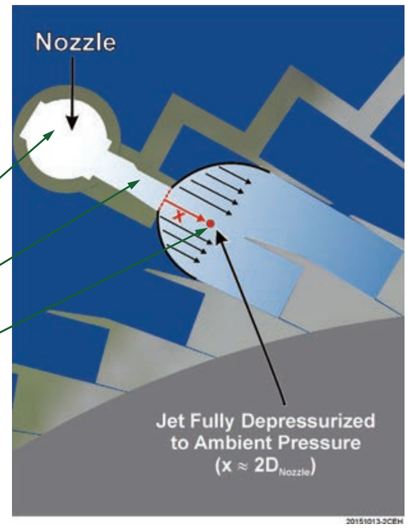
 © 2019 Electric Power Research Institute, Inc. All rights reserved.

Terry Turbine Injector Nozzle Mechanics



Nominal Conditions with 1-Phase Steam:

- RPV Pressure: 7.50E6 Pa
- Nozzle Stagnation Pressure: 3.22E6 Pa
- Nozzle Throat Static Pressure: 1.77E6 Pa
- Nozzle Throat Velocity: 462 m/s
- Depressurized Jet Velocity at x/D=2: 916 m/s



 © 2019 Electric Power Research Institute, Inc. All rights reserved. EPRI ELECTRIC POWER RESEARCH INSTITUTE

MAAP 5.04 HPCI-RCIC Model Thermal Hydraulics

Turbine torque from jet impingement

$$T_{net} = \varepsilon_T r W [2 u_T \cos\beta - r\omega (1 + \cos\beta)]$$

T_{net} = Net turbine torque applied to pump fluid
r = Wheel radius W = Jet mass flow rate
u_T = Fully depressurized jet velocity ω = angular speed
β = Jet entry angle into wheel bucket ε_T = Pump efficiency

Fully depressurized jet velocity

$$u_T - u_{t2} = \frac{(P_{t2} - P_T) A_{t2}}{W}$$

P_{t2} = Static pressure in nozzle throat P_T = Ambient pressure in turbine volume
A_{t2} = Flow area in nozzle throat W = Jet mass flow rate
u_{t2} = Sonic velocity in nozzle throat u_T = Fully depressurized jet velocity

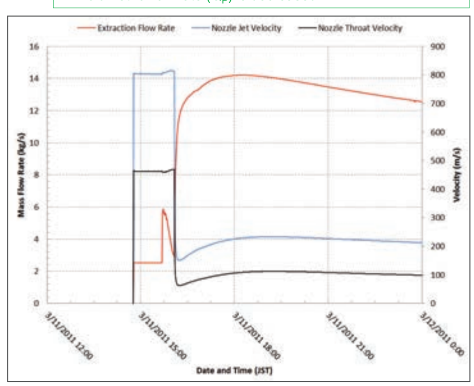
Pump volumetric flow rate

$$Q_P = \frac{T_{net}\omega}{h}$$

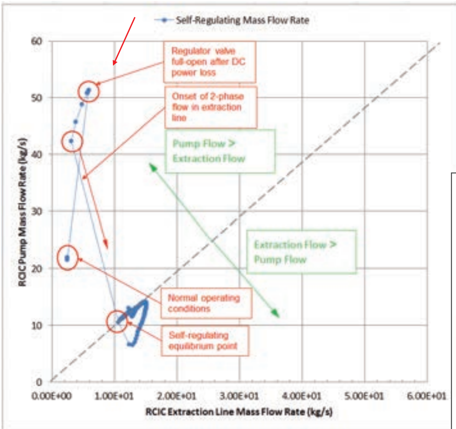
Q_P = Pump volumetric flow rate $T_{net}\,\omega$ = Net power applied to pump fluid
T_{net} = Net torque applied to pump fluid ω = angular speed
h = Pump total developed pressure head

2-Phase Water-Steam Unattended Operation (During Loss of AC/DC Power):

- Compared to 1-phase flow, 2-phase flow moderately increases the extraction line mass flow rate (W).
- However, 2-phase flow dramatically decreases the sonic velocity at the nozzle throat (u_{t2}) and depressurizing jet velocity (u_T) entering the turbine buckets.
- Jet mass flow rate (W) and jet velocity (u_T) control the net turbine torque (T_{net}) and turbine power ($T_{net}\,\omega$).
- Therefore, the pump power ($T_{net}\,\omega$) is decreased and pump volumetric flow rate (Q_P) is decreased.

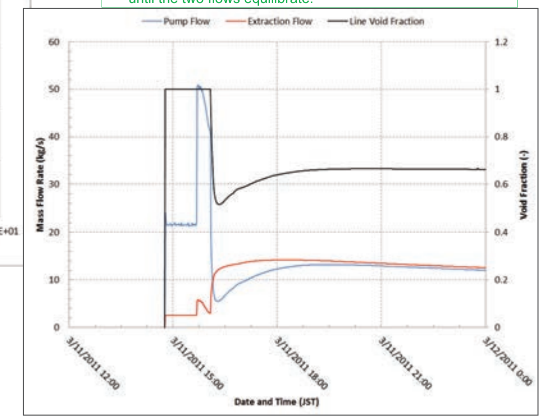


 © 2019 Electric Power Research Institute, Inc. All rights reserved. EPRI ELECTRIC POWER RESEARCH INSTITUTE



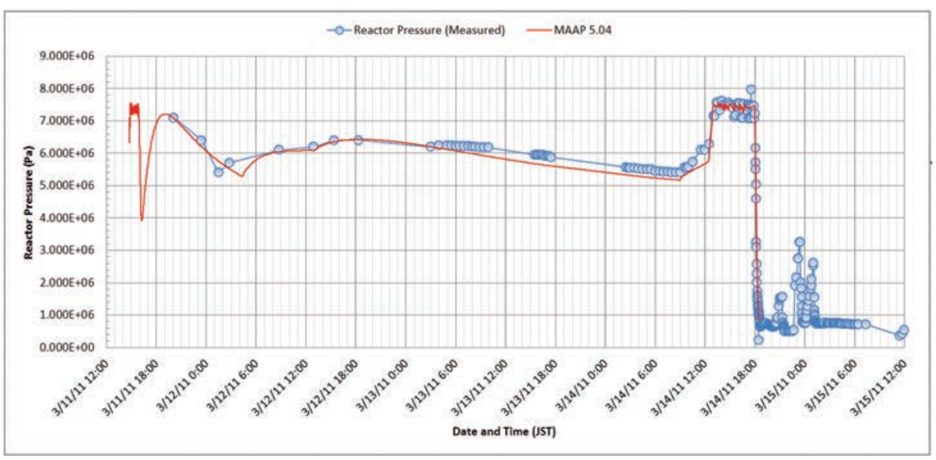
2-Phase Water-Steam Unattended Operation (During Loss of AC/DC Power):

- 2-phase flow exhibits self-regulating feedback between the RCIC pump flow into the vessel and the RCIC extraction line flow out of the vessel.
- The self-regulating feedback is controlled by the 2-phase void fraction entering the extraction line.
- Demonstrated in the 1F2 benchmark.
- Onset of two-phase flow in extraction line increases extraction line flow and decreases pump flow.
- Self-regulating void fraction adjusts line flow and pump flow until the two flows equilibrate.



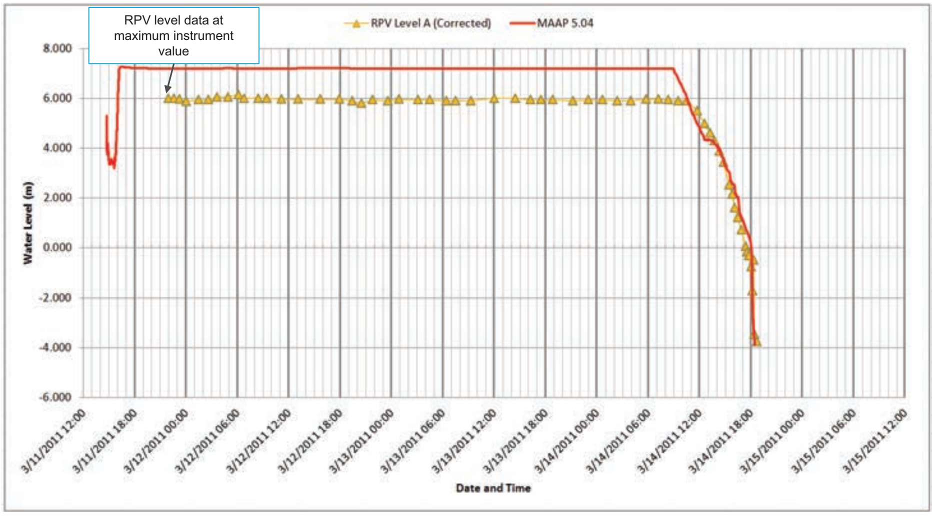
 © 2019 Electric Power Research Institute, Inc. All rights reserved.

Comparison of 1F2 Results (14-JAN-2016): MAAP 5.04 with New HPCI-RCIC Model : 3/11/2011 14:46 to 3/14/2011 18:00



 © 2019 Electric Power Research Institute, Inc. All rights reserved.

Comparison of 1F2 Results (14-JAN-2016): MAAP 5.04 with New HPCI-RCIC Model : 3/11/2011 14:46 to 3/14/2011 18:00



 © 2019 Electric Power Research Institute, Inc. All rights reserved.

HPCI-RCIC Model Summary

- A mechanistic model for turbine-driven HPCI and RCIC systems has been developed for inclusion in MAAP 5.04 draft development code.
- Mechanistic model has a thorough technical basis in two-phase critical flow and two-phase free jet depressurization mechanics.
- During normal operations, model is based upon simultaneous choked flow at the regulator valve and at the nozzle. Choked flow at the nozzle dictates jet depressurization conditions that govern the turbine power and consequent pump flow rate.
- During unattended operation (due to loss of AC/DC power), regulator valve fails wide open. Thus, choked flow occurs only at the nozzle. Again, jet conditions dictate turbine power and pump flow rate.
- Model accommodates 2-phase flow in extraction line if RPV level control is lost during AC/DC power loss.
- Unattended operation with 2-phase flow demonstrates a self-regulation behavior in which the 2-phase void fraction variation in extraction line adjusts extraction flow and pump flow until these flows reach equilibrium.
- Self-regulation behavior dictates the long-term RPV pressure response in the 1F2 accident.
- Benchmarks of the HPCI-RCIC model:
  - System Design Specifications: Model predicts well the known design performance parameters for both RCIC and HPCI systems at both the upper pressure limit and lower pressure limit.
  - 1F2 Accident: Long-term RPV pressure comparison is excellent.
- The issue of RCIC termination is a separate issue. Outside the scope of the current project. Note, the data has trends which may indicate an orderly shutdown of RCIC. This result could occur if RCIC protection logic circuits were re-energized as part of DC power restoration efforts. More study is required.

 © 2019 Electric Power Research Institute, Inc. All rights reserved.


Together…Shaping the Future of Electricity
www.epri.com
© 2019 Electric Power Research Institute, Inc. All rights reserved.
ELECTRIC POWER RESEARCH INSTITUTE

C.3.3.3. 1F3 Vessel Failure

Fukushima Unit 3 PCV Pressure and RPV Plenum Wall Failure Interpretations

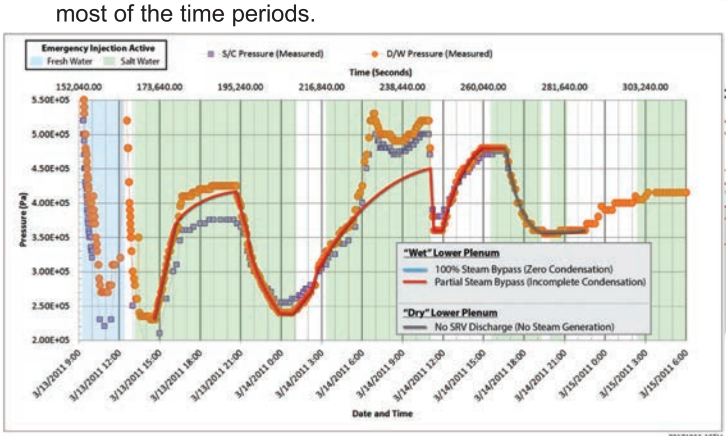
Chris Henry
Fauske & Associates, LLC (FAI)

Reactor Safety Technology Expert Panel Forensics Meeting
Washington, DC
November 18-19, 2019

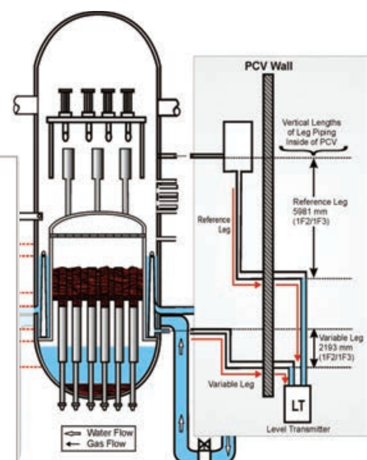
© 2019 Electric Power Research Institute, Inc. All rights reserved.

Periods 4.0 to 5.1 PCV Pressure Interpretation

- Red line represents a “trend line” (not a code result).
- Red line shows what we have learned from measured data and related technical basis for the stratified S/C pool.
- Emergency injection graphic shows emergency injection was active during most of the time periods.



RPV Water Level Nominal State during Periods 4.0 to 5.1

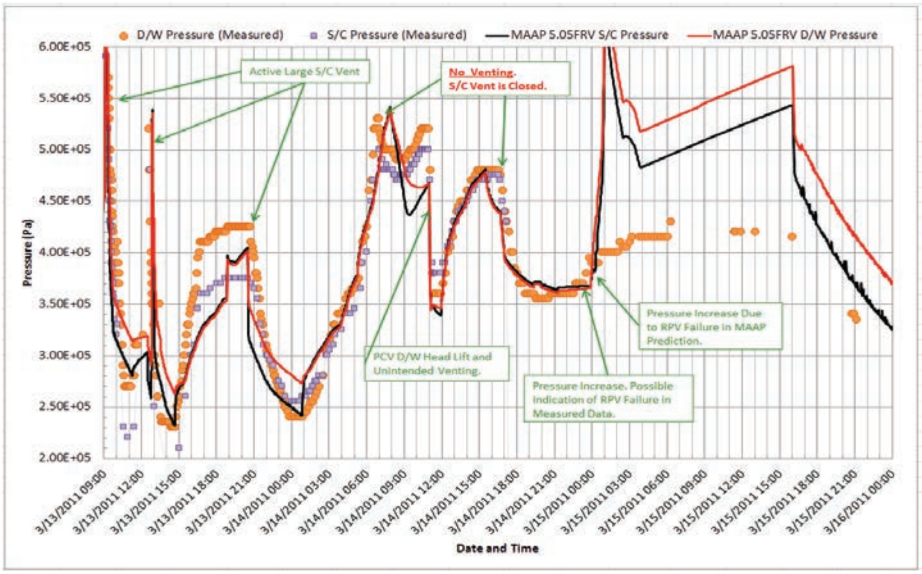


2

© 2019 Electric Power Research Institute, Inc. All rights reserved.

Periods 4.0 to 5.1 PCV Pressure Interpretation

MAAP 5.05FRV PCV D/W and S/C Pressure Comparison with 1F3 Measured Data for Periods 4.0 to 6.0

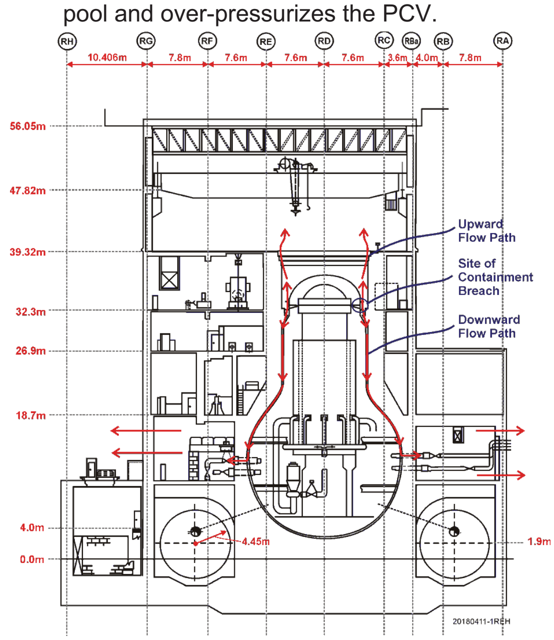


3

© 2019 Electric Power Research Institute, Inc. All rights reserved.

Period 6.0 PCV Closure Head Lift and Discharge

At 11:01, core debris relocation yields steam discharge that bypasses the pool and over-pressurizes the PCV.

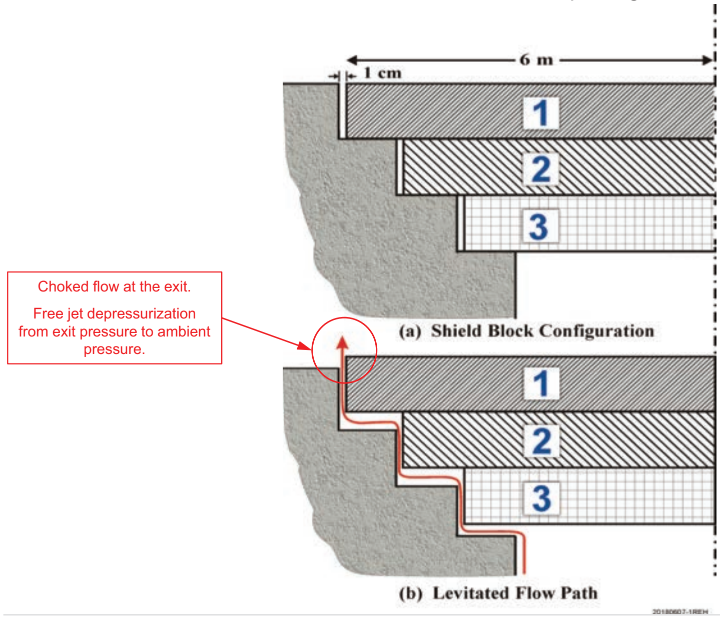


4

© 2019 Electric Power Research Institute, Inc. All rights reserved.

Period 6.0 PCV Closure Head Lift and Discharge

At 11:01, PCV closure head lift and discharge pressurizes the reactor well.
Pressure levitates the shield blocks, opening the flow path.

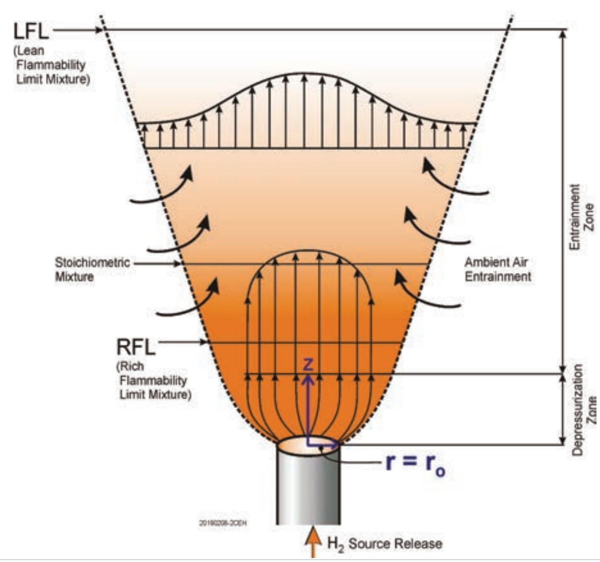


5

© 2019 Electric Power Research Institute, Inc. All rights reserved.

Period 6.0 PCV Closure Head Lift and Discharge

- At 11:01, free jet depressurization at the exit entrains ambient air in the operating deck, generating a combustible mixture and triggering explosion.
- Jet momentum persists after the initial explosion. Jet momentum lifts the roof.



6

© 2019 Electric Power Research Institute, Inc. All rights reserved.

Period 6.0 PCV Closure Head Lift and Discharge

- Use 1F1 explosion as a reference.
- 1F1 post-explosion pressure is minimal and decreasing rapidly at this time.
- 1F1 roof vertical movement is minimal because post-explosion lift force is trivial. No vertical jet to provide lift force in the 1F1 explosion.

- Contrast 1F3 explosion with the 1F1 explosion.
- 1F3 post-explosion pressure is minimal and decreasing rapidly at this time, similar to the 1F1 explosion.
- However, in 1F3, the vertical jet momentum persists after the initial explosion. Jet momentum lifts the main roof segment 200-250 m above the ground.
- This behavior is a distinguishing characteristic of the jet's fundamental role in the 1F3 explosion

© 2019 Electric Power Research Institute, Inc. All rights reserved.

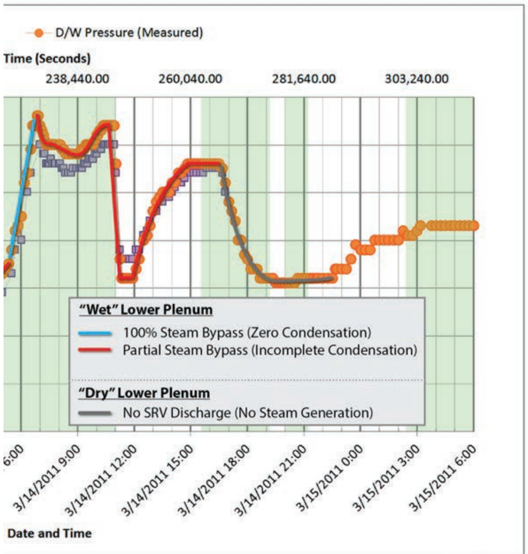
Period 6.0 PCV Closure Head Lift and Discharge

- A flow rate of 72 kg/s and a velocity of 3080 m/s gives the upward H_2 jet a momentum of 2.2×10^5 N as it flowed into the environment after the explosion.
- The 12 m jet outer diameter entrained the surrounding air in a constant momentum process $W_1 U_1 = W_2 U_2$.
- Entrainment and burning of the H_2 jet on the refueling floor would over-pressurize the building siding and structures <u>within a fraction of a second</u>.
- The entrainment velocity is ~8% of the jet velocity. Evaluating the entrainment until the jet velocity slows to 10 or 5 m/s (local wind velocity) gives a vertical plume rise of 290 to 569 m.
- Since the refueling floor is at an elevation of 39 m, the height of the plume would be 329 to 608 m, depending on the wind velocity. This is consistent with the observed plume behavior.
- The plume and the building destruction, can be explained by a H_2 mass on the order of tens of kilograms (25-75 kg), <u>not</u> hundreds of kilograms.

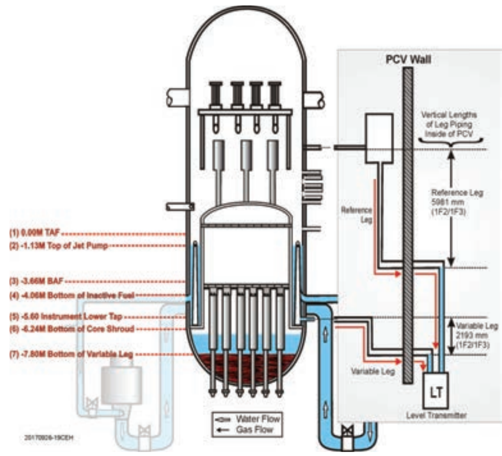
© 2019 Electric Power Research Institute, Inc. All rights reserved.

Period 6.0 PCV Pressure and RPV Plenum Wall Failure

- Red line, blue line, and gray line represent "trend lines" (not a code result).
- Emergency injection graphic shows emergency injection was active during the green parts of the time period.



RPV Water Level Nominal State during Period 6.0

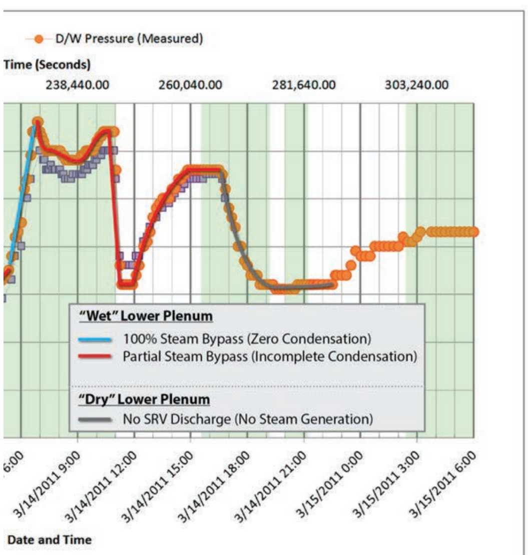


© 2019 Electric Power Research Institute, Inc. All rights reserved.

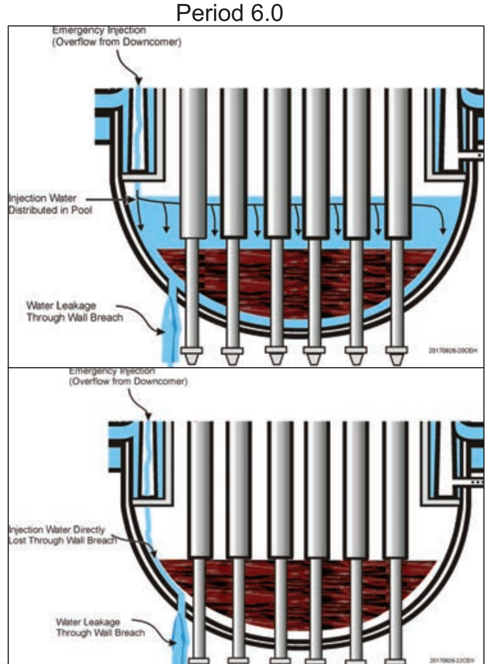
EPRI ELECTRIC POWER RESEARCH INSTITUTE

Period 6.0 PCV Pressure and RPV Plenum Wall Failure

- Red line, blue line, and gray line represent "trend lines" (not a code result).
- Emergency injection graphic shows emergency injection was active during the green parts of the time period.



RPV Water Level Nominal State during Period 6.0

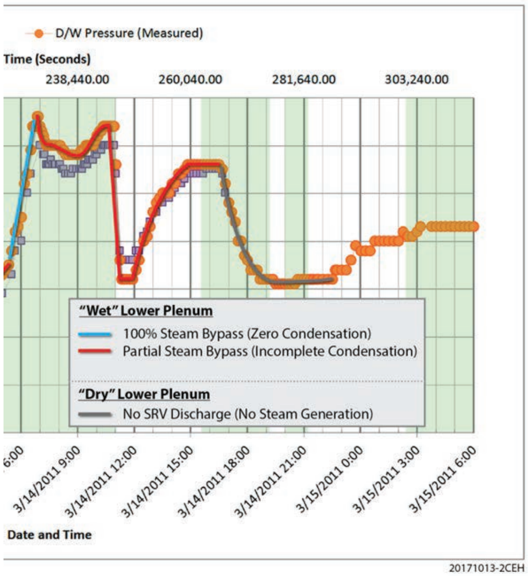


© 2019 Electric Power Research Institute, Inc. All rights reserved.

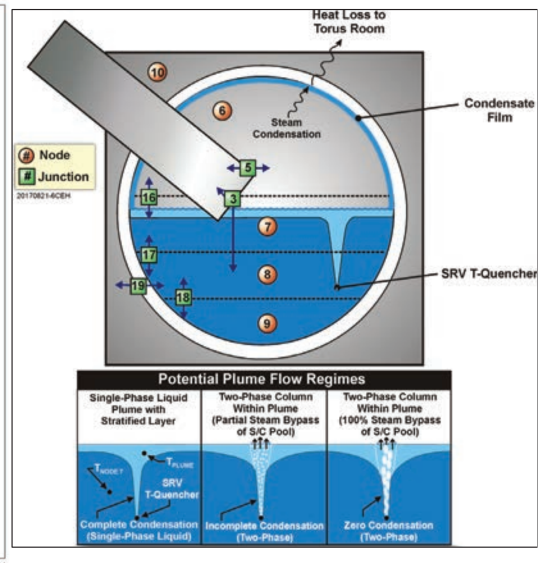
EPRI ELECTRIC POWER RESEARCH INSTITUTE

Period 6.0 PCV Pressure and RPV Plenum Wall Failure

- Red line, blue line, and gray line represent "trend lines" (not a code result).
- Emergency injection graphic shows emergency injection was active during the green parts of the time period.



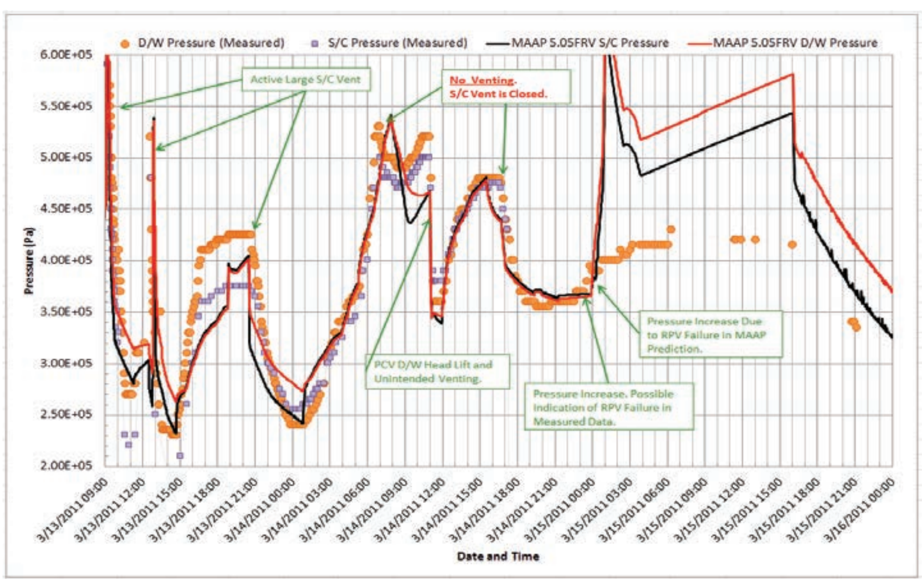
Steam plume source and steam condensation sink in the BWR Mark I suppression chamber (S/C).



© 2019 Electric Power Research Institute, Inc. All rights reserved.

Period 6.0 PCV Pressure and RPV Plenum Wall Failure

MAAP 5.05FRV PCV D/W and S/C Pressure Comparison with 1F3 Measured Data for Periods 4.0 to 6.0



© 2019 Electric Power Research Institute, Inc. All rights reserved.

1F3 Pedestal End-State Configuration from TEPCO Inspection

- TEPCO inspection reveals 1F3 extensive damage to the lower plenum wall and CRD ex-vessel drive rod housings.
- Multiple CRD ex-vessel drive rod housings at the RPV centerline are missing.
- CRD support grid extensive damage due to lower plenum axial strain.
- CRD control blade housings from inside of the lower plenum are now ex-vessel.
- Melting and ablation of the control blade housing wall.
- Substantial accumulated debris in the pedestal region.
- Unit 2 (1F2) inspection of lower plenum region shows substantially less damage.

© 2019 Electric Power Research Institute, Inc. All rights reserved.
ELECTRIC POWER RESEARCH INSTITUTE

Summary Evaluation of the Wet Lower Plenum Scenario Versus Established 1F3 Trends and Events

Start Time	End Time	Trend or Event	Wet Lower Plenum Scenario	Remarks
(JST)	(JST)		Consistency with Trend or Event	
3/13/2011 10:00	3/13/2011 10:00	Rapid water level rise in the jet pumps	CONSISTENT	Assumes water level trends are physically reliable. Debris relocation from core to lower plenum. Pressurization displaces water into the jet pumps.
3/13/2011 12:05	3/13/2011 12:05	Rapid water level rise in the jet pumps	CONSISTENT	Assumes water level trends are physically reliable. Debris relocation from core to lower plenum. Pressurization displaces water into the jet pumps.
3/13/2011 14:40	3/13/2011 20:40	Period 4 Pressurization	CONSISTENT	A natural outcome of the water-covered debris in the plenum.
3/13/2011 20:40	3/14/2011 0:30	Period 4 De-pressurization	CONSISTENT	Large S/C vent open
3/14/2011 0:30	3/14/2011 11:00	Period 5 Pressurization	CONSISTENT	A natural outcome of the water-covered debris in the plenum.

Start Time	End Time	Trend or Event	Wet Lower Plenum Scenario	Remarks
(JST)	(JST)		Consistency with Trend or Event	
3/14/2011 6:00	3/14/2011 11:00	Period 5 RPV Water Level Restoration	CONSISTENT	Following the RPV water level data.
3/14/2011 11:01	3/14/2011 11:01	Rapid water level rise in the jet pumps	CONSISTENT	Assumes water level trends are physically reliable. Debris relocation from core to lower plenum. Pressurization displaces water into the jet pumps.
3/14/2011 11:01	3/14/2011 11:01	Pressurization due to Core Debris Relocation into Water. Results in PCV head lift and R/B explosion.	CONSISTENT	If core debris relocation occurs, debris jet-water interaction is a natural outcome of water accumulation in the plenum.
3/14/2011 11:01	3/14/2011 11:20	11:01 PCV De-pressurization	CONSISTENT	This is an indication of PCV head small leakage when the head re-seats after the 11:01 jet discharge.
3/14/2011 11:55	3/14/2011 16:40	Period 6 Pressurization	CONSISTENT	A natural outcome of the water-covered debris in the plenum.
3/14/2011 16:40	3/14/2011 20:00	Period 6 De-pressurization without active venting	CONSISTENT	An natural outcome of the transition from wet plenum to dry plenum.
3/14/2011 23:30	3/14/2011 23:30	Re-pressurization of PCV after hours of constant pressure behavior.	CONSISTENT	An outcome of the extensive failure of the RPV plenum wall.

© 2019 Electric Power Research Institute, Inc. All rights reserved.
ELECTRIC POWER RESEARCH INSTITUTE

Together…Shaping the Future of Electricity

15

© 2019 Electric Power Research Institute, Inc. All rights reserved.

EPRI | ELECTRIC POWER RESEARCH INSTITUTE

C.3.4. Panel Discussion

SA CODE KNOWLEDGE GAPS

M. T. Farmer and K. Robb

1

HISTORY: LWRS REACTOR SAFETY TECHNOLOGIES (RST) GAP ANALYSIS RESULTS

Category	Identified Gap	Rank	RST R&D Addressing Gap	Bridging Activities
In-Vessel Behavior	**Assembly/core-level degradation**	**1** [A]	• *MAAP/MELCOR Crosswalk (completed)* • Fukushima Forensics	• TSG Tool Development (not completed) • SAMG analyses (not completed)
	Lower head	**2** [A,B]	• Fukushima Forensics	
	Vessel failure	**4** [A,B]	• Fukushima Forensics • Collaboration with CNL through CNWG	
Ex-Vessel Behavior	**Wet cavity melt relocation and MCCI**	**5** [A,B]	• Enhanced SAWA/SAWM modeling capability • ROSAU test program • Fukushima Forensics	
Containment-Reactor Building Response	**H_2 stratification/combustion**	7 [A]	• Fukushima Forensics	
	H_2/CO monitoring	*10* [C]	***Gap Resolved***	
	Organic seal degradation	12 [A]	• Fukushima Forensics	
	PAR performance	13 [C]	-	
Emergency response equipment performance	**RCIC/AFW**	3 [A]	• RCIC modeling and testing program • FY18 NEUP (SNL & TAMU) • Fukushima Forensics	
	BWR SRVs	6 [A]	• Fukushima Forensics	
	PORVs	11 [A]	• Fukushima Forensics	
Additional Phenomenology	**Raw water**	8 [A,C]	• FY18 NEUP (KSU & ORNL) • Fukushima Forensics	-
	Fission product transport/pool scrubbing	9 [A,C]	• Fukushima Forensics	-

[A] Panel consensus was that Fukushima Forensics offer best opportunity for insights in these areas
[B] Panel consensus was that uncertainties in these areas are dominated by uncertainties related to assembly/core-level degradation; thus, addressing assembly/core-level uncertainties should be higher priority
[C] Panel recommended to monitoring ongoing international R&D efforts in these areas to gain insights

2

2

SARNET2 SA RESEARCH PRIORITIES

Items ranked as High

Category	Research Issue
Phenomena during in-vessel accident progression	Core coolability during re-flooding and thermal-hydraulics within particulate debris
	Corium behaviour in lower head
	Integrity of RPV due to external vessel cooling
	RPV vessel failure mode (for BWR)
Phenomena that could lead to early containment (or reactor building) failure	Hydrogen mixing, combustion/detonation
	Melt relocation into water and particulate formation
	FCI incl. steam explosion: melt into water, ex-vessel
Phenomena that could lead to late containment failure	MCCI: molten pool configuration and concrete ablation (for stratified oxidic/metallic melt)
	Ex-vessel corium coolability, top flooding
Phenomena of release and transport of fission products	Oxidizing environment impact on source term
	RCS high temperature chemistry impact on source term
	Containment chemistry impact on source term
	Existing and innovative filtered containment venting systems
Phenomena in spent fuel pool (SFP) storages	Fuel assembly behaviour in spent fuel pool scenarios
New topics related to severe accidents	Instrumentation for severe accidents

Highlighted items correspond to US expert panel identified gaps

W. Klein-Heßling, et al., "Conclusions on severe accident research priorities," Annals of Nuclear Energy, 74, pp 4-11, 2014.

3

A FEW DETAILS ON KNOWLEDGE GAPS

Modeling Gap	Discussion/Rationale
Effect of flashing in wet leg of PTs used to measure water height in core	• Accurate core water level measurement is a critical data need during a SA. • SA codes are used as drivers for plant simulators that are used to train operators.
Core debris holdup on below vessel structure(s)	• Observed in Fukushima • Can affect containment P-T response, and release fission products directly to containment atmosphere
Potential for breakout and re-spreading of deep debris accumulations	• Deep melt accumulations observed in Fukushima and calculated (MELTSPREAD3). • For undercooled sequences, this material will likely heatup and spread further. • Evaluating extent of spreading important for assessing coolability. • Plan on implementing (in MELTSPREAD) as part of ROSAU program
Impact of water on core debris spreading.	• Lack of model validation data with reactor materials. • Evaluating extent of spreading important for assessing coolability. • Being investigated as part of the ROSAU program.
Effect of high metal content on core debris coolability	• Existing coolability data focused on low metal content PWR-type core debris. • Evaluating extent of spreading important for assessing coolability. • Being investigated as part of the ROSAU program.
Melt stream breakup in water during relocation and impact on spreading	• Can impact debris coolability and extent of melt spreading • Implemented in MELTSPREAD (done)
Develop multi-nodal MCCI modeling capability coupled to a reasonably detailed water inventory model	• Significant variations in debris depths/locations/compositions may be expected. • Accurate debris coolability and water inventory assessments need to recognize this. • Initial steps taken in CORQUENCH4 • Plan to complete implementation as part of ROSAU program

4

4

OTHER MODELING IMPROVEMENT NEEDS

- Upper internals heatup and relocation – 100 t material available
- Modeling of SC heatup/stratification
  - Affects containment response/pressurization, pool scrubbing/DF

5

Nuclear Engineering Division
Argonne National Laboratory
9700 South Cass Avenue
Argonne, IL 60439

www.anl.gov

Argonne National Laboratory is a U.S. Department of Energy
laboratory managed by UChicago Argonne, LLC