U.S. Efforts in Support of Examinations at Fukushima Daiichi- November 2020 Meeting Notes with Updated Information Requests

Nuclear Engineering Division

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Meeting Notes with Updated Information Requests

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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 information also has important implications for the safety and operation of U.S. commercial nuclear power plants. This document summarizes results from the Fiscal Year 2021 (FY2021) 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.

A key aspect of prior U.S. efforts, the updated list of information requests, is included in this FY2021 report to ensure that they are transmitted to organizations within Japan. This report also continues to emphasize how information obtained from the affected reactors at Daiichi has been and will continue to be used to update severe accident management strategies 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 for existing and new reactors.

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ACRONYMS AND ABBREVIATIONS

AC	Alternating Current
ADS	Automatic Depressurization System
AFW	Auxiliary Feed Water
AM	Accident Management and Prevention
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARC-F	Analysis of Information from Reactor Buildings and Containment Vessels in Fukushima Daiichi Nuclear Power Station
ATENA	Atomic Energy Association
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
EFPD	Effective Full Power Day
ELAP	Extended Loss of AC Power
EOP	Emergency Operating Procedure
EPA	Environmental Protection Agency
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 Americas
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
LEISAN	Large-scale Equipment for Investigation for Severe Accident in Nuclear reactors
LIBS	Laser Induced Breakdown Spectroscopy
LNG	Liquefied Natural Gas
LT	Level Transmitter
LWR	Light Water Reactor
LWRS	Light Water Reactor Sustainability
L1, L2, L3, and L4	Level location designations (see Figure D-2)
MAAP	Modular Accident Analysis Program
MADE	Mock-up Assembly DEgradation
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
MUSA	Management of Uncertainties in Severe Accidents
NA	No information available or Not Applicable
NANTeL	National Academy for Nuclear Training e-Learning
NC	Non-Condensable
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 Regulation Authority of Japan
NRC	Nuclear Regulatory Commission
NTTF	Near Term Task Force
NUGENIA	NUclear GENeration II & III Alliance
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 FeedWater
TDR	Time Domain Reflectometry
TEM	Transmission Electron Microscopy
TEPCO Holdings	Tokyo Electric Power Company Holdings, Inc.
TIP	Traversing In-core Probe or Top Innovative Practice
TMI-2	Three Mile Island Unit 2
TSC	Technical Support Center
TSG	Technical Support Guidance
TTEXOB	Terry TM Turbine Expanded Operating Band
T1, T2, T3, and T4	Thermocouple designations (see Figure D-2)
TVA	Tennessee Valley Authority
U.S.	United States
V&V	Verification and Validation
VIP	Vessel and Internals Program
VTI	Technical Research Centre of Finland
W	Westinghouse
WW or W/W	Wetwell
XRD	X-Ray Diffraction
XRF	X-Ray Florescence
ZS	Turbine wheel with an 18 inch diameter turbine wheel and stainless steel surfaces.
1F1	Fukushima Daiichi Unit 1
1F2	Fukushima Daiichi Unit 2
1F3	Fukushima Daiichi Unit 3
1F4	Fukushima Daiichi Unit 4

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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 combustible gas 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 information obtained from these units, the U.S. Department of Energy (U.S. DOE) initiated an effort in 2014 for U.S. and Japanese experts in plant safety and operations to meet and discuss recent investigation results from the affected plants at Daiichi. Since its inception, key findings and recommendations are documented in annual reports and other publications.[2 through 9]

1.1. Objectives and Limitations

The U.S. Department of Energy Office of Nuclear Energy (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 for several reasons:
 - 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; and
- 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 References [2 through 9], 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 [10] 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.[11] In addition, both the PWR Owners Group (PWROG) and BWR Owners Group (BWROG) have implemented updates to their severe accident guidance (SAG) and training to address insights from the forensic effort. These guidelines and training will continue to be enhanced as further insights are gained from ongoing work related to the events at Fukushima.

Third, it is beyond the scope of the U.S. DOE Forensics Effort to develop an international program. It is recognized, however, 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. On-going Japan-led post-Fukushima programs (Section 2.2.3) provide an initial basis for this desired long-term international framework. 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 apply examination information in several ways:

- 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.[12] 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,[13,14] 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[15] and the Methods for Estimation of Leakages and Consequences of Releases (MELCOR) code.[16] 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 in completing 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.[5]

Unique U.S. expertise provides Japanese organizations 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 [1], 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 obtained 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, led by Japanese organizations, may be appropriate to support post-accident examinations. 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 in this 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. U.S. experts from industry, universities, and national laboratories participate in this process. Experts from the U.S. NRC, the U.S. DOE, TEPCO Holdings, the Japan Nuclear Regulation Authority (NRAJ), the 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.[7] 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 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 for 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 to examination requests 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 [7]

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 the top blue box on 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.[17] 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 Japanese organizations that participate in expert panel meetings.

Figure 1-1. Objective 2 activities (organizations and programs defined in list of acronyms and abbreviations)

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

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

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[18] and other Japanese organizations, including JAEA, NDF, the Government of Japan, the Ministry of Health, Labor, and Welfare, the Ministry of Economy, Trade and Industry (METI), and the Nuclear Regulation Authority (NRA or NRAJ). Each year at Forensics Effort meetings, presentations based on recently released information are provided by representatives from organizations in Japan (e.g., TEPCO Holdings, JAEA, NDF and NRAJ) and by U.S. topic area leads. TEPCO

[†] The expert panel added this fourth area in FY2016.

[‡] This fifth area was added in FY2018.

Holdings reports documenting unconfirmed and unresolved issues also receive special attention in the forensics effort.[19 through 24] The website created by the Institute of Applied Energy (IAE)[25] is also an important reference for this effort. In addition, as discussed in Appendix B of Reference [3], a website has been developed by this program to archive key references used by U.S. experts to complete these evaluations. At the time this FY2021 report was completed, there were 588 references archived on this website (a significant increase over the 230 references archived at the time that Reference [3] was published).

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
- 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 insights 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 results associated with information coming from the affected units. For each area, prioritized questions of interest were identified; available information was reviewed; and insights gained from evaluating this information were documented. Where appropriate, information requests were revised based on new examination and evaluation results. Additional details, such as the benefits, use, and suggested methods for obtaining higher priority, near-term examination activities were also updated. In FY2020, it was decided that the program would gain more benefit from a more concise report that emphasizes new information and insights. This FY2021 report follows the FY2020 approach. However, a key aspect of the U.S. efforts, the updated list of information requests, is still included in this more concise 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 U.S. DOE, those completed by U.S. 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 [3].

1.4. Report Objectives and Organization

As noted above, this FY 2021 report focuses upon new information and insights that affect changes to findings and recommendations developed by U.S. participants in this effort. Section 2 provides an overview of presentations and items discussed during the FY2021 meeting. Section 3 highlights key findings and recommendations from these meetings. References for this 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 2021 U.S. Forensics Meeting. 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 pub-

lication. Appendices D and E provide additional information about two items of special interest that were discussed during the FY2021 meetings: coolant injection suspension tests (Topic Area 3) and investigations of the 1F1 and 1F3 combustion events (Topic Area 4).

2. FY2021 EXPERT PANEL MEETING HIGHLIGHTS

This section highlights presentations and discussions that occurred during the FY2021 Expert Panel meeting for the U.S. DOE sponsored Forensics Effort. Appendix A includes an agenda and list of meeting participants. Appendix C includes presentations from participants wishing to include them in this publication. Highlights from the meeting are organized into four categories: introductory U.S. presentations (Section 2.1), recent results obtained from Japanese organizations (Section 2.2), U.S. topic area evaluations (Section 2.3), and U.S. presentations related to systems analysis codes development and application activities (Section 2.4). As appropriate, topic area evaluations include updates to key items found in prior year reports, such as questions of special interest, tables summarizing recent examination insights, and recommendations for additional information requests.

2.1. Introductory U.S. Presentations

Introductory presentations were provided by representatives from the U.S. DOE and the U.S. NRC.

2.1.1. Overview of U.S. DOE-Sponsored Efforts to Support Forensics Examinations

Mr. Damian Peko, U.S. DOE program manager, welcomed participants and provided an overview of the DOE-sponsored efforts to support forensics examinations at the affected reactors at Daiichi. As emphasized by Mr. Peko, the focus of this effort is to obtain as much information as possible from the examination activities without adversely affecting NDF efforts to expeditiously complete D&D of the Fukushima reactors and site cleanup. He also reviewed the approach used by this effort and how it benefits participants from Japan and the U.S. In his presentation (see Appendix C.1.1), Slide 4 lists tangible benefits to the U.S. operating fleet and potential benefits to advanced reactors (LWRs and non-LWRs).

The FY2021 meeting was conducted virtually (due to COVID19 restrictions). Although Mr. Peko recognized that in-person meetings facilitate better communication between U.S. and Japanese experts, he observed that this virtual meeting provided an opportunity for broader participation (the number of FY 2021 meeting attendees increased by over 70% from the number of FY2020 meeting participants). Furthermore, the BWROG Fukushima Response Committee and the Emergency Procedures Committee tied remotely to the FY 2021 meeting (more than an additional 20 individuals); and Dr. Gabor, Jensen-Hughes, observed that his meeting summaries are provided to cognizant staff at Exelon, which funds his participation.

2.1.2. Overview of U.S. NRC-sponsored Severe Accident and Source Term Activities

Dr. Hossein Esmaili, Acting U.S. NRC Office of Nuclear Regulatory Research Fuel and Source Term Code Development Branch Manager, also welcomed participants and provided an overview of NRC-sponsored severe accident and source term research activities. As emphasized in Slides 3 and 4 of his presentation (see Appendix C.1.2), these research activities provide important input to several agency regulatory activities. Dr. Esmaili reviewed how MELCOR development and application activities benefit from and contribute to the U.S. Forensics Effort. He also reviewed other on-going international experimental and analysis programs related to MELCOR, MACCS, and SCALE development and applications, including the U.S. NRC-led ROSAU program at ANL and Japan-led post-Fukushima programs (see Section 2.2.3).

2.1.3. FY2020 Findings and Recommendations

Dr. Joy Rempe, Rempe and Associates, LLC, and Technical Lead of the DOE U.S. Forensics Effort, also welcomed participants and provided a summary of information documented in the FY2020 report.[2] Following the approach described in the presentation by Mr. Peko (Section 2.1.1), U.S. experts reviewed information presented by representatives of NDF, TEPCO Holdings, and JAEA during the FY 2020 meeting and formulated several conclusions regarding the complexity of D&D activities at Daiichi and the importance of images obtained from recent investigations in 1F. In her presentation (see Appendix C.1.3), Dr. Rempe reviewed FY2020 findings and conclusions; emphasizing how forensics information was used to verify updated severe accident guidance related to venting and water addition and to reduce uncertainties in severe accident systems analysis codes. She also emphasized the FY2020 observation that information from D&D activities could provide important insights in the areas of radiation protection methods, maintenance requirements, new reactor design and siting, and radiation cleanup. Although FY2020 reviews did not identify any new information requests, she observed U.S. experts clarified text for several information requests and concluded that several existing requests had been completed. Finally, she reviewed FY2020 findings and conclusions regarding the importance of Fukushima-related activities, such as on-going TerryTM Turbine testing. Dr. Rempe concluded her presentation with an overview of the meeting agenda and a link from which participants could access presentation material.

2.1.4. Summary

These introductory discussions led to several recommendations:

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

Recommendation: In future years, U.S. Forensics Expert Panel Meetings should include options for in-person and virtual participation.

2.2. Recent Results from Japan

New information from Daiichi was presented by three Japanese organizations (NRAJ, TEPCO, and JAEA). These presentations were very detailed and provided significant new information. Selected results from these presentations were considered in subsequent U.S. topic area evaluations (see Section 2.3).

2.2.1. Japan Nuclear Regulation Authority (NRAJ)

In his presentation, Mr. Masaya Yasui, Special Coordinator for International Collaboration on Nuclear Regulation, summarized interim results from on-going NRAJ investigations of current conditions at 1F1, 1F2, 1F3, and 1F4. His presentation (see Appendix C.2.1) includes links to several videos (see Slides 3 and 12) with supporting information. NRAJ has been reviewing existing data from plant instrumentation (including seismometer data and radiation measurements) and obtaining additional data, using standard methods (e.g., sampling and smears) and new technologies, such as gamma camera imaging. Likewise, damage to components and systems was evaluated using existing visual information (photos and videos) supplemented with new images from their own inspections, higher resolution videos of the explosions at

1F1 and 1F3, and laser scanning measurements. During the last year, interim results from NRAJ investigations have been discussed with Japanese experts in reactor safety and plant operations. NRAJ hosted meetings with representatives from JAEA, TEPCO, the Atomic Energy Association (ATENA), and other organizations.

Results from NRAJ investigations are providing several new insights having the potential to affect future D&D activities, current understanding of the progression of the events at each unit, and global nuclear safety. Specific areas of investigation and preliminary findings include:

- **SGTS Vent Pipes** - Evaluations of contamination levels in the 1F1/1F2 vent pipes identified several items of special interest:
 - (1) Contamination levels measured in 1F2 vent pipes are significantly higher than in 1F1, although 1F2 venting was unsuccessful;
 - (2) Contamination levels measured in the lower parts of the 1F1/1F2 shared stack were significantly higher than those measured in upper locations; and
 - (3) 1F1/1F2 vent piping contamination levels are higher than in the 1F3/1F4 vent piping.

NRAJ evaluations identified reasons for the latter two items. Item 2 is attributed to differences in how the older 1F1 SGTS piping terminates at the bottom of the shared stack; similar designs may exist at other sites in Japan and in other countries. Item 3 is attributed to differences in conditions associated with the accident progression (such as the suppression chamber water temperature and release timing). Investigations to determine the factors that led to Item 1 are underway.

- **Shield Plugs** - Evaluations of contamination levels beneath the shield plugs of 1F1, 1F2, and 1F3 led to several preliminary findings (Slide 18 in Appendix C.2.1):
 - (1) Cs-137 measurements under the shield plugs of 1F2 and 1F3 are in the 10^{16} Bq (10^6 Curie) range, with levels for 1F2 somewhat higher than in 1F3;
 - (2) Measurements for Cs-137 under the shield plugs of 1F1 are more than two orders of magnitude lower, in the 10^{14} Bq (10^4 Curie) range.

NRAJ observed that Item 1 may explain the relatively small amount of Cs-137 released to the atmosphere from these units. Differences for release mechanisms between 1F1 and 1F2/1F3 are being discussed. The amount of Cs-137 present within the shield plug is under investigation by NRAJ because of upcoming activities related to D&D and long-term storage of this shield plug. However, this topic is also of interest for reactor safety. NRAJ emphasized these measured contamination levels are preliminary; confirmatory measurements for the Cs-137 contamination will be conducted within the next year.

- **1F3 and 1F1 Explosion Insights** - NRAJ has been carefully scrutinizing the 1F3 explosion (Slide 24 of Appendix C.2.1). Working with television broadcast companies, NRAJ has obtained higher resolution videos of the 1F1 and 1F3 explosions. The higher resolution images and videos presented at NRAJ-hosted meetings show localized 1F3 building deformation followed by an orange fire-flame and a rising smoke-ball. Although informed by the “multiple explosion hypothesis” proposed in the U.S. Forensics Efforts FY2018 report [4], NRAJ evaluations indicate this hypothesis is contradicted by the following findings:
 - (1) Localized building deformation could be caused by an explosion. Subsequent events, however, are not easily attributed to explosions. In particular, the coexistence of a stable fire-flame and an explosion seems unrealistic.
 - (2) An orange fire-flame suggests organic materials may have contributed to the amount of combustible gases present in the reactor building. The existence of organic material could explain the dark color of the smoke-ball and why the rising smoke-ball is covered with smoke.

- (3) Considering the presence of steam, it would be difficult for the reactor building to retain the concentrations of hydrogen required to produce the 1F3 explosion described in [4].
- (4) Application of the Dorofeev formula [26] may not be appropriate for hydrogen and if applied, large uncertainties should be acknowledged.
- (5) Images of the 4th floor at 1F3 indicate that damage was caused by multi-stage phenomena.

In higher resolution videos, NRAJ detected orange, but paler, flames in the fire associated with the 1F1 explosion. The NRAJ believes similar organic material, with different concentrations, could have existed in combustible gases within 1F1 and 1F3 (due to the difference in color). The origin of the organic material is being discussed, but no conclusions have yet been reached by NRAJ. NRAJ review of seismometer data (Slide 27 of C.2.1) indicates the 1F1 explosion imparted similar or higher forces to the ground (with most energy being in a downward direction).

As discussed in Section 2.3.4 and Appendix E, U.S. experts observed that temperature-induced degradation of polymers in cables within the PCV could contribute to combustible gas generation within the PCV. However, U.S. participants also commented that any conclusions regarding reactants based on flame colors must consider ‘real world’ conditions in the reactor building prior to the combustion event. For example, the presence of sodium within the reactor building could also produce a yellow colored flame.

- **1F3 SRV Performance under Station Blackout (SB) Conditions** - In reviewing 1F3 data, NRAJ detected a period of time (see Slide 21 of Appendix C.2.1) in which the RPV pressure seems to oscillate in a range below the normal SRV opening point and above the normal closing point. Although the initial SV response was due to the loss of nitrogen accumulator pressure, the reason that this behavior continued after the initial pressure drop (at around 19:40 on March 11) has not been identified.

Mr. Yasui emphasized these insights are preliminary. Efforts are underway to document final results. He expressed interest in future collaborations with U.S. experts on these findings. In particular, he expressed interest in collaborating on insights related to combustible gas effects and indicated that once efforts are completed, the high resolution videos of the 1F1 and 1F3 explosions and an NRAJ report on their 1F combustion events findings will be released. Additional discussions on this topic may be found in Section 2.3.4 and Appendix E.

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

Three presentations were given by representatives from TEPCO (see Appendix C.2.2): Dr. Shinya Mizokami discussed SGTS filter train investigations; Dr. Michal Cibula presented plans for future 1F1 and 1F2 PCV investigations; and Dr. Mizokami reviewed recent evaluation results of 1F3 RCIC operation.

2.2.2.1. SGTS Filter Train Investigations

In this presentation (see Appendix C.2.2.1), Dr. Mizokami reported initial results from and plans for future SGTS filter train investigations. In recognition of the close relationship between activities to support decommissioning and reactor safety (see NDF principals discussed in Section 2.1.1.2 of [2]) and because of concerns that aging effects might affect data integrity, TEPCO initiated these investigations in August 2020 and plans to complete them in February 2021. Using robots with cameras, dosimeters, and gamma cameras, detailed contamination measurements were obtained from the filter train, vent line and around the rupture disc. Future investigations will obtain chemical form and composition information from filter

smear samples. Using this information, investigations are gaining insights about fission product and hydrogen transport during venting, including insights about the potential of backflow into the SGTS filter train and the potential of inflow into the reactor building. At the time of this briefing, preliminary insights include:

- 1F3 rupture disk contamination levels are higher than upstream or downstream levels, suggesting enhanced deposition from turbulence occurred in the disk area;
- 1F2 rupture disk contamination levels were lower than 1F3 values, and no significant differences were observed at upstream and downstream locations, suggesting the 1F2 rupture disk never opened; and
- 1F3 SGTS piping surface level contamination level measurements, using gamma camera imaging, suggest backflow from the vent line into the SGTS filter train. In addition, Dr. Mizokami observed that the 1F2 discharge line was shorter than the stack height, suggesting 1F2 releases were at much lower heights.

Following the meeting, a release from TEPCO Holdings [27] provided results from contamination level measurements obtained near the Reactor Building Closed Cooling Water System (RCW) heat exchanger indicating dose rates of more than 1 Sv/hr (similar to values measured near the SGTS filter train). Water samples from this system, which are planned to be obtained during the latter part of 2021, will be evaluated and used for planning future work in this area.

2.2.2.2. Future 1F1 and 1F2 PCV Investigations

PCV internal investigations provide crucial information, serving as a basis for future decisions regarding debris retrieval, transportation, and storage, as well as a source of information to clarify our understanding of the accident progression within each unit. Dr. Michal Cibula (see Appendix C.2.2.2) provided current estimates for relocated fuel debris distribution and presented plans for near-term investigations within the 1F1 PCV [initial activities to install equipment started but were delayed due to concerns regarding Primary Loop Recirculation (PLR) instrumentation piping interference with other equipment being installed] and plans for debris retrieval from the 1F2 PCV [preparations for installing equipment and expanding penetration diameter were expected to start in November 2020]. Based on prior investigations, using muon technology and robotic investigations, it is believed nearly all the 1F1 core materials have relocated into the PCV and about two thirds of the 1F2 core materials have relocated into the PCV.

Future entries into the 1F1 PCV involve the use of a submersible ROV inserted through the X-2 penetration. Passageways must be cleared of obstructions to allow entry of a larger robot; ultimately, it is planned to use a robot with sampling capabilities. Guide rings will be installed as a way to manage the cables and tethers required to control the robot. Investigations are planned within the pedestal and drywell area.

Future entries into the 1F2 PCV will occur through the X-6 penetration using a device with grasping tools, rotating cutters, and an extension arm capable of removing debris deposits (up to 10 cm in size). A spraying device will be installed in the nearby X-53 penetration to suppress any dust generated during X-6 penetration operations. Using the step-by-step approach emphasized by NDF presentations at the FY2020 Forensics Expert Panel Meeting (see Section 2.1.1.1 of [2]), smaller amounts of debris will first be obtained and examined to inform subsequent investigations and debris removal. To accommodate the bigger devices required to remove larger amounts of debris in subsequent entries, the X-6 penetration will be

enlarged. It is planned that removed samples will be evaluated in the newly constructed JAEA hot cell facility (see Appendix 2.2.3).

2.2.2.3. 1F3 SRV and RCIC System Operation

In this presentation (see Appendix C.2.2.3), Dr. Mizokami discussed recent results from an assessment of 1F3 SRV and RCIC system operation during the accident. To avoid the need to start and stop the RCIC system (and to save DC battery power), operators opened the RPV injection and return lines to the Condensate Storage Tank (CST). As water was injected into the RPV, some appears to have been sent back to the CST. Hence, there is no way to know how much water was injected into the RPV. In addition, pressure data indicate the SRVs cycled in a “safety valve mode”, driven by pressure differences in the RPV and PCV, at pressures lower than their normal setpoint for opening and closing. After the Station Blackout (SBO), there was no additional gas supplied to the SRV nitrogen (N₂) accumulator. It was speculated that the lower observed pressure for valve operation (observed by NRAJ in Section 2.2.1) might be due to partially open SRVs from relaxation of valve springs (associated with the Young's Modulus high-temperature behavior) at elevated temperatures. A “tuned” RELAP analysis was completed to investigate possible SRV and RCIC system operating modes leading to the observed RPV pressure data. Results indicate SRV cycling was required to reproduce the observed pressure behavior, but appear to over-predict the pressure drop from cold water injection. Differences in measured and predicted RPV pressures were speculated to be due to over-mixing calculated by RELAP. In response to a question about possible RCIC leakage and overestimation of 1F3 decay heat, Dr. Mizokami indicated credible RCIC leakage rates and 1F3 decay heat rates were evaluated in the RELAP analyses, but such evaluations could not sufficiently reduce differences between measured and predicted 1F3 RPV pressures.

2.2.3. Japan Atomic Energy Agency (JAEA)

An overview of post-Fukushima related JAEA activities conducted within the Collaborative Laboratories for Decommissioning Science (CLADS) (see Appendix C.2.3) was presented by Drs. Tadahiro Washiya, Masaki Kurata, Shinichi Koyama, and Akira Nakayoshi. JAEA also provided several references with additional information pertaining to these presentations.[28 through 32]

Dr. Washiya emphasized four CLAD research areas: fuel debris characterization, accident progression evaluations, waste management, and remote system and sensing technology development. As evidenced in other Japanese presentations, important insights about the conditions of the affected 1F units are being obtained using advanced technologies, such as the portable gamma-ray imaging cameras, remote detection of gamma-ray and neutron fluxes, three-dimensional visualization technologies to detect hotspots, and plastic scintillation fiber monitors to evaluate decontamination effectiveness and to detect contaminated water leakage from water tanks.

Dr. Kurata discussed results from accident progression calculations investigating the impact of “core energy” on vessel lower head response. Evaluations explored subsequent accident progressions for two significantly different cases: a case with low core material energy at slumping and a case with high core material energy at slumping. To support these evaluations, CLADS has conducted plasma heating tests with prototypic structural and absorber materials and simulant fuel materials (SIMDEBRIS) and tests at the MADE (Mock-up Assembly DEgradation) and the LEISAN (Large-scale Equipment for Investigation for Severe Accident in Nuclear reactors) facilities for a range of 1F accident conditions. In-situ observa-

tions from these tests are providing insights about material liquefaction and solidification temperatures and about prototypic material relocation behaviors. When asked, Dr. Kurata indicated that no eutectic reactions have been observed.

Dr. Koyama described efforts to use SIMDEBRIS to develop and benchmark new defueling and debris characterization methods. These efforts also support JAEA to prepare for characterizing 1F debris at the Okuma Analysis and Research Center, which will be performed in two new hot cell facilities: Laboratory 1, which is under construction, will be used for low and medium level radioactive rubble and secondary wastes; and Laboratory 2, which is under design, will be used for radioactive analysis, chemical and mechanical characterization of fuel debris.

Drs. Nakayoshi and Kurata provided an overview of the status of three JAEA-led OECD projects: Preparatory Study on Analysis of Fuel Debris (PreADES); Analysis of Information from Reactor Buildings and Containment Vessels in Fukushima Daiichi Nuclear Power Station (ARC-F); and Thermodynamic Characterization Of Fuel Debris and Fission Products based on Scenario Analysis of Severe Accident Progression at Fukushima Daiichi Nuclear Power Station (TCOFF). The PreADES project, which will end in December 2021, focuses on collecting information for fuel debris characterization, identifying needs for fuel debris analysis, and preparing for a future international R&D framework for fuel debris analysis. The ARC-F project focuses on collecting, compiling, and managing data and information from examinations inside 1F reactor buildings and containment vessels as well as refining analysis of accident progressions at the affected units. The TCOFF project focuses on improving thermodynamic databases and fuel degradation and fission product release and transport models used in simulating severe accident progression. Although this project has been completed, a follow-on effort, TCOFF-2, is planned.

2.3. Topic Area Evaluations

Presentations provided by area leads in each topic area highlight results presented by representatives from NRAJ, TEPCO, and JAEA during this meeting (Section 2.2), other information released by organizations from Japan during the last year, and relevant progress on related U.S. activities.

2.3.1. Topic Area 1 - Component/System Performance

Topic Area 1, which is led by Dr. Jeff Gabor, Jensen Hughes, and Dr. Kevin Robb, Oak Ridge National Laboratory (ORNL), focuses on recent examination information to address the following questions:

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

The Topic Area 1 presentation, emphasized information provided by Mr. Yasui, NRAJ (Section 2.2.1), previously released NRAJ information,[33] and Dr. Mizokami, TEPCO, (see Section 2.2.2.3) regarding

1F3 RCIC and SRV operation and performance. In their review of this information, the Topic Area 1 leads and expert panel discussion emphasized several items:

- The RPV pressure drop at 16:03 on March 11, 2011 was due to RCIC cold water injection with corresponding steaming;
- Subsequent RPV pressure cycling suggests the SRV opened and closed at pressures below the valve setpoint pressure;
- The SRV opening/closing setpoint drifted downward, possibly due to relaxation of valve spring or other sources of steam loss such as RCP seal leakage (several meeting participants stated that SRVs commonly have leakage issues during normal operation);
- The reduced N₂ accumulator pressure could also have led to a partially open SRV and the observed RPV pressure response; and
- For these time periods, RCIC extraction steam is insufficient to alone remove decay heat, as evidenced by the RPV pressure ultimately increasing.

Topic Area 1 discussions led to several insights and recommendations for future activities:

Insight: Increased containment pressure impacts on SRV operation are typically considered in probabilistic risk assessments (PRAs). It is recognized that valves reclose at elevated containment pressure, and this behavior is considered in defining the BWR Primary Containment Pressure Limit.

Insight: Influence of reduced accumulator pressure on intermediate opening of SRV with possible area reduction is not typically addressed in PRAs.

Recommendation: Given the importance of SRVs in EOPs and SAMGs, additional analyses, and possibly testing, should be performed to evaluate the risk impact of SRV behavior due to low N₂ accumulator pressure. The performance of SRVs, which may vary with vendor, design, and environmental conditions, should be better understood and communicated to plant operators. Analyses should consider the effects of PCV pressure and temperature, operator actions, core decay heat, RCIC steam requirements, and RCIC cold water injection. Additional radiation measurement data from plant piping, if available, may provide insights about valve operation.

Recommendation: Further evaluations should be performed to understand the cause for the observed 1F3 RPV pressure drop, considering the effects of RCP leakage, RCIC performance (and impact of operator actions during the 1F3 event to conserve battery power), and code nodalization.

Some clarifications were made to existing examination requests (see Appendix B), but area leads did not propose any new items. Topic area leads, however, updated the Reference [3] 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 [34]	NA	NA
	No damage observed on outside [35]		
X-51 PCV penetration ^c	NA	No damage observed; pressurized water could not penetrate blockage in standby liquid cooling system line [36, 37]	NA
X-53 HPCI steam supply penetration (1F2/1F3) ^d	High dose rate measured [38]	No damage observed [39]	No damage observed [40]
X-6 PCV penetration (CRD hatch)	NA	Melted material [41, 42]	No damage observed from inside [43]
Equipment hatch	NA	NA	Water puddle [44, 45] unknown source
Personnel hatch and nearby penetrations	No damage observed [46]	NA	NA
HPCI pipe penetration ^e	No damage observed, but high dose rates measured; traces of flow and white sediment observed [38, 46,47]	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) [38,48]	Dose surveys do not indicate leakage from PCV through TIP guides. High dose levels in samples of materials from TIP indexer [49]	NA
WW vacuum breaker line	Leakage on expansion joint of one line (X-5E) [50]	NA	NA
DW/WW vent bellows	Water leakage attributed to vacuum line above [50]	No leakage observed [51]	
DW sand cushion drain pipe	Leakage [52]	No leakage observed [51]	NA
SC water level	Almost full [24]	Middle [24]	Full [24]
DW Water level	~2 m[24]	~0.2 m[24]	~6 m[24]
Torus room	Partially flooded [53, 54]	Partially flooded [55]	Partially flooded [55]
	Rusted handrails/equipment [34]	Non-rusted handrails/equipment [34,56]	Non-rusted handrails/equipment [34,57]
	NA	Some room penetrations tested, no leakage observed [58]	NA
MSIV room	Limited view obtained [59]	Water leakage cannot be observed [60]	Leakage in Line D near bellows [61]

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

Area	1F1	1F2	1F3
DW shield plugs	Reactor well shield plug displaced [62]	Possible leakage [63]	Leakage likely due to radiation measurements at head and presence of H ₂ burn [24,64]
DW head/flange	No obvious PCV flange deformations observed; but elastic stretching of bolts during event possible [Appendix C.2.3.2 of FY20 report]	NA	NA
RCIC or other low SC piping	NA	Suspected leak location, not confirmed [34]	NA
RPV upper head	NA	NA	NA
RPV lower head	Ex-vessel debris images, dose surveys, and sample examinations indicate failure [24,65,66]	Ex-vessel debris and images confirm failure [64]	Ex-vessel debris images confirm failure [64]
SGTS vent path	High dose levels in vent path confirms rupture disk (RD) operation [67]	High dose levels in vent path, without RD disk operation, indicates backflow from 1F1 vent piping into 1F2 vent piping [67]	Elevated dose levels downstream of rupture disk confirms operation of RD; HEPA filter dose levels confirms backflow from 1F3 SGTS piping into 1F4 SGTS piping [67]

- a. Nomenclature: [Clear]: NA; no information available; [Red]: available information indicates damage or leakage; [Orange]: available information suggests possible damage or impairment; [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.

2.3.2. Topic Area 2 - Radionuclide Surveys and Sampling

Topic Area 2, which is led by Drs. Nathan Andrews and David Luxat, Sandia National Laboratories (SNL), focuses on insights from recent radionuclide survey and sampling information from the affected units at 1F. During the Topic Area 2 presentation (see Appendix C.3.2), Dr. Andrews focused upon results from recent NRAJ evaluations of radiation surveys, and contamination level measurements (and related information) obtained from various 1F locations [near the bottom of the 1F1/1F2 vent stack, the shield plugs (1F1, 1F2, and 1F3), 1F1/1F2 SGTS piping, PCV vent piping (1F1/1F2 and 1F3/1F4), and the 1F1 DW head] and pool scrubbing experimental data. This presentation considered information presented in Section 2.2.1 and at recent NRAJ-hosted meetings in Japan.[70 through 77]

Topic Area 2 discussions identified several important insights:

Insight: Shield plug contamination level measurements and dose rates for 1F2 and 1F3 are significantly higher than measurements of 1F1, suggesting the amount of fuel materials remaining within the 1F1 PCV may be lower.

Insight: Gamma camera images obtained with robots provide an effective method for remote detection and visualization of radiation hot spots within the reactor buildings.

Insight: 1F2 SGTS Cs-137 contamination levels are an order-of-magnitude higher than values obtained for 1F1; in particular, at location 14 near the ground level of the 1F2 vent stack (Slide 6 of AppendixC.3.2).

Insight: 1F1 hardened vent pipe contamination levels are higher than 1F3; investigations of plant data (previously obtained and new pool scrubbing experimental data) suggest the 1F1 RPV was breached prior to venting, reducing the amount of pool scrubbing prior to venting.

Insight: Surface radiation levels measured on 1F2 SGTS piping, as discussed by Dr. Mizokami, may be due to the failure of the 1F2 rupture disk to open. NRAJ is investigating several potential causes for this behavior, including the effects of steam concentration, suppression pool temperature, and pressure.

Insight: The 1F2 rupture disk failure to open is consistent with dosimetry and pressure data: high levels of contamination measured in the 1F2 torus room; suppression chamber CAMS data indicating an 1F2 release in the 78-80 hour timeframe; and 1F2 PCV pressure over 1.5 times its design pressure in the 78-80 hour time period. High levels of ground contamination measured near 1F1 and 1F2 are consistent with other available information, such as earlier 1F1 releases (which experienced earlier core heatup) and the potential for essentially ground level releases from 1F2 due to the configuration of the RCIC room and reactor building.

During meeting discussions, there was considerable interest in the radiation release heights assumed in prior 1F2 consequence evaluations. Several post-Fukushima studies, including the EPRI Fukushima Technical Evaluation [78] and the Containment Protection and Release Reduction (CPRR) technical bases development efforts,[79,80] performed consequence studies to simulate potential off-site doses and public health consequences arising in an ELAP event. Detailed assessments of release height elevation, beyond classification as building or stack release, were not performed in these studies. Past evaluations primarily focused on identifying a plausible source of release from 1F2, given the occurrence of elevated site boundary and off-site dose rate measurements around the time of initial 1F2 core degradation. In the EPRI evaluation,[78] the dominant release characteristic influencing the representation of measured site-boundary and off-site airborne and ground dose rates was the time at which radiological release to the environment was assumed to occur. This was due to the relatively short time periods during which the wind was blowing inland coincident with precipitation.

Postulating an early release point from the RCIC system aids in understanding the site boundary and off-site dose measurements taken during the time when initial 1F2 core degradation was progressing. It further provides a reason for the significant long-term contamination of torus room water observed at 1F2. Beyond these early releases from 1F2, it is likely that a more significant release path was the DW head, which likely began leaking a few hours after releases from 1F2 commenced. In the EPRI evaluation,[78] it was found that 1F2 core degradation and drywell head leakage, with a brief period of wind blowing toward the northwest in conjunction with precipitation, were important factors in representing the overall level of ground contamination detected northwest of the power plant.

The Topic Area 2 discussions regarding venting and radiation release led to the following recommendation:

Recommendation: Given the importance of containment venting in the EOPs and SAMGs, additional analyses, and possibly testing should be performed to evaluate the risk impact of vent valve and vent line rupture disc behavior. These evaluations should consider the height and timing of radiation released from each unit.

Although some clarifications were made on existing examination requests (see Appendix B), Topic Area 2 leads did not propose any new items.

2.3.3. Topic Area 3 - Debris Endstate

The Topic Area 3 lead, Dr. Mitch Farmer, Argonne National Laboratory (ANL), provided insights from Fukushima information related to debris endstate location. In his presentation (see Appendix C.3.3), Dr. Farmer discussed implications from 1F1, 1F2, and 1F3 coolant suspension tests (tests in which water injection to the RPV was suspended to provide insights regarding RPV and PCV thermal response and regarding water leakage locations and size; see [3, 81, 82, and 84]). Dr. Farmer also reviewed evaluations of Cs-137 contamination levels based on information in [70] and provided some thoughts regarding the capabilities and prioritization of hot cell data from the new CLADS hot cells being constructed at the Okuma Analysis and Research Center.[83]

Discussion time during the FY2021 Forensics Panel meeting was limited. In response to a request from U.S. experts, Dr. Farmer agreed to provide more detailed information to support his evaluations of the coolant suspension tests. This information is found in Appendix D of this report.

Topic Area 3 discussions resulted in several insights and recommendations:

Insight: Coolant injection suspension test data implications regarding mass and location of relocated debris are consistent with other sources of information, such as robotic examinations and muon tomography investigations. Anomalies in some thermocouple data may be explained by considering thermocouple position with respect to water level and heating effects due to steam production.

Insight: Evaluations of debris endstate locations based on measured Cs-137 shield plug evaluations are consistent with findings obtained from other methods (robotic and muon tomography evaluations). Results also emphasize the importance of forensics investigations and systems analysis code evaluations for completing D&D activities.

Insight: Thermal response from coolant injection suspension tests might provide a technical basis for reducing (or possibly terminating) water injection and lead to reduced contaminated water generation at the affected 1F units. Scoping calculations presented by Dr. Farmer, see Appendix D, indicate decay heat levels of remaining fuel containing materials within the PCV and RPV are sufficiently low that water injection rates could be significantly reduced (and possibly terminated). Results from these scoping calculations should be confirmed using more detailed models available in systems analysis codes.

Recommendation: Additional consideration should be given to the proposal of using coolant suspension test data as a basis for reducing water injection rates, and possibly terminating water injection, to the

affected units at Daiichi. However, results from these scoping calculations should first be confirmed using more detailed models available in systems analysis codes.

Insight: Capabilities proposed for new CLAD hot cells will provide data required for effective D&D and for gaining insights needed to enhance reactor safety.

On-site facilities are beneficial for more rapidly obtaining data required for D&D activities (eliminating the time and costs required for transportation to off-site facilities). U.S. experts observe that benchmarked results from debris containing prototypic materials would add confidence in data obtained from these new facilities. Some properties, such as debris porosity, morphology and particle size distribution, are critical for several activities: designing casks; developing processes for debris drying, storage, and transportation; and assessing debris coolability phenomena that affect water addition strategies. In light of these considerations, TEPCO and JAEA may wish to revisit the priorities for properties indicated in [83]. Updates to the detailed justifications for this information are provided in Appendix B.2 (see Tables B-10, B-14, and B-16).

Recommendation: Additional consideration should be given to the U.S. request for obtaining examination information that could be used to characterize debris morphology (e.g., porosity, shape distribution, size distribution). Such data are important for future D&D activities and for accident mitigation strategies.

Dr. Farmer did not propose any changes to the U.S. list of examination requests related to Topic Area 3 in Appendix C of Reference [3]). However, U.S. experts provided additional information to clarify the importance of requested debris morphology data (see Appendix B.2).

2.3.4. Topic Area 4 - Combustible Gas Effects

As noted in Section 2.2.1 and Appendix C.2.1,[67] NRAJ investigations are providing new insights regarding the combustion events observed at 1F1 and 1F3. These insights stem from detailed evaluations of seismometer data, observations from higher resolution videos of the explosions, and field investigations at 1F3. As part of their efforts, NRAJ considered information in Forensics Effort reports.[2 through 4] As documented in [2 through 4], however, prior U.S. discussions on Topic 4 did not lead to a consensus opinion about several aspects of the combustion events that occurred at 1F.

In his presentation (see Appendix C.3.4), the lead for Topic Area 4, Dr. Wison Luangdilok, H2 Technology, LLC and Fauske and Associates, LLC (FAI), used information from several references [85 through 93] to further explore the hypothesis that the 1F3 explosion could be a detonation. Dr. Luangdilok supported this hypothesis by presenting images of damage from the 1F3 explosion, available literature about pipe buckling and concrete damage from detonations, and results from recent calculations of the blast overpressure required to incur the observed damage. His calculation results indicated that between 2100 and 2400 kg of hydrogen equivalent (with 5% CO) from oxidation of core components and MCCI was required to cause the observed explosion in 1F3 and subsequent migration from 1F3 to cause the explosion in the 1F4 building. He concluded that the duration of the continuous orange flame (~20 ms) in videos of the 1F3 explosion indicated there was too much hydrogen for a deflagration. He suggested some differences in the 1F1 and 1F3 explosion may be due to the lack of premixing prior to the 1F3 explosion (with localized accumulation of combustible gases, the presence of oxygen, and the potential for 1F3 near-surface detonations).

Discussion time was limited during the FY2021 Forensics Panel meeting, but U.S. experts agreed that more review was needed on this topic. Since this meeting, additional information was released by NRAJ.[94 through 97] In addition, Dr. Wison Luangdilok provided additional information.[98] Appendix E was developed to summarize this information with related findings and recommendations. This appendix discusses evaluations provided by NRAJ for each type of information: seismometer data; satellite and high resolution video images; and damage observed in field surveys. Recognizing there are significant uncertainties in available information, the appendix focuses on reviewing information to identify, and where possible reduce, uncertainties affecting accident management strategies. Several U.S. experts (Drs. Sud Basu, Michael Corradini, Randall Gauntt, Wison Luangdilok, and Chan Paik) reviewed a draft of Appendix E. Their comments were considered in the final version included in this report.

Based on the Topic Area 4 discussions and additional material provided in Appendix E, there are several insights and recommendations:

Insight: NRAJ and U.S. evaluations of the explosions observed at 1F1 and 1F3 are providing new insights with respect to accident progression at these units.

There are uncertainties in predicting combustible gas generation, mixing, and transmission. NRAJ evaluations indicate the importance of considering localized conditions (temperatures, gas concentrations, humidities, etc.) and multi-directional transmission effects.

Insight: Experts are unable to reach a consensus about the hypothesis of a detonation occurring at 1F3, but this is not significant with respect to accident management.

Recommendation: Topic Area 4 investigations should remain focused on identifying and reducing, where possible, uncertainties that impact accident management strategies.

There are uncertainties in estimating the conditions existing at various locations within 1F3. Although estimates of the forces exerted by the 1F3 explosion are important for safely completing D&D, U.S. forensics evaluations should focus on obtaining insights affecting accident management strategies, such as the ability to predict combustible gas generation. Hence, U.S. experts emphasize the importance of proposed 1F examinations efforts within the RPV (see Appendix B.1) and on-going ROSAU tests (see Appendices C.1.2 and C.4.1).

Recommendation: Representatives from the NRAJ and the U.S. Forensics Effort should continue to communicate as new information related to combustible gas generation and ignition becomes available.

Recent NRAJ evaluations have provided several important new insights, including the following:

- Evaluations of seismometer data indicate the energy exerted by the 1F1 explosion imparted similar or higher forces to the ground.
- Evaluations of high resolution images suggest that 1F3 experienced a multi-stage explosion.
- Damage assessments also emphasize the importance of localized effects. Damage observed in some components suggest smaller, localized, detonations may be possible; but other nearby components were undamaged.

These NRAJ evaluations, however, are preliminary. Models applied to extract insights invoke many simplifying assumptions. NRAJ is actively discussing new information and insights with stakeholders in Japan. Representatives from NRAJ indicated U.S. participation was also valuable, citing the 1F3 explosion

hypothesis identified and documented in prior Forensics Effort reports by the Topic Area 4 lead, Dr. Wison Luangdilok. Because of the insights being gained from NRAJ investigations, U.S. participants agreed that continued interactions with NRAJ were of interest.

No changes were proposed to the U.S. list of examination requests related to Topic Area 4 in Appendix C of Reference [3]).

2.3.5. Topic Area 5 - Operations and Maintenance

For Topic Area 5, representatives from the BWR Owners Group (BWROG) provided information emphasizing the importance of information coming from Forensics Examinations. The information included several presentations and a video:

- ***TerryTM Turbopump Testing 2020 Update.*** In this presentation (see Appendices C.3.5.1 and C.3.5.2 for presentation and backup slides), Randy Bunt provided an update on the TerryTM Turbine Expanded Operating Band (TTEXOB) project, an international collaborative effort between the BWROG, the Institute for Applied Energy (IAE), DOE [with participation by Idaho National Laboratory (INL), SNL, and Texas A&M University (TAMU)], and EPRI. The goal of this project, which was initiated to investigate the long duration RCIC performance observed at 1F2 and 1F3, is to expand and define the actual operating limitations (margins) of TerryTM Turbine systems (i.e., RCIC/TDAFW) used in the nuclear industry.

Funding reductions in the Fall of 2019 led the project to reduce the originally proposed scope and eliminate the full scale testing planned in Milestones 5 and 6 (MS 5/6). To address data gaps associated with this reduced scope, such as the lack of a full scale duration test with steam, a self regulation full scale test, and a test to assess the impact of steam quality, a MS 5/6 Hybrid test was defined to provide additional confidence in the scaling factor and self regulation test data obtained from testing at Texas A&M University (TAMU).

EPRI compared data collected from plant owners/operators using GS TerryTM turbines under a variety of inlet steam pressures with injection for greater than four hours to data obtained from the TAMU facilities. Results indicate that, in general, data correlated well when an appropriate correction factor is applied. The ZS-1^{**} steam-water/air-water test is being performed to obtain additional data to increase confidence in the correction factor.

- ***Severe Accident Interactive Learning (SAIL).*** This video provides an overview of the updated BWROG training program for plant control room and emergency response center staff. The SAIL training relieves plant owners/operators from having to develop and provide their own training, resulting in significant cost savings. In their introductions to this video, Mr. Bill Williamson, Dr. Phil Ellison, and Mr. Bunt observed that revisions to this training, as well as the technical support guidance, were significantly influenced by evaluations of Daiichi examination information and by insights gained from presentations and discussions at U.S. Forensics meetings. As observed by Dr. Ellison, “The DOE Forensics insights are being utilized to improve plant operations, reduce total operations cost and improve safety in the U.S. and worldwide through integration with BWROG projects”. The

^{**} ZS-1 designates the first test with an 18 inch diameter turbine wheel (“Z” series turbines) and stainless steel surfaces allowing the wheel to be in contact with steam (S). The absence of an “S” indicates the turbine wheel cannot be in contact with steam. Turbines used in plants are “GS” series (with 24” turbine wheels that can be in contact with steam).

SAIL program has been nominated for an NEI Nuclear Energy Assembly 2020 Top Innovative Practice (TIP) Award.

- **Operations and Maintenance-Related Topics.** In this presentation (see Appendix C.3.5.3), Mr. Bunt discussed two items illustrating the importance of information from Daiichi on plant operations and maintenance:
 - (1) The first item, “Lessons Learned from D&D Efforts”, reinforces the FY2020 recommendation regarding the need to provide a document summarizing ‘best practices’ learned from Daiichi D&D by reviewing tables from an Environmental Protection Agency (EPA)-sponsored report.[100] These tables compare the technical effectiveness (efficiency, cost, waste generation, schedule, and safety) of new technologies being deployed in Japan with available U.S. D&D options.
 - (2) The second item, “Safety Improvements Required for Restart”, highlights safety measures implemented at Kashiwazaki-Kariwa Units 6 and 7 to meet new regulatory requirements. The plant has added many safety enhancements, such as alternate cooling systems, enhanced venting and filtration systems, additional hydrogen monitors and recombiners, backup power capabilities, additional countermeasures to withstand natural disasters, and new severe accident and emergency response measures. In addition, evaluations are conducted to confirm the adequacy of the new measures.

Topic Area 5 discussions emphasized several insights and recommendations:

Insight: Expert panel members expressed support for the TerryTM Turbopump Test program, which offers the potential for important reactor safety insights and reduced plant operating costs. The project has adopted a strategy to offset the reduced funding identified in FY2019.

Insight: Expert panel members expressed support for the systematic approach being used to implement lessons learned from the Forensics Examination information into SAIL training and underlying severe accident guidance.

Insight: Members of the expert panel are interested in lessons that can be obtained in the area of radiation protection, and several information requests documented in Appendix B (RB-1, RB-2, RB-5, PC-6, PC-9, PC-12) have the potential to affect plant maintenance activities. Representatives from several organizations (e.g., DOE, BWROG, EPRI, and NRC) expressed interest in participating in such an effort. It is planned to start this activity during FY2021.

Recommendation: To increase the impact of information from Daiichi, an information bulletin should be prepared regarding radiation protection ‘best practices’ learned from Daiichi D&D activities. With participation by BWROG, EPRI, and NRC, the DOE forensics program should lead this effort during FY2021.

Area 5 leads did not propose any changes to Topic Area 5 recommendations (see Section 7 of Reference [3] or the U.S. list of examination requests related to Topic Area 5 in Appendix C of Reference [3]).

2.4. Systems Analysis Code Development and Application Activities

2.4.1. MAAP Update and Related EPRI-Sponsored Activities

In her presentation (see Appendix C.4.1), Ms. Kelli Voelsing provided an overview of recent EPRI severe accident research activities, including recent activities to enhance and apply the EPRI-sponsored

Modular Accident Analysis Program (MAAP) code. EPRI continues to participate in several international projects, such as the OECD PreADES, ARC-F, ROSAU, and Management of Uncertainties in Severe Accidents (MUSA) projects. She highlighted several Fukushima Lessons-Learned improvements made to the MAAP code:

- Improved detail in BWR primary system thermal hydraulic models;
- A new containment fluid stratification model to allow simulation of the suppression pool phenomena inferred from 1F3 PCV pressure data;
- RCIC system and HPCI turbine performance models that allows simulation of the self-regulating RCIC system operational mode seen at 1F2;
- Ex-vessel relocation and debris coolability modeling improvements to reflect images of holdup and relocation of debris containing materials and relocated components observed in investigations of 1F1, 1F2, and 1F3; and
- New corium spreading and molten core concrete interaction erosion models inferred from images obtained during 1F1 examinations.

In addition, EPRI has sponsored MAAP modernization updates, which include converting the code to C++, an automated testing suite, a modernized user interface, and improved user documentation. The re-structured code allows more rapid deployment of new models to address emerging industry needs to simulate new fuels and designs.

Ms. Voelsing also discussed recent efforts to use MAAP to perform uncertainty assessments in their evaluations of the events at Fukushima.[79 through 102] EPRI completed these assessments by defining a baseline (i.e., “best estimate”) scenario that best matched available data and then conducting sensitivity analyses to evaluate the importance of parameters affecting severe accident progression. Uncertainty analyses were then performed with a more limited set of ‘important’ parameters. Results from these uncertainty evaluations were used to optimize BWROG EPG/SAG implementation efforts, identifying actions and decisions that should be emphasized in training and drills. Results found containment venting to be the most important action. Machine learning theory was used to determine influential actions, such as the success and timeliness of water addition that would affect containment venting pressure, the blowdown effect on steam cooling, and the effectiveness of venting at various pressures. Results found that if these three actions are successfully completed, cesium release to the environment will be minimized to acceptable levels.

Subsequent discussions emphasized several important insights and recommendations for future U.S. activities:

Insight: The Forensics effort has led to several improvements in MAAP models, and uncertainty analyses with the improved code have been used to optimize SAMG training.

Recommendation: New RCIC models in MAAP and MELCOR should be benchmarked against Tennessee Valley Authority data in which the RCIC system ran on April 27, 2011 after a tornado (subsequent to the meeting, this plant data was provided to EPRI and SNL).

Insight: The MAAP uncertainty analysis is impressive; considerable effort was required to develop the uncertainty distributions for MAAP input parameters.

Recommendation: An activity should be completed to compare risk-important MELCOR and MAAP input parameters and associated uncertainty distributions.

2.4.2. MELCOR Update and Related NRC-Sponsored Activities

In this SNL presentation (see Appendix C.4.2), Dr. David Luxat provided an overview of recent activities to enhance the NRC-sponsored MELCOR code, which is used for regulatory safety and risk assessment, regulatory decision-making, and scoping assessments to evaluate designs of reactor and non-reactor facilities. MELCOR modernization efforts, related to the code architecture, numerics, performance, capabilities, and maintenance, have led to the following enhancements:

- architectural evolution activities, which separate physics and numerics embedded within MELCOR models, improve calculation efficiency;
- hydrodynamic modeling updates introduce new numerical methods and model interfaces to enhance code robustness and efficiency; and
- code architecture updates improve numerical solution schemes required for implementing new models for fuel and structure degradation and material interactions.

Many of these modernization efforts were motivated by needs identified in Fukushima simulations and by expected new or improved models required to simulate phenomena observed in examinations of the affected reactors at Daiichi, such as improved simulation of the following:

- structures beneath the RPV lower head and within the reactor cavity
- fission product release and transport (including chemical interactions)
- material interactions
- information provided to operators by water level measurement instrumentation (the need for this model was first identified in the FY2017 Forensics Expert Panel report [5])
- enhanced modeling of TerryTM Turbine to consider operator actions and new data from Texas A&M tests (see Section 2.3.5)
- uncertainty assessment enhancements to consider model formulation

This last item, which was elaborated upon by Mr. Lucas Albright, is exploring the effects of underlying biases in existing MELCOR models (to consider ‘unknown’ unknowns in our state-of-knowledge with respect to severe accident progression).

Subsequent discussions emphasized several important insights and recommendations:

Insight: Several industry experts emphasized the importance of the new water level sensor model in MELCOR to provide an indication of instrumentation readings available to plant staff (and compare it with actual predicted water levels within the RPV and suppression chamber). It was observed that in response to Generic Letter 84-23[103], the BWRs located at one U.S. plant site (and possibly others) implemented significant physical changes in the routing of the water level sensor reference leg, decreasing the portion within the PCV from approximately 6 to 0.6 m (20 to 2 feet). As this model is applied, it will be important to obtain accurate plant-specific geometry information.

Recommendation: A water level sensor model, similar to the one implemented in MELCOR, should be implemented in MAAP.

Insight: There are many ‘unknown’ unknowns, such as the performance of SRVs, the RCIC, and other phenomena identified in evaluations of the accidents at 1F1, 1F2, and 1F3. Although there are uncertainties in characterizing many aspects of BWR severe accident progression, several participants cautioned about complex uncertainty evaluations, observing the lack of knowledge to quantify some model input values, such as the timing and mass of water addition, may dominate phenomenological uncertainties.

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. Implementation of these insights and new models can significantly improve our ability to predict the progression of future accidents, prevent loss of knowledge, and maintain the code state-of-practice.

2.5. Updates to U.S. Information Requests

As described in Section 1.1, a primary objective of the forensics effort is to provide 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 FY2015. Every year, these information requests are reviewed and as appropriate, updated. Appendix B presents the current version of these information requests. Since FY2015, several new information requests were added and the status of several U.S. information requests was modified. The requests discuss why the information is needed and how obtained information will be used. U.S. participants factored in experience from TMI-2 examinations, prioritizing information that would be beneficial for defueling efforts and for operations and safety. In addition, participants considered information provided by representatives from TEPCO Holdings and other Japanese organizations (e.g., JAEA, NDF, and NRAJ). Requests were updated to reflect new data obtained from recent 1F examinations and planned future investigations.

During this FY2021 meeting, experts did not identify any new information requests. However, participants noted progress made on several requests, refined existing requests, provided additional documentation on how information was used, and clarified 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 (summary information requests found in Appendix B.1 as well as detailed descriptions for selected information requests found in Appendix B.2) include the following:

- In several cases, new information was provided for “completed” requests because organizations within Japan are deploying new technologies, such as the gamma imaging camera, not available at the time these initial requests were developed. Hence, some “completed” and partially complete information requests (e.g., RB-3a, RB-3c, RB-4, RB-5, RB-11, RB-12, PC-1, PC-3d, PC-5, PC-6) now acknowledge the potential for additional information to be obtained. Detailed descriptions for Appendix B.2 PC-1, PC-3d, and PC-6 were also updated to reflect the potential for this new information.
- Additional information that is now, or will soon be, available was denoted on Reactor Building Requests RB-3 and RB-4.
- Additional RB-5 information that has become available, such as the NRAJ independent surveys (including the use of the gamma imaging capability), are also of interest (in particular, gamma camera images of 1F3 WW vent valves). In addition, new 1F1/1F2 SGTS stack sample evaluations should include detection of metals (Ca, Si, Mg, and Al) found in oxides from MCCI.
- RB-9 was updated to reflect new information provided by NRAJ and planned March 2021 TEPCO 1F2 shield plug investigations.
- RB-10 was updated to reflect new 1F1 information provided by TEPCO.

- The justification for RB-11 was clarified to reflect the importance of this information for benchmarking accident progression evaluations.
- RB-5 (as noted above), RB-14, PC-3b, PC-17, PC-20, and PC-22 were updated to clarify that sample evaluations should include detection of metals (Ca, Si, Mg, and Al) found in oxides from MCCI samples;
- RB-15 was updated to reflect new information obtained from NRAJ and planned examinations by TEPCO.
- The Appendix B.2 detailed descriptions for PC-3d, PC-17, PC-18, PC-19, PC-20, and PC-22 (Tables B-10, B-14, and B-16) were updated to reflect the importance of obtaining information to characterize debris morphology and particle size distribution data. This information is used in debris drying, storage, and transportation activities required for D&D as well as for benchmarking and reducing uncertainty in models for predicting debris coolability and MCCI phenomena.

2.6. Summary

The DOE established the U.S. Forensics Effort to work with TEPCO Holdings and other Japan organizations to learn what information is being obtained from the affected reactors at Daiichi and to communicate this information to cognizant U.S. experts that could use this information to enhance safety of the U.S. commercial fleet. FY2021 meeting presentations and discussions again emphasize the importance of this effort and the benefit being obtained by the nuclear enterprise. Key findings and associated recommendations from this meeting are summarized in Section 3.

3. KEY FINDINGS AND ASSOCIATED RECOMMENDATIONS

Information obtained from Daiichi is required to inform D&D activities. This information also has important implications for the safety and operation of U.S. commercial nuclear power plants. In 2014, the DOE began funding 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, JAEA, and NRAJ) participate in expert panel meetings, discussing data obtained from 1F examinations and planned future investigations.

Since its inception, key findings and recommendations are documented in annual reports and other publications. In FY2020, it was decided the program would gain more benefit from a more concise report that emphasizes new information and insights. This FY2021 report has followed the FY2020 approach. In this section, Section 2 recommendations developed during FY21 discussions are grouped into seven key findings.

Finding 1 and Associated Recommendations:

Available Fukushima-related information from Japan and discussions of this information at DOE Forensics Expert Panel meetings continue to benefit the U.S. operating fleet.

Forensics information and evaluations have led to updates in severe accident guidance and associated technical support guidelines and training. Tables 2-4 and 2-5 of the 2019 Forensics Effort report [3] cite examples illustrating how owners groups have used insights from forensics examinations to update guidance and support procedures for severe accident prevention and mitigation. Of special interest this year were the BWROG SAIL video and the EPRI presentation describing the process for optimizing new BWROG training using the systems analysis code, MAAP, which was updated based on insights from Daiichi. This optimized training focuses on actions having the most impact with respect to risk reduction.

The EPRI and SNL presentations list new models, motivated by forensics examination information and plant data, recently implemented into the MAAP code (e.g., more detailed thermal hydraulic models, an improved containment model allowing suppression chamber stratification, a new RCIC model, improved ex-vessel debris relocation models allowing holdup on ex-vessel structures, new ex-vessel MCCI and spreading models, etc.) and MELCOR code (e.g., models for simulating water level sensors, the RCIC system, and holdup of relocated fuel materials on ex-vessel structures).

Although in-person meetings offer more direct communication between experts in reactor safety and operations, the virtual meeting allowed more broad participation (attendance increased by over 70%). In addition, DOE Forensics Panel meeting discussions benefit more than registered meeting attendees. Portions of the meeting were broadcast to the Fukushima Response Committee and the Emergency Procedures Committee (more than an additional 20 individuals); and Dr. Gabor observed that his meeting summaries are provided to cognizant staff at Exelon, which funds his participation.

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

Recommendation: In future years, U.S. Forensics Expert Panel Meetings should include options for in-person and virtual participation.

Finding 2 and Associated Recommendations:

U.S. evaluations of information from Fukushima are of interest to several organizations within Japan.

Information provided in U.S. presentations and associated discussions continue to raise issues of interest to Japan. During the FY2021 meeting, there was special interest in information presented in the Topic Area 3 and 4 presentations, leading to several recommendations.

Recommendation: Additional consideration should be given to the proposal of using coolant suspension test data as a basis for reducing water injection rates, and possibly terminating water injection, to the affected units at Daiichi. However, results from these scoping calculations should first be confirmed using more detailed models available in systems analysis codes.

Recommendation: Representatives from the NRAJ and the U.S. Forensics Effort should continue to communicate as new information related to combustible gas generation and ignition becomes available.

Recommendation: Topic Area 4 investigations should remain focused on identifying and reducing, where possible, uncertainties that impact accident management strategies.

Finding 3:

The TerryTM Turbine Expanded Operating Band (TTEXOB) project offers the potential for important reactor safety insights and reduced plant operating costs.

The goal of this project, which was initiated to investigate the long duration RCIC performance observed at 1F2 and 1F3, is to expand and define the actual operating limitations (margins) of the TerryTM turbine systems (i.e., RCIC/TDAFW) used in the nuclear industry. Results offer the potential for important reactor safety insights and reduced plant operating costs. The TTEXOB project has adopted a strategy that should allow it to overcome funding reductions and obtain the data required to meet its goals.

Finding 4 and Associated Recommendation:

Japan D&D efforts include multiple demonstrations of effective use of new technologies, such as unique robots, gamma-ray imaging cameras, drones, three-dimensional visualizations of radiation levels and temperatures, and muon technologies. In addition to providing information, not originally anticipated when U.S. experts originally defined information requests, applications of these new technologies could 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 increase the impact of information from Daiichi, an information bulletin should be prepared regarding radiation protection ‘best practices’ learned from Daiichi D&D activities. With participation by BWROG, EPRI, and NRC, the DOE forensics program should lead this effort during FY2021.

Finding 5 and Associated Recommendations:

Important new insights are expected from on-going and planned near-term investigations.

Presentations by NRAJ, TEPCO, and JAEA emphasize important new insights are being obtained with respect to our understanding of the vessel failure, hydrogen combustion, debris heatup and relocation, and

the performance of components and systems such as the SRVs, rupture disks, WW vents, and the RCIC system. Subsequent presentations by U.S. experts emphasize the potential risk importance of such insights. 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.

Several potentially risk important topics identified in FY2021 Expert Panel discussions, which are documented in this report, warrant additional investigation. Several specific recommendations pertain to these topics.

Recommendation: Further evaluations should be performed to understand the cause for the observed 1F3 RPV pressure drop, considering the effects of RCP leakage, RCIC performance (and impact of operator actions during the 1F3 event to conserve battery power), and code nodalization.

Recommendation: Given the importance of SRVs in EOPs and SAMGs, additional analyses, and possibly testing, should be performed to evaluate the risk impact of SRV behavior due to low N₂ accumulator pressure. The performance of SRVs, which may vary with vendor, design, and environmental conditions, should be better understood and communicated to plant operators. Analyses should consider the effects of PCV pressure and temperature, operator actions, core decay heat, RCIC steam requirements, and RCIC cold water injection. Additional radiation measurement data from plant piping, if available, may provide insights about valve operation.

Recommendation: Given the importance of containment venting in the EOPs and SAMGs, additional analyses, and possibly testing, should be performed to evaluate the risk impact of vent valve and vent line rupture disc behavior. These evaluations should consider the height and timing of radiation released from each unit.

Finding 6:

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., regarding its benefits and future applications) was revised for several requests. In some cases, additional clarification was provided regarding desired information.

Finding 7 and Recommendation:

Capabilities proposed for new CLAD hot cells will provide data required for effective D&D and for gaining insights needed to enhance reactor safety.

On-site facilities are beneficial for more rapidly obtaining data required for D&D activities (eliminating the time and costs required for transportation to off-site facilities). U.S. experts observe that benchmarked results from debris containing prototypic materials would add confidence in data obtained from these new facilities. Some properties, such as debris porosity, morphology and particle size distribution are critical in designing casks and processes for debris drying, storage, and transportation and in assessing debris coolability that affect water addition strategies. In light of these considerations, TEPCO and JAEA may wish to revisit the priorities for properties indicated in [83].

Recommendation: Additional consideration should be given to the U.S. request for obtaining examination information that could be used to characterize debris morphology (e.g., porosity, shape distribution, size distribution). Such data are important for future D&D activities and for accident mitigation strategies.

Finding 8 and Associated Recommendations:

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 certain aspects of the accidents.

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. Implementation of these insights and new models into these codes can significantly improve our ability to predict the progression of future accidents, prevent loss of knowledge, and maintain the code state-of-practice.

Recommendation: New RCIC models in MAAP and MELCOR should be benchmarked against Tennessee Valley Authority data in which the RCIC system ran on April 27, 2011 after a tornado (subsequent to the meeting, this plant data was provided to EPRI and SNL).

Recommendation: An activity should be completed to compare risk-important MELCOR and MAAP input parameters and associated uncertainty distributions.

Recommendation: A water level sensor model, similar to the one implemented in MELCOR, should be implemented in MAAP.

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APPENDIX A. FY2021 Meeting Agenda and Attendee List

A.1. November 10-12, 2020 Meeting Agenda

Reactor Safety Technology Expert Panel Forensics Meeting

Agenda

November 10-12, 2020

Virtual Meeting (WebEx: Must Respond to Registration Email for Each Day)

Times relative to start time: 7:00 AM U.S. Eastern Time (UTC-07:00; 9:00 PM Tokyo)

Tuesday, November 10, 2020

00:00	Welcome and Overview of DOE Activities	A. Duncan D. Peko, DOE-NE
00:05	Welcome and Update on Related NRC Activities	H. Esmaili, NRC
00:15	FY20 Findings and Recommendations and FY21 Meeting Agenda	J. Rempe, Rempe and Associates, LLC
00:25	NRA-Japan Activities -1F3 Hydrogen Explosion -Contamination of Unit 1 and 2 Hardened vent line -Cs trapped on the under-surface of shield-plugs -Unstable operation of SRVs under SBO conditions	M. Yasui, JNRA
01:10	TEPCO Update and Discussion -SGTS filter train investigation results on Units 3/4 and investigation plan for Unit 1/2 -Preparations for Upcoming 1F1 PCV Investigation and 1F2 Fuel Debris Removal -1F3 Pressure Behavior during RCIC Operation, related to NRA activities	S. Mizokami M. Cibula S. Mizokami, TEPCO Holdings
02:20	Break	All
02:30	JAEA Update and Discussion -OECD Projects -R&D Activities (experimental Activities to Investigate 1F Accident Progression, Fuel Debris Examinations, Remote System and Sensing Technology, etc.) -Overview of New Hot Cell Facilities	T. Washiya M. Kurata S. Koyama A. Nakayoshi, JAEA
03:15	Adjourn	All
DOE Staff:	Brenda Peacock, IT issues: email: Brenda.Peacock@nuclear.energy.gov Mark Yale, Slides: email: Mark.Yale@nuclear.energy.gov Meeting Materials: https://1drv.ms/u/s!ApliCoIj18LBgexI4WswypMkjTbkqA?e=fA0ucP	

Reactor Safety Technology Expert Panel Forensics Meeting

Agenda November 10-12, 2020

Virtual Meeting (WebEx: Must Respond to Registration Email for Each Day)
Times relative to start time: 7:00 AM U.S. Eastern Time (UTC-07:00; 9:00 PM Tokyo)

Wednesday, November 11, 2020

00:00	Topic 1 – Component and System Performance
-1F3 Pressure Behavior- Based on TEPCO and JNRA Information
(RCIC, SRVs, HPCI, rupture disc performance)
-Revisions to Information Requests (as needed)	K. Robb, ORNL/
J. Gabor, Jensen
Hughes
00:30	Topic 2 – Dose Estimates and Radiation Surveys
- Radiation Concentration and Spectra Evaluation in Reactor
Buildings
- New Cs-137 Measurements (in preparation of SGTS removal
operations)
- Dose and Radiation Measurements
- Pool Scrubbing Experimental Efforts
-Revisions to Information Requests (as needed)	N. Andrews,
D. Luxat,
SNL
01:00	Topic 3 - Core Debris Location Evaluations
-Coolant suspension tests (TEPCO)
-Evaluations of Cs-137 levels to infer debris location (JNRA)
-Thoughts about Laboratory-2 capabilities (JAEA)
-Revisions to Information Requests (as needed)	M. Farmer, ANL
01:30	<i>Break</i>	All
01:45	Topic 4 – Combustible Gas Effects
-Insights/Comments on 1F3 Hydrogen Combustion based on JNRA
Evaluations
-Revisions to information requests (as needed)	W. Luangdilok
H2Technology/FAI
02:15	Topic 5 – Operations & Maintenance
-RCIC Testing Update
-Emergency Procedures Operational Aspects
-Effective Decontamination Methods (Example EPA Report)
-Nuclear Safety Reform Measures and Risk Reduction (Measures
Implemented at Other Plants)
- Information Needs (as needed)	R. Bunt
P. Ellison
B. Williamson,
BWROG
03:15	Adjourn	All
- DOE Staff:* Wendy Funk, IT issues:
email: Wendy.Funk@nuclear.energy.gov
Mark Yale, Slides:
email: Mark.Yale@nuclear.energy.gov

Meeting Materials:

<https://1drv.ms/u/s!ApliCoIj18LBgexI4WswypMkjTbkqA?e=fA0ucP>

Reactor Safety Technology Expert Panel Forensics Meeting

Agenda November 10-12, 2020

Virtual Meeting (WebEx: Must Respond to Registration Email for Each Day)
Times relative to start time: 7:00 AM U.S. Eastern Time (UTC-07:00; 9:00 PM Tokyo)

Thursday, November 12, 2020

00:00	Related EPRI Activities -MAAP Updates -Results from BWR Uncertainty Studies -Desired examination information to reduce uncertainties	K. Voelsing, EPRI
00:45	Recent MELCOR Development Activities - Water level instrumentation models - Uncertainties - Desired examination information to reduce uncertainties	D. Luxat N. Andrews L. Albright, SNL
01:30	<i>Break</i>	All
01:45	Updates to Information Requests and Next Steps -Updates to Information Requests -Action items and schedule	Rempe, Rempe and Associates, LLC
03:15	<i>Adjourn</i>	All

DOE Staff: Patrick M. Hughes, IT issues:
email: Patrick.Hughes@nuclear.energy.gov
Tedros Abraham, Slides:
email: Tedros.Abraham@nuclear.energy.gov

Meeting Materials:
<https://1drv.ms/u/s!ApliCoJj18LBgexI4WswypMkjTbkqA?e=fA0ucP>

A.2. November 10-12, 2020 Meeting Attendees

Name	Organization
Lucas Albright	Sandia National Laboratories
Don Algama	U.S. Nuclear Regulatory Commission
Phillip Amway	Exelon Corporation, BWR Owners Group
Nathan Andrews	Sandia National Laboratories
Sudhamay Basu	McGill Engineering Associates
Frances Bolger	Electric Power Research Institute
Randolph Bunt	Southern Nuclear Company, BWR Owners Group
Michel Call	U.S. Nuclear Regulatory Commission
Michal Cibula	TEPCO Holdings JAEA
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
Jeffrey R. Gabor	Jensen Hughes
Randall O. Gauntt	Gauntt Technical Safety Associates, LLC
Harutaka Hoshi	Japan Nuclear Regulation Authority
Shungo Ito	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Kohei Iwanaga	Japan Nuclear Regulation Authority
Karen Kirkland	Texas A&M University
Kenneth Klass	Talen Energy, BWR Owners Group
Wataru Kikuchi	Japan Nuclear Regulation Authority
Tatsuro Kobayashi	TEPCO Holdings
Retsu Kojo	Japan Nuclear Regulation Authority
Jun Kondo	Embassy of Japan
Shinichi Koyama	Japan Atomic Energy Agency
Steven Kraft	Kraft-Contente, LLC
Masaki Kurata	Japan Atomic Energy Agency
Emily Larson	U.S. Nuclear Regulatory Commission
Richard Lee	U.S. Nuclear Regulatory Commission
Roy Linthicum	PWR Owners Group
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
Paul McMinn	Fauske and Associates, LLC

Name	Organization
Shinya Mizokami	TEPCO Holdings JAEA (CLADS)
Akira Nakayoshi	Japan Atomic Energy Agency
Junichi Nakano	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Matt Nudi	Electric Power Research Institute
Takuya Okamoto	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Chan Paik	Fauske and Associates, LLC
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
David Skeen	U.S. Nuclear Regulatory Commission
Masami Taira	Japan Nuclear Regulation Authority
Hirokazu Tanoue	Japan Nuclear Regulation Authority
Kelli Voelsing	Electric Power Research Institute
Richard Wachowiak	Jensen Hughes
Tadahiro Washiya	Japan Atomic Energy Agency
Paul Whiteman	Framatome
Bill T. Williamson II	Tennessee Valley Authority, BWR Owners Group
Masaya Yasui	Japan Nuclear Regulation Authority

APPENDIX B. Information Requests

As described in Section 1.1, a primary objective of the U.S. forensics effort is to provide 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 FY2015. Every year, these information requests are reviewed and as appropriate, updated. Appendix B.1 presents the current version of these information requests, which are organized into tables for each location (e.g., the reactor building, the PCV, and the RPV).

The requests discuss why the information is needed and how obtained information will be used. U.S. participants also factored in experience from TMI-2 examinations in selecting requests. Hence, this appendix only lists information requests judged to be beneficial for defueling efforts and for operations and safety. In addition, participants considered information provided by representatives from TEPCO Holdings and other Japanese organizations (e.g., JAEA, NDF, and NRAJ). Requests were updated to reflect new data obtained from recent 1F examinations and planned future investigations.

Since FY2015, several new information requests were added and the status of several U.S. information requests was modified. Requests that have been addressed are shaded in light gray. In several cases, however, new information is being provided for “completed” requests. Organizations within Japan are deploying new technologies, such as the gamma imaging camera, not available at the time when these initial requests were developed. Hence, some “completed” information requests (e.g., RB-3a, RB-3c, RB-4, RB-5, RB-11, PC-1, PC-3d, PC-5, PC-6) acknowledge the potential for additional information to be obtained.

Several 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.
- **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 PC-17, PC-18, PC-19, PC-20, and PC-22 are combined (see Table B-14).

[†] 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)	<ul style="list-style-type: none"> 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 additional information may not be obtained).
RB-2	Photos/videos of HPCI System after disassembly (1F1, 1F2, and 1F3)	<ul style="list-style-type: none"> 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 additional information may not be obtained and 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)	<ul style="list-style-type: none"> 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.).	Initial request completed; additional information may be available due to new technology.	TEPCO Holdings has obtained information (Dose rate distribution measurement around SGTS filter was performed for 1F3 and 1F4. Visual inspection inside RB was performed from view of integrity of structures for 1F4). If additional images are
RB-3b	Photos/videos of damaged walls and structures (1F3)	<ul style="list-style-type: none"> 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	obtained as part of D&D activities, please include reference length scales (or information about component dimensions) of damaged components, if possible. In particular, if D&D strategy allows, additional photos of the shield plugs for 1F1, 1F3, and 1F4 are of interest.
RB-3c	Photos/videos of damaged walls and structures (1F4)	<ul style="list-style-type: none"> 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.).	Initial request completed; additional information available may be due to new technology.	When shield plugs are removed, time lapsed videos during removal are requested. Photos after debris removal are also of interest. 1F1 and 1F4 shield plug surveys have been completed and images provided.
RB-4	Photos/videos of damaged walls and components and radionuclide surveys (1F2)	<ul style="list-style-type: none"> Cause of depressurization. Cause of H₂ 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.).	Initial request completed; additional information may be available due to new technology.	TEPCO Holdings has dose distribution information. In addition, NRAJ completed gamma camera investigation of 1F2 refueling floor as independent investigations.

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)	<ul style="list-style-type: none"> Leakage path identification. Accident progression benchmarks. 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. 1F1 WW vent line 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. NRAJ completed independent surveys and is sharing results (including the use of the gamma imaging capability). Additional gamma camera images of 1F3 WW vent valves are of interest. New 1F1/1F2 SGTS shared stack sample evaluations should include detection of metals (Ca, Si, Mg, and Al) in concrete oxides.
RB-6	Radionuclide surveys and sampling of ventilation ducts (1F4)	<ul style="list-style-type: none"> Isotope concentration could be used for determining source of H₂ 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 becomes 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)	<ul style="list-style-type: none"> • 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.	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)	<ul style="list-style-type: none"> • 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.	<p>Images obtained by TEPCO Holdings have been archived per request of NRAJ for Unit 4 (see NRAJ website). TEPCO has published report on Units 5 and 6 and on Daini. TEPCO Holdings will review and provide additional images of interest.</p> <p>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.</p> <p>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.</p> <p>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).</p>

Table B-1. Information requests for the reactor building

Item	What/How Obtained	Why	Benefit /Use	When	Status
RB-9	a) DW Concrete Shield Plug Radionuclide surveys (1F1, 1F2, and 1F3 - after debris removed)	<ul style="list-style-type: none"> To understand leakage 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 and shared. 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). TEPCO plans 1F2 shield plugs surveys by March 2021 and will share obtained information.
	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)	<ul style="list-style-type: none"> Potential leakage paths for RN and hydrogen release.^b To develop lessons learned regarding seal performance under high radiation/high temperature 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 NRAJ investigations or other 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)	<ul style="list-style-type: none"> Potential leakage paths for RN and hydrogen release. To develop lessons learned regarding penetration performance under high radiation/high temperature 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 and TEPCO Holdings 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, 1F3, and 1F4 containment hardpipe venting pathway, SGTS and associated reactor building ventilation system	<ul style="list-style-type: none"> To assess performance of SGTS under high temperature and radiation conditions.^d To develop lessons learned regarding their performance under high radiation/high temperature conditions Accident progression benchmarks. 	Improved AM strategies (Plant improvements). Improved understanding of events, assist D&D efforts.	Initial request completed; additional information may be available due to new technology.	<p>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. TEPCO is planning to robotic investigation of 1F1/1F2 SGTS room by March 2021.</p> <p>1F2 and 1F3: Photos and dose rate of SGTS trains and venting pathway available. NRAJ has obtained additional information using new technologies (e.g, gamma cameras).</p>
RB-12	Photos/ videos at appropriate locations near identified leakage points in 1F1, 1F2, and 1F3.	<ul style="list-style-type: none"> 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.	Initial request completed; additional information may be available due to new technology.	<p>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).</p> <p>NRAJ has and is sharing additional information using new technologies (e.g, gamma cameras).</p>

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	<ul style="list-style-type: none"> 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.	<p>1F2: TEPCO Holdings has some visual information related to 1F2 MSIV.</p> <p>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).</p>
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.	<ul style="list-style-type: none"> Presence of Ca/Al/Si/Mg 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 include detection of metals (Ca, Si, Mg, and Al) in concrete oxides.

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	<ul style="list-style-type: none"> 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 NRAJ, and are being shared by NRAJ. TEPCO Holdings has obtained and shared some dose rate measurements in the area around the surge tank. and RCW water sampling information. Additional sampling is planned during 2021.

- 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 *SGTS*. 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.	<ul style="list-style-type: none"> Determine how head lifted. Determine peak temperatures. Look for indicators of degradation due to high radiation and high temperature hydrogen, including hydrogen-induced embrittlement. 	<p>AM Strategies; What happened with respect to the leak path; better simulations for training. Assist D&D efforts.</p> <p>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.</p>	Now (initial data and photos) and later (if head removed).	<p>The US is interested in comparing procedures used by the US and TEPCO. Information obtained by TEPCO Holdings has been archived per request of NRAJ. TEPCO Holdings observed tensioning is done based on gap requirements; and no records are available. TEPCO Holdings has obtained photos indicating:</p> <p>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.</p> <p>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.</p> <p>1F3: Deformation of part of shield plug was observed, which was found in the visual inspection after removing building rubble.</p> <p>Additional photos, similar to those obtained for 1F1 shield plug, may be possible as advanced technologies become available and/or as radiation levels decrease.</p> <p>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.</p>
PC-2	Photos/videos and radionuclide surveys/sampling of IC (1F1)	<ul style="list-style-type: none"> 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)	<ul style="list-style-type: none"> 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)	<ul style="list-style-type: none"> 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 provided results from examinations of initially available debris samples within the PCV. The US requests that future debris sample examinations include detection of metals (Ca, Si, Mg, and Al) in concrete oxides.
	c) Photos/video, RN surveys, and sampling of debris and water samples near the pedestal wall and floor (1F1-1F3)	<ul style="list-style-type: none"> 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)	<ul style="list-style-type: none"> 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 advanced technologies, such as ultrasonic tomography systems could be deployed. 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)	<ul style="list-style-type: none"> 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	<ul style="list-style-type: none"> 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	<ul style="list-style-type: none"> 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 or in results from advanced technology evaluations, such as the gamma cameras, 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)	<ul style="list-style-type: none"> 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 or in results from advanced technology evaluations, such as the gamma cameras, 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	<ul style="list-style-type: none"> • 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	<p>TEPCO Holdings completed some examinations and recalibrations; no additional examinations are planned. If additional information becomes available, it will be shared.</p> <p>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.</p> <p>In 1F2, it was found SLC injection tube in RPV was stuck, which indicates blockage by molten core.</p> <p>-New thermocouple was inserted into nearby N-10 nozzle to reinforce RPV temperature monitoring in Oct. 2012.</p> <p>-Beforehand SLC line integrity was confirmed by injecting water and monitoring discharge pressure change.</p> <p>-Pressurized water of about 7MPa could not penetrate SLC line into RPV.</p>
PC-8	Examinations and operability assessments of 1F1, 1F2, and 1F3 ex-vessel sensors and sensor support structures ^d	<ul style="list-style-type: none"> • 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 	<p>BWR and possible PWR equipment qualification life; better qualifications for training.</p> <p>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.</p>	Completed, but images of penetrations associated with PCV pressure sensors are of interest.	<p>No additional operability assessment planned, but additional visual information may become available.</p> <p>TEPCO Holdings has completed some evaluations and recalibrations. TEPCO Holdings provided additional information regarding sensor qualification envelop and conditions exposed to during the accident.</p>

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	<ul style="list-style-type: none"> 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	<ul style="list-style-type: none"> 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	<ul style="list-style-type: none"> 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 NRAJ. 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	<ul style="list-style-type: none"> 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	<ul style="list-style-type: none"> 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	<ul style="list-style-type: none"> 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	<ul style="list-style-type: none"> 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	<ul style="list-style-type: none"> 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.	<ul style="list-style-type: none"> • 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 include detection of metals (Ca, Si, Mg, and Al) in concrete oxides.
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.	<ul style="list-style-type: none"> • 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 include detection of metals (Ca, Si, Mg, and Al) in concrete oxides.

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	<ul style="list-style-type: none"> • Identification of material could provide an indicator of peak structure temperatures and potential for structure failure. • Possible model improvements. 	<p>Assist D&D efforts for determining debris location.</p> <p>Modeling improvements for ex-vessel holdup have been implemented in MAAP and informed accident management strategies and risk assessment metrics.</p>	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'	<ul style="list-style-type: none"> • 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. 	<p>Assist D&D efforts for recriticality prevention, debris stabilization, locating fuel-containing materials, and debris removal and storage.</p> <p>Improved accident management strategies.</p>	Now.	<p>TEPCO Holdings is also interested in this. Future robot examination may obtain such samples.</p> <p>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 include detection of metals (Ca, Si, Mg, and Al) in concrete oxides. In addition, the US suggests that future examinations provide information about the presence of short-lived fission product isotopes.</p>
PC-21	Images from examinations in 1F3 X-53 penetration	<ul style="list-style-type: none"> • 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)	<ul style="list-style-type: none"> • 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. 	<p>Assist D&D efforts for recriticality prevention, debris stabilization, locating fuel-containing materials, and debris removal and storage.</p> <p>Potential modeling improvements for debris coolability during MCCI and inform accident management strategies and risk assessment metrics.</p>	Now and later	<p>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.).</p> <p>The US requests that future sample examinations include detection of metals (Ca, Si, Mg, and Al) in concrete oxides.</p>

- 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.
- 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.
- 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.
- 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.
- 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.
- 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.	<ul style="list-style-type: none"> • Code assessments. • 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.	<ul style="list-style-type: none"> • Code assessments. • 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.	<ul style="list-style-type: none"> • Code assessments. • Possible model improvements (for predicting peak temperatures, displacement, 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.	<ul style="list-style-type: none"> 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.
	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.				
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.	<ul style="list-style-type: none"> 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.	<ul style="list-style-type: none"> • Code assessments. • 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.	<ul style="list-style-type: none"> • Code assessments. • 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.	<ul style="list-style-type: none"> • Code assessments. • 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.	<ul style="list-style-type: none"> • Code assessments. • 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).	<ul style="list-style-type: none"> • 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.).	<ul style="list-style-type: none"> • 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

<ul style="list-style-type: none"> 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):
<p>RB-9b: Photos/videos and dose surveys around mechanical seals and hatches and electrical penetration seals</p> <p>RB-10: Photos/videos of 1F1 (vacuum breaker), 1F1, 1F2, and 1F3 PCV leakage points (bellows and other penetrations)</p> <p>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.</p>
<ul style="list-style-type: none"> Benefits - Safety, Operational, Economic, D&D, or other benefits:
<p>Safety - Desired for improving reactor safety analysis models and accident management.</p> <p>Operational - Provides for weak link assessment of penetration capacity under high radiation/high temperature conditions.</p> <p>Economic - Provide insight into seal performance capability; could be used to adjust maintenance and inspection</p> <p>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.</p>
<ul style="list-style-type: none"> Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:
<p>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.</p>
<ul style="list-style-type: none"> Methods/Tools Needed to Collect Information or Data:
<ul style="list-style-type: none"> 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)
<ul style="list-style-type: none"> Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:
<p>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</p>
<ul style="list-style-type: none"> Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)
<p>Identification of actual penetration characteristics (e.g. geometry, seal material) may be needed to apply observations to other units.</p> <p>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.</p> <p>U.S. industry should develop a list of high interest penetrations/areas because of maintenance benefits and provide to TEPCO Holdings.</p> <p>Tabletop exercises with operation and reactor safety experts should be conducted to develop potential penetration failure scenario list.</p>

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

<ul style="list-style-type: none"> 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):
<p>RB-15: Reactor Building Closed Cooling Water system (RCW) inspection 1F1</p> <p>Water level measurement of RCW.</p> <p>Dose survey around RCW surge tank.</p> <p>Images of the RCW system inside of containment are desired if obtained during D&D.</p>
<ul style="list-style-type: none"> Benefits - Safety, Operational, Economic, D&D, or other benefits:
<p>Safety - Desired for understanding 1F1 accident progression and the potential role of the RCW during an accident.</p> <p>Operational - Provides insights about component performance under high radiation/high temperature conditions.</p> <p>D&D - Could influence D&D efforts by identifying leakage locations.</p>
<ul style="list-style-type: none"> Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:
<p>TEPCO Holdings and the U.S. expert panel have identified the potential 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.</p>
<ul style="list-style-type: none"> Methods/Tools Needed to Collect Information or Data:
<ul style="list-style-type: none"> 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.
<ul style="list-style-type: none"> Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:
<p>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.</p> <p>Long-term, images of the RCW inside of containment (sump heat exchanger piping) may be obtained during D&D or its planning.</p>
<ul style="list-style-type: none"> Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)
<p>Identifying the design water volume of the RCW system.</p>

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

<ul style="list-style-type: none"> 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):
<p>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.</p> <p>This information is of interest both prior to event and during debris removal.</p> <ul style="list-style-type: none"> 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. Additional photos, similar to those obtained for 1F1 shield plug, may be possible as advanced technologies become available and/or as radiation levels decrease.
<ul style="list-style-type: none"> Benefits - Safety, Operational, Economic, D&D, or other benefits:
<p>Operational - Provides insights about degradation under high radiation/high temperature conditions.</p> <p>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.</p>
<ul style="list-style-type: none"> Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:
<p>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.</p>
<ul style="list-style-type: none"> Methods/Tools Needed to Collect Information or Data:
<ul style="list-style-type: none"> Mostly photographic
<ul style="list-style-type: none"> Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:
<p>When reactor head is opened for decommissioning purposes.</p>
<ul style="list-style-type: none"> Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)
<p>None</p>

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

<ul style="list-style-type: none"> 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):
<p>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)</p> <p>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.</p>
<ul style="list-style-type: none"> Benefits - Safety, Operational, Economic, D&D, or other benefits:
<p>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.</p>
<ul style="list-style-type: none"> Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:
<p>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.</p>
<ul style="list-style-type: none"> Methods/Tools Needed to Collect Information or Data:
<ul style="list-style-type: none"> 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
<ul style="list-style-type: none"> Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:
<p>Near-term and/or later (Sample removal possible within next 2 years).</p>
<ul style="list-style-type: none"> Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)
<p>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.</p> <p>Evaluation of this information may require composition information for concrete (to distinguish between sand and concrete).</p>

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

<ul style="list-style-type: none"> • 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):
<p>PC-3b: PCV liner examinations (photos/videos and metallurgical exams); (1F1-1F3)</p> <p>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 thermochemical 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.</p>
<ul style="list-style-type: none"> • Benefits - Safety, Operational, Economic, D&D, or other benefits:
<p>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.</p>
<ul style="list-style-type: none"> • Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:
<p>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.</p>
<ul style="list-style-type: none"> • Methods/Tools Needed to Collect Information or Data:
<ul style="list-style-type: none"> • Irradiation resistant high-resolution imaging system. • Laser imaging systems to reconstruct liner distortion and/or ablation profiles.
<ul style="list-style-type: none"> • Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:
<p>Near-term and/or later.</p>
<ul style="list-style-type: none"> • Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)
<p>None.</p>

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

<ul style="list-style-type: none"> 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):
<p>PC-3c: Photos/video, RN surveys, and sampling of pedestal wall and floor (1F1-1F3).</p> <p>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.</p>
<ul style="list-style-type: none"> Benefits - Safety, Operational, Economic, D&D, or other benefits:
<p>Determining the pedestal wall and floor structural integrity as well as RN distributions is important for safety evaluations of D&D activities.</p>
<ul style="list-style-type: none"> Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:
<p>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.</p>
<ul style="list-style-type: none"> Methods/Tools Needed to Collect Information or Data:
<ul style="list-style-type: none"> 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.
<ul style="list-style-type: none"> Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:
<p>Near-term and/or later (Sample removal possible within next 2 years).</p>
<ul style="list-style-type: none"> Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)
<p>None.</p>

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

<ul style="list-style-type: none"> 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):
<p>PC-3d: Concrete erosion profile; photos/videos and sample removal and examination (1F1-1F3)</p> <p>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, porosity, permeability, etc.) of the debris and concrete. A sufficient number and size 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. New technologies, such as ultrasonic tomography systems, are available for deep penetration scanning of concrete structures and assessing delamination from rebar. If it can be shown that it is possible to implement them within the PCV (and it is possible for such system to work in high radiation conditions), these systems might be useful for imaging core melt ablated into concrete.</p>
<ul style="list-style-type: none"> Benefits - Safety, Operational, Economic, D&D, or other benefits:
<p>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.</p>
<ul style="list-style-type: none"> Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:
<p>Debris characterization parameters, such as morphology, particle size distribution, porosity, and permeability, are important for removal, drying and storage activities. From the viewpoint of understanding debris coolability, porosity measurements would be very valuable. Even more valuable would be permeability measurements to evaluate the extent that the porosity is interconnected. This is important not only from the viewpoint of understanding coolability, but also from the viewpoint of being able to dry out the debris before it is canned for long-term storage. In addition, the above information is important for benchmarking and reducing uncertainty in models for predicting molten core concrete interaction (MCCI) phenomena.</p> <p>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.</p>
<ul style="list-style-type: none"> Methods/Tools Needed to Collect Information or Data:
<ul style="list-style-type: none"> 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.
<ul style="list-style-type: none"> Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:
<p>Near-term and/or later (Sample removal possible within next 2 years).</p>
<ul style="list-style-type: none"> Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)
<p>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.</p> <p>Evaluation of this information may require composition information for concrete (to distinguish between sand and concrete).</p>

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

<ul style="list-style-type: none"> 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): 	
<p>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)</p> <p>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.</p>	
<ul style="list-style-type: none"> Benefits - Safety, Operational, Economic, D&D, or other benefits: 	
<p>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.</p>	
<ul style="list-style-type: none"> Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed: 	
<p>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.</p>	
<ul style="list-style-type: none"> Methods/Tools Needed to Collect Information or Data: 	
<ul style="list-style-type: none"> Irradiation resistant high-resolution imaging system Hot cell elemental analysis system D&D cutting and removal tools able to extract materials 	
<ul style="list-style-type: none"> Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3: 	
<p>Near-term and/or later (Robotic examinations underway).</p>	
<ul style="list-style-type: none"> Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations) 	
<p>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.</p>	

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

<ul style="list-style-type: none"> 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):
<p>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.</p>
<ul style="list-style-type: none"> Benefits - Safety, Operational, Economic, D&D, or other benefits:
<p>BWR AM Strategies (plant mods, etc.) and better simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.).</p>
<ul style="list-style-type: none"> Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:
<p>To determine RPV failure mode.</p> <p>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].</p> <p>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.</p>
<ul style="list-style-type: none"> Methods/Tools Needed to Collect Information or Data:
<ul style="list-style-type: none"> 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)
<ul style="list-style-type: none"> Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:
<p>Near-term and/or later.</p>
<ul style="list-style-type: none"> Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)
<p>None.</p>

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

<ul style="list-style-type: none"> 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):
<p>PC-6: Visual inspections of 1F1, 1F2, and 1F3 SRVs including standpipes in the torus and drywell (interior valve mechanisms)</p>
<ul style="list-style-type: none"> Benefits - Safety, Operational, Economic, D&D, or other benefits:
<p>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.).</p>
<ul style="list-style-type: none"> Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:
<p>To determine if there was any failure of SRVs and associated piping.</p>
<ul style="list-style-type: none"> Methods/Tools Needed to Collect Information or Data:
<ul style="list-style-type: none"> Irradiation resistant high-resolution imaging system (including new technologies, such as the gamma camera applications deployed by NRAJ)
<ul style="list-style-type: none"> Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:
<p>Near-term and/or later.</p>
<ul style="list-style-type: none"> Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)
<p>None.</p>

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

<ul style="list-style-type: none"> 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): <p>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.</p> <p>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.</p> <p>PC-19: Chemical analysis (XRF) of black material discovered on CRD exchange rail in 1F2 at X-6 penetration location</p> <p>PC-20: Chemical analysis of black material on 'existing vertical wall structure' in 1F1 picture outside pedestal doorway</p> <p>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).</p> <p>These five information requests focus on the chemical composition of materials observed 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. Evaluations should also consider data to address recriticality concerns and debris cutting, drying, and storage requirements [e.g., debris composition and morphology (e.g., crack, gaps, porosity, permeability, particle size and shape distribution)].</p>
<ul style="list-style-type: none"> Benefits - Safety, Operational, Economic, D&D, or other benefits: <p>Required for D&D; desired for improving reactor safety analysis models and accident management.</p>
<ul style="list-style-type: none"> Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed: <p>As emphasized in Table B-10, debris characterization parameters are important for removal, drying and storage activities and for benchmarking and reducing uncertainty in models for predicting molten core concrete interaction (MCCI) phenomena. Additional information is also needed to 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.</p> <p>Additional PC-19 analysis can be used to assess the extent of in-vessel cladding oxidation. PC-18 evaluations can be used to determine if core debris is present at the X-100B location, thereby providing insights on the extent of core debris relocation which is also a critical uncertainty impacting accident management strategies. Knowledge gained from these analyses will be used to enhance these strategies. Data from PC-17 can be used to determine if sediment composition 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.</p>
<ul style="list-style-type: none"> Methods/Tools Needed to Collect Information or Data: <ul style="list-style-type: none"> Hot cell elemental analysis system and/or in-situ elemental analysis using Laser Induced Breakdown Spectroscopy (LIBS) and/or X-ray Fluorescence (XRF). Robotics systems for collecting samples, and for probing / determining the sediment (loose material) depth at X-100B. 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, given the current low overall level of decay heat.
<ul style="list-style-type: none"> Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3: <p>Near-term and/or later.</p>
<ul style="list-style-type: none"> Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations) <p>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.</p>

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.[68] 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.[69] 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.[69]

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

<ul style="list-style-type: none"> 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):
<p>PC-21: Images from examinations in 1F3 X-53 penetration</p> <p>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.</p> <p>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, porosity, water permeability, particle shape and size distribution, 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.</p>
<ul style="list-style-type: none"> Benefits - Safety, Operational, Economic, D&D, or other benefits:
<p>Required for D&D; desired for improving reactor safety analysis models and accident management.</p>
<ul style="list-style-type: none"> Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:
<p>As emphasized in Table B-10, debris characterization parameters are important for removal, drying and storage activities and for benchmarking and reducing uncertainty in models for predicting molten core concrete interaction (MCCI) phenomena. Additional information is also needed to 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. Improved knowledge will be used to enhance accident management strategies.</p> <p>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.</p> <p>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.</p>
<ul style="list-style-type: none"> Methods/Tools Needed to Collect Information or Data:
<ul style="list-style-type: none"> 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
<ul style="list-style-type: none"> Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:
<p>Near-term and/or later (Sample removal possible within next 2 years).</p>
<ul style="list-style-type: none"> Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)
<p>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.</p> <p>Evaluation of this information may require composition information for concrete (to distinguish between sand and concrete).</p>

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

<ul style="list-style-type: none"> • 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):
<p>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.</p> <p>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.</p>
<ul style="list-style-type: none"> • Benefits - Safety, Operational, Economic, D&D, or other benefits:
<p>Improved AM strategies; Improved simulations for training.</p>
<ul style="list-style-type: none"> • Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:
<ul style="list-style-type: none"> • Code assessments and validation of current structural yielding modeling used in codes • Possible model improvements.
<ul style="list-style-type: none"> • Methods/Tools Needed to Collect Information or Data:
<ul style="list-style-type: none"> • Visual inspection
<ul style="list-style-type: none"> • Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:
<p>Near-term and/or later.</p>
<ul style="list-style-type: none"> • Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)
<p>None.</p>

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

<ul style="list-style-type: none"> 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):
<p>RPV-4:</p> <p>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.</p> <p>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.</p> <p>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.</p> <p>d) Photos/videos of 1F1, 1F2, and 1F3 core plate and associated structures</p> <p>RPV-5</p> <p>a) Remote mapping of 1F1, 1F2, and 1F3 core through shroud wall from annular gap region (muon tomography and other methods, if needed)</p> <p>b) Mapping of end state of core and structural material (visual, sampling, hot cell exams, etc.)</p> <p>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.</p>
<ul style="list-style-type: none"> Benefits - Safety, Operational, Economic, D&D, or other benefits:
<p>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.</p>
<ul style="list-style-type: none"> Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:
<p>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.</p>
<ul style="list-style-type: none"> Methods/Tools Needed to Collect Information or Data:
<ul style="list-style-type: none"> 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
<ul style="list-style-type: none"> Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:
<p>Near-term and/or later (Sample removal possible within next 2 years).</p>
<ul style="list-style-type: none"> Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)
<p>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.</p>

APPENDIX C. Selected FY21 Presentations

This appendix contains presentations from participants wishing to have them published in this report. Presentations are organized according to the meeting agenda found in Appendix A.1 (i.e., introductory presentations are provided in Appendix C.1; presentations describing recent results from Japanese organizations are found in Appendix C.2; U.S. topic areas presentations are found in Appendix C.3; and presentations related to systems analysis codes development and application activities, are in found in Appendix C.4). Section 2 highlights key points discussed during these and other presentations during the meeting.

C.1. Introductory Presentations

C.1.1. U.S. DOE Forensics Program Overview

U.S. DEPARTMENT OF

ENERGY

Nuclear Energy

US Efforts to Support Examinations at Fukushima Daiichi – Expert Panel Meeting

Damian Peko

DOE Program Manager, US Department of Energy

November 10, 2020

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

U.S. Efforts Coordinated with Phase 2 Roadmap D&D Activities and Other Programs

3

Tangible U.S. Benefits from Forensics Program

- Continued safe and economic operation of the existing fleet is essential aspect for public acceptance of advanced SMRs (LWR and non-LWRs).
 - Previous ANL severe accident safety testing and analysis programs credited by NEI with saving the BWR fleet over \$1B and keeping several plants operational in response to the NRC hardened vent order following the reactor accidents at Fukushima
 - Acceptance of post-Fukushima U.S. industry approach (FLEX, reliance on existing plant instrumentation with water addition and early venting) allowed plants within the US to continue to operate.
 - It behooves us to be aware of on-going international efforts to deploy new SSCs in advanced designs to address issues observed at Daiichi (e.g., SSCs to monitor hydrogen production, water level, IC system performance, etc.).
- Forensics effort continues to evaluate new examination information related to system performance and phenomena observed during events at Daiichi (e.g., the hydrogen explosions, the performance of SSCs, etc.).
 - Information needed to address knowledge gaps in severe accident understanding and improve operator guidance for severe accident management.
 - In the case of the RCIC system, safety insights from this information may be used to support changes in operation and operator guidance with safety and economic benefits.
- New technologies being deployed at Daiichi that could reduce plant maintenance costs and personnel exposures.

C.1.2. U.S. NRC Severe Accident Program Overview

Severe Accident & Source Term Research International Collaboration & Code Development

Reactor Safety Technology Expert Panel Forensics Meeting
Virtual Platform
November 10-12, 2020

Hossein Esmaili
Office of Nuclear Regulatory Research

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 MELCOR and MACCS
 - MELCOR and MACCS user group meetings including MELCOR Code Assessment Program (MCAP) in U.S., European User Group Meeting (EMUG) in Europe, and Asian User Group Meeting (AMUG) in Asia
 - Share experience with the NRC codes, identify code errors, perform code assessments, and identify areas for code improvements, experiments, and model development

Regulatory Applications

3

Role of Codes for Source Term

4

U.S. NRC Current Severe Accident Collaborations

- OECD/NEA projects
 - Preparatory Study on Analysis of Fuel Debris (**PreADES**) (2017-2021) [Assist in Fukushima Decontamination & Decommissioning]
 - Analysis of Information from Reactor Buildings and Containment Vessels of Fukushima Daiichi Nuclear Power Station (**ARC-F**) (2019-2021): [Continued engagement of international SA/ST experts in Fukushima activities]
 - Hydrogen Mitigation Experiment for Reactor Safety II (**HYMERES**) (2017-2021) [experimental data for CFD analysis of hydrogen behavior in containment]
 - NEA Reduction of Severe Accident Uncertainties (**ROSAU**) (2020-2024) [molten core concrete interaction experiments]
 - Experiments on Source Term for delayed Releases (**ESTER**) (2020-2024)
- **DENOPI**: NRC participation in the IRSN project to investigate fuel coolability during a loss of coolant accident in SFP (2014-2021)
- **MUSA**: NRC participation in Management and Uncertainties of Severe Accidents (MUSA) Under the European Commission's (EC's) Horizon 2020 project (2019-2024)

5

MELCOR Development

- Accident Tolerant Fuel (ATF)
 - Panel of international severe accident experts PIRT and address significant phenomenological issues and improve MELCOR. Consider HBU and HALEU
 - QUENCH-ATF: Experiments for ATF cladding materials in the QUENCH facility at Karlsruhe Institute of Technology (KIT)
 - Source Term calculations
- Code Modernization
 - Improvements in code structure to strengthen performance, numerics, capability, and maintenance
 - 5 year program - Incremental changes to different code packages for hydrodynamic, in-vessel and ex-vessel accident progression and fission product release and transport
- Non-LWR Advanced Reactor Design
 - "NRC Non-Light Water Reactor Vision and Strategy: Volume 3: Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis," Revision 1, January 2020
 - Source term demonstration calculations using MELCOR for three representative designs including a heat pipe reactor, a gas cooled reactor, and a molten salt cooled pebble bed reactor
 - "NRC Non-Light Water Reactor Vision and Strategy: Volume 5: Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle," Draft, November 2020

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C.1.3. U.S. DOE Forensics Program FY2020 Findings and Recommendations

U.S. DEPARTMENT OF
ENERGY

Nuclear Energy

FY20 Findings and Recommendations and FY21 Meeting Agenda

Joy Rempe
Technical Lead, Rempe and Associates, LLC

November 10, 2020

FY20 Activities

U.S. DEPARTMENT OF
ENERGY

Nuclear Energy

FY20 Findings and Recommendations

-
- **Finding 1** - The complexity of D&D activities at Daiichi is unprecedented.
 - **Finding 2** - Information obtained from the affected reactors continues to be implemented in severe accident management strategies and systems analysis code evaluations.
 - **Associated 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** - 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.
 - **Associated 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.

Nuclear Energy

FY20 Findings and Recommendations

- **Finding 5** - 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.
 - **Associated 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** - 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.
 - **Associated 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 events will be limited and models of physical phenomena that provided such insights will be lost.

3

Nuclear Energy

FY20 Findings and Recommendations

- **Finding 7** - 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.
 - **Associated 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** - The Terry™ Turbine project offers the potential for important reactor safety insights and reduce operating costs.
 - **Associated 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

Remaining Meeting Topics*

- NRA-Japan Activities - Yasui
- TEPCO Activities Update and Discussion- Mizokami and Cibula
- JAEA Activities Update and Discussion– Washiya, Kurata, Koyama, and Nakayoshi
- Topic Area Lead discussions:
 - Area 1 - Components/System Performance – Robb and Gabor
 - Area 2 - Radionuclide Surveys/Sampling – Andrews
 - Area 3 - Core Debris Location Evaluations – Farmer
 - Area 4 – Combustible Gas Effects - Luangdilok
 - Area 5 – Operations and Maintenance – Bunt, Ellison, and Williamson
- Related EPRI Activities – Voelsing
- Recent MELCOR Development Activities – Luxat, Andrews, and Albright
- Updates to Information Requests and Next Steps - All (led by Rempe)

*See Agenda for Presentation Schedule

Link to FY21 Agenda, Viewgraphs, FY20 Report & Draft FY21 Report (when available):

<https://1drv.ms/u/s!ApliColj18LBgexl4WswypMkjTbkqA?e=fA0ucP>

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C.2. New Information from Japan

C.2.1. Nuclear Regulation Authority Japan Investigations and Analysis

Presentation Material

Investigation and analysis of the TEPCO's Fukushima Daiichi NPS Accidents

November, 2020
Masaya, YASUI

The Secretariat of the Nuclear Regulation Authority Japan
(Tentative English Translation)

★This presentation bases on the on-going discussion. The NRA has not reached any condition yet. Therefore, the information shared can not be understood as solid.

Topics 1 Introduction

- The NRA started the 2rd phase investigation/analysis of Fukushima Daiichi Accidents on September 2019
- The NRA team has investigated various areas which could not be reached beforehand.
- Major investigation areas are:
 - Unit1: Operating floor(Radiation measurement and Gamma camera)
 - Unit2: Operating floor(Radiation measurement and Gamma camera)
Smear sampling of RB
 - Unit3: 3rd and 4th floor(Video and Radiation measurement)
 - Unit4: Whole RB with Laser-scanning

(URL)

<Meeting URL>

7th: <https://www.youtube.com/watch?v=ERb0DYc3slc>

8th: https://www.youtube.com/watch?v=_s7N6cT4KmY

9th: https://www.youtube.com/watch?v=_s7N6cT4KmY

10th: https://www.youtube.com/watch?v=Qx8RvEgw_E4

11th: <https://www.youtube.com/watch?v=rexVt67Mqlg>

12th: <https://www.youtube.com/watch?v=wo2llfnjoW4>

13th: <https://www.youtube.com/watch?v=N6tPXHVrATk>

14th: <https://www.youtube.com/watch?v=wZbj0w2iifU>

3

Unit1 Operation Floor (5F)

4

Unit1 Operation floor by Gamma-camera picture

Photo and Data by NRA 2020/7/30

Unit2 Operation floor by Gamma-camera

Data by NRA 2020/1/30

Smear-sampling

Sampling

sample

Photo by NRA 2020/10/8

7

Unit3 R/B 3rd floor

Photo by NRA 2020/9/18

8

Unit3 R/B 4rd floor

9

Unit4 R/B 2nd floor ceiling

10

Result of Laser Scanning Unit4 4th floor Ceiling

11

<RB Video>

(1) Unit2 R/B Video (2020.Oct 8 by NRA)

<https://www.youtube.com/watch?v=xgN2kCO9n9k>
<https://www.nicovideo.jp/watch/so37644822>

(2) Unit3 R/B 3rd Floor video Video (2020.Sept 18 by NRA)

<https://www.youtube.com/watch?v=2ogtUCOnQDg>
<https://www.nicovideo.jp/watch/so37669375>

(3) Unit3 R/B 4rd Floor Video (2020.Sept 18 by NRA)

<https://www.youtube.com/watch?v=bQ-ROid740c>
<https://www.nicovideo.jp/watch/so37669649>

(4) Unit1 R/B SGTS Room Video (2020.Oct.9 by NRA)

<https://www.youtube.com/watch?v=7KX-mmwh93c>
<https://www.nicovideo.jp/watch/so37669753>

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Contamination in Hardened Vent System

13

14

Contamination of Main Stack

測定前における推定と測定についての留意事項

□ 測定・分析結果の精度と相対比較の妥当性

高レンジGM管: 0.01~9999mSv/h
低レンジGM管: 0.01~9999μSv/h
精度±20% (セシウム137 20度C)

Topics 3

Possible Cs Capture with shield-plugs

17

* 1 The shield-plugs had been dis-located (may be the time of Hydrogen Explosion.)

* 2 The figure might be changed

18

Unit1 Operation Floor (5F)

(参考) プラグ内部調査での撮影映像 (1)

2017/3/30 廃炉汚染水対策チーム会合事務局会議資料
「福島第一原子力発電所1号機オペレーティングフロア調査結果(中間)について」抜粋

- 目的：調査カメラ（能動スコープ）をウェルプラグがずれて隙間が開いている箇所から内部へ挿入し、ウェルプラグの状態を確認する
- 調査期間：2016年12月～2017年2月
- 調査結果：ウェルプラグ上段/中段に加え、下段も正規の位置からずれていることを確認した。

ウェルプラグイメージ図（西側）

中段北側より撮影(北⇒南)

19

Topics 4

Unstable Operation of SRVs under SBO condition

20

RPV Pressure (Unit 3)

21

4. (構造図)

出典: 第14回事故分析検討会資料2-3

B-1の区間で生じるS/RVの中間開状態

Unit3 Hydrogen Explosion (Multi-Step Phenomenon Hypothesis)

23

Multi-Step Phenomenon Hypothesis

- ① Prior hydrogen explosion occur in R/B (may be in 4th floor); the northwest -ward force is imposed.
- ② A rift appears in the southeast roof of R/B and fire outbreak.
- ③ Apart from this, second explosion/combustion occur and smoke rise from the middle of the roof.
- ④ Remaining hydrogen burns and smoke blows up.

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Picture of Unit1 and Unit3 Hydrogen Explosion

25

Locations of Seismic Observation Recorders within the Fukushima Dai-ichi Nuclear Power Station

Maximum acceleration recorded by the seismic observation recorder at the occurrence of the explosion at Unit 1, Unit 3 and Unit 4 R/B (Gal)

	Unit 1 Explosion March 12 at 15:36			Unit 3 Explosion March 14 at 11:01			Unit 4 Explosion March 15 at 6:12		
	South-north direction	East-west direction	Vertical direction	South-north direction	East-west direction	Vertical direction	South-north direction	East-west direction	Vertical direction
Observation point A	81	52	120	14	15	21	2	2	2
Observation point B	284	129	138	45	18	28	4	3	3
Observation point C	2,320 [※]	2,392 [※]	1,956	115	158	490	11	9	6
Observation point D	102	91	231	36	51	173	6	7	11
Observation point E	39	22	26	26	24	30	5	5	11

※ This value is only for information as it was out of range (2,000 Gal) of the sensor. (※)

Japan Government Accident Investigation Excerpt from Final Report 2021.07.23

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Japan Government Accident Investigation Excerpt from Final Report 2021.07.23

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Thank you for your Attention

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C.2.2. Tokyo Electric Power Holdings, LLC (TEPCO Holdings) Investigations

C.2.2.1. SGTS Investigations

SGTS filter train investigation results on Units 3/4 and investigation plan for Unit 1/2

TEPCO

2020/11/10

MIZOKAMI Shinya

Tokyo Electric Power Company Holdings, Inc.

1. Introduction

TEPCO

- TEPCO started the investigation in R/B from this JPFY.
- In this activity, information will be collected before losing its evidence related to accident due to aging.
- As a first location of the investigation, we selected the SGTS room where SGTS filter train and rupture disc are located

2. Past investigation results and objective

TEPCO

Past results of dose rate measurement around SGTS train filter

- Unit 1 : On 2011, more than 5[Sv/h] was confirmed in front of SGTS room
(This is the first trial to obtain information inside unit 1 SGTS room.)
- Unit 2 : On 2014, about 1[Sv/h] was confirmed at SGTS train. And there was no contamination around rupture disc.
- Unit 3 : On 2011, dose rate was measured around SGTS filter train.
- Unit 4 : On 2011, dose rate was measured around SGTS filter train.
(Dose rate in unit 3,4 were lower than those of unit 1,2. Unit 4 measured dose rate indicated the back flow from unit 3.)

Objectives of the investigation

- Obtaining the information inside SGTS room of unit 1 to 4, such as dose rate or contamination level at filter train, vent line, rupture disc.
→The results help to understand the FP behavior during PCV hard vent and unexpected consequence, such as back flow to SGTS filter train and inflow of FP and hydrogen to R/B.
- Smear sampling of contaminated filters inside filter train in unit 3 and 4
→By conducting smear sample analysis, it is expected to obtain the information related to the chemical composition and chemical form etc..

2

4. Instrument and carrier employed in the investigation

TEPCO

- Robot for unit 1,2 investigation : PackBot

- camera, dosimeter, gamma imager, etc. can be carried.

- Gamma imager :

- Image of panoramic picture and radiation dose distribution can be obtained as an integrated information.

Example of the panoramic image by gamma imager.

3

5. Schedule

TEPCO

- Investigation started in August 2020 and will complete February 2021.
- Preliminary investigation for unit 1 and 2 completed by end of August.
- Investigation for unit 3 and 4 are now underway. Dose rate distribution was obtained by gamma imager. We are preparing to open the SGTS filter train.

Investigation item	2020/8	2020/9	2020/10	2020/11	2020/12	2021/1	2021/2
Mock-up test					モックアップ		
Unit 1 investigation	Preliminary investigation	Develop a detail operation plan			Main investigation		
Unit 2 investigation	Preliminary investigation	Develop a detail operation plan				Main investigation	
Unit 3 investigation		Main investigation					
Unit 4 investigation			Main investigation				

4

6. Measurement result in SGTS room in unit3 at point 4

TEPCO

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6. Measurement result in SGTS room in unit 3 at point 6

TEPCO

6. Measurement result in SGTS room in unit 3 around rapture disc

TEPCO

7. Plan of another investigation for unit 3 and 4

TEPCO

- Smear sampling of SGTS filter
 - By opening the door of SGTS filter train, smear of filters, HEPA, Charcoal, bag filter, will be sampled.
- Radionuclide analysis for smear sample
 - Radionuclide analysis in Fukushima Daiichi
 - ✓ Total alpha analysis
 - ✓ Total beta analysis
 - ✓ Gamma spectrometry (Cs-137, Cs-134, Co-60, etc.*)
 - Radionuclide analysis in outside Fukushima Daiichi
 - ✓ (TBD) Analysis focusing on the particle :
 - SEM-EDS,WDS
 - TEM-EDS, electron diffraction for special particle(s)
 - ✓ (TBD) Nuclide inventory analysis :
 - ICP-MS
 - Alpha spectrometry
 - Sr-90 quantitative analysis

*:Co-60,Ru-106,Ag-108m,Sb-125,Ba-133,Cs-134,Cs-137,Eu-154,Am-241

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9. Result of preliminary investigation in unit 2 (8/27)

TEPCO

9

8. Result of preliminary investigation in unit 1 (8/25)

TEPCO

1 0. Main investigation on unit 1 and 2 SGTS room

TEPCO

- By considering the preliminary investigation results, measuring point by using gamma imager will be determined.
- Measuring the dose rate on the surface of filter train by robot
- Only in unit 2, measuring the dose rate of rupture disc and vent line

Direction of back flow

- × : measuring point by γ imager
- : Confirmed available area for robot investigation
- : Low dose rate area (human work)

1 1. Summary

TEPCO

- Unit 1 : Preliminary robot investigation was completed. The area where robot can crawl was specified.
It was confirmed that values of measured dose rate near SGTS filter train were larger than other points.
- Unit 2 : Preliminary robot investigation was completed. The area where robot can crawl was specified.
It was confirmed that values of measured dose rate near SGTS filter train were larger than other points same as previous investigation in 2014.
- Unit 3 : Dose rate distribution in SGTS room was obtained by using gamma imager. It was confirmed the contamination on piping corresponding to back flow to SGTS filter.
We are now in preparation for smear sampling on SGTS filter.

12

TEPCO

以下、参考資料

13

(参考) 過去の主な調査状況 (1号機)

TEPCO

- SGTS室北側の扉付近にて5Sv/h以上と高い線量率を確認 (2011年)

1号機SGTS室付近の線量率測定結果

14

(参考) 過去の主な調査状況 (2号機)

TEPCO

- ラブチャディスク周辺に汚染のないことを確認 (2014年)

2号機ラブチャディスク周辺の線量率測定結果

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3. 調査の概要

TEPCO

- 1～4号機のSGTS室内において、フィルタトレイン周辺、格納容器ベント配管を中心に、室内から広範に画像、線量情報を取得する。
- また、可能であれば3、4号機の調査では、フィルタトレインを開放し、内部の様子を確認および汚染物試料の入手（スミア採取）を試みる。
- 3、4号機のSGTS室内は線量が低く、作業効率性等ふまえ、人力で調査を実施する。一方、1、2号機のSGTS室内は、線量が高いことが確認されているため、ロボットにて遠隔で調査を実施する。

4号機SGTSフィルタトレインの様子

16

（参考）過去の主な調査状況（2号機）

TEPCO

- SGTSフィルタトレインA系では1Sv/h以上と高い線量率を確認（2014年）

*) フィルタトレイン表面から約20cm離れた位置(フィルタ中心面より約65cm)で測定した線量値

2号機SGTSフィルタトレインA系の線量率測定結果

17

(参考) 過去の主な調査状況 (2号機)

TEPCO

- SGTSフィルタトレインB系では1Sv/h弱と高い線量率を確認 (2014年)

2号機SGTSフィルタトレインB系の線量率測定結果

18

(参考) 過去の主な調査状況 (3号機)

TEPCO

- SGTSフィルタトレイン周辺の線量率を確認 (2011年)

3号機SGTSフィルタトレイン周辺の線量率測定結果

19

(参考) 過去の主な調査状況 (4号機)

TEPCO

- SGTSフィルタトレイン周辺の線量率から3号機の格納容器ベントガスが4号機原子炉建屋内に逆流した徴候を確認 (2011年)

4号機SGTSフィルタトレイン周辺の線量率測定結果

C.2.2.2. PCV Internal Investigations

Preparations for Upcoming 1F1 PCV Investigation and 1F2 Fuel Debris Retrieval

2020/11/10
CIBULA Michal
Tokyo Electric Power Company Holdings, Inc.

PCV Internal Investigation

- PCV Internal Investigations provide crucial information serving as a basis for decision making regarding future debris retrieval, transportation and storage, as well as source of information for further accident development clarification (Forensics)
 - ① Primary Containment Vessel (PCV) internal investigation (ongoing)
(including sampling)
 - ② Small-scale retrieval of fuel debris (Unit 2 planned for FY2021)
 - ③ Large scale retrieval of fuel debris (gradually increasing scope of ②)
- Collection and analysis of small amounts of deposit samples from the bottom of the PCVs of Units 1 and 2 scheduled for FY2020 and FY2021, respectively
- Sample analyses in JAEA facilities (Ibaraki Prefecture)
- On-site sample analysis facility (JAEA) currently under construction (for future samples)

1

Unit 1 PCV conditions

- Assumed fuel debris distribution: almost all in the bottom of the PCV
- Feb ~ May 2015: reactor core investigation using muon analysis method
- Apr 2015: 1st floor grating investigation using shape-changing robots (outside pedestal)
- Mar 2017: Investigation of lower part of PCV by hanging sensor equipment from the 1st floor grating (outside pedestal)
- No investigation inside the pedestal has been conducted yet

2

Unit 1 PCV Internal Investigation – Outline

- Investigation conducted through X-2 penetration – double door worker access
- Submersible robot used for various investigations

3

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

Unit 1 PCV Internal Investigation – Access route construction

1. Installation of 3 isolation valves

Completed on 2019.5.10

2. Outer door drilling

Completed on 2019.5.23

3. Inner door drilling(AWJ)

Completed on 2020.4.22

4. Cutting of obstacles inside PCV (ongoing)

5. Installation of 3 guide pipes

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

4

Unit 1 PCV Internal Investigation – Access route construction

- Removal of obstacles to allow introduction of ROV
- Handrail cutting in progress
- Grating cutting in progress

Image of grating cutting scope
(view from the RPV side)

Image of cut handrail (vertical part)

Front of X-2 penetration before
the accident
(from inside of PCV)

Image of grating cutting scope
(RPV is on the right side)

Image of cut grating

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

5

Unit 1 PCV Internal Investigation – ROV route inside PCV

- ROV introduced through X-2, following two main routes
- Guide rings (preventing cable entanglement) attached to the jet deflectors
- PCV investigations conducted by ROV passing through guide rings
- Investigation inside the pedestal area will be attempted
- Sampling will be conducted outside of pedestal area

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Unit 1 PCV Internal Investigation through the X-2 penetration

- Six types of submersible ROVs to provide following functions.

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

※ The thickness of sediments and whether or not fuel debris is present, and the thickness of that debris, are unknown. They are shown in the concept diagram for explanatory purpose.

7

Unit 1 debris distribution – estimation of conditions inside

RPV/PCV (Points of focus in future surveys)

Proposed schedule

TEPCO

- Countermeasures for interference with PLR instrumentation piping being reevaluated

Unit 2 PCV conditions

- Assumed fuel debris distribution: part of fuel debris on the pedestal floor
- Mar ~ Jul 2016: reactor core investigation using muon analysis method
- Feb 2018: PCV internal investigation using guide pipe through X-6 penetration
- Feb 2019: Investigation in the lower area of pedestal through X-6. Grasping and moving pebble-like deposits
- Rock-like deposits could not be moved

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Unit 2 PCV Internal Investigation through the X-6 penetration

- Access/investigation device being developed with purpose of ascertaining the distribution of structures and deposits inside the pedestal
- Low water level allows use of X-6 penetration for investigation
- Monitoring of radioactive substances concentration in dust to confirm no impact on the surrounding environment due to leakage of gases from the PCV

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

11

Unit 2 Fuel Retrieval preparation

0. Investigation of deposits in X-6, removal of monitoring instruments

- Investigation of conditions inside X-6, removal of monitoring instruments

1. Isolation room installation

- Isolation room installed prior to the hatch opening

2. X-6 hatch opening

- Using designated equipment

3. Removal of deposits in X-6

- [Low · High pressure water] pushing the deposits
- [AWJ] cables cutting
- [Pushing tool] Pushing cables

4. Installation of arm equipment

5. Internal investigation/trial retrieval work

① PCV internal investigation using arm equipment

② Trial debris retrieval using arm equipment

Tip of the debris retrieval equipment

(Note)

- Isolation valve : installed to separate inside/outside of the PCV
- AWJ (Abrasive water jet) : A processing machine using abrasives mixed with high pressure water in order to improve machinability

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Outline of deposits investigation inside X-6 of Unit 2

- Removal of deposits from X-6 penetration to allow access of arm-type device (Low · High pressure water, AWJ, pushing device, etc.)
- Investigation of the deposits to be reflected in mock-up/removal procedure
- Investigation contents
 - Contact investigation (Checking the state before/after the contact with finger device)
 - Distribution investigation (3D scan)

Full view of the device

Mock-up of contact investigation

Outline of contact investigation unit

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Results of deposits investigation inside X-6 of Unit 2 (1/3)

TEPCO

- Change of deposit shape after contact confirmed
- deposits are not fixed onto the penetration wall

14

Results of deposits investigation inside X-6 of Unit 2 (2/3)

TEPCO

- Cables movability confirmed
- Cables are not fixed onto the penetration wall

15

Results of deposits investigation inside X-6 of Unit 2 (3/3)

TEPCO

- deposits, cables condition and X-6 hatch hollowed part※ visually confirmed

①deposits conditions

②deposits conditions

③Cables and hollowed part conditions

④Cables conditions

※Investigation location is provisional

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Spray device installation

IRID
TEPCO

- Generation of radioactive dust expected during X-6 deposits removal and obstacles removal operations
- Spraying device installation in the nearby X-53 penetration
- Expansion of the current penetration from $\phi 50$ to $\phi 130$ in order to install the device ($\phi 100$)
- Suppression of dust dispersion during the operations using X-6

Unit 2 R/B 1st floor
Penetrations location

Current state

Current opening ($\phi 50$)

Expansion

Expansion to $\phi 130$
by processing machine
(hole saw)

Image of X-53 closure flange expansion

Image of spraying
(View from inside PCV)

Tip of spraying device

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Unit 2 PCV Internal Investigation through the X-6 penetration

- Access/investigation device with wand on the end for attaching suitable measuring equipment depending on investigation purpose
- Collecting small amounts of deposits from inside the pedestal with a small sampling tool attached to the wand (planned for 2021)

Proposed schedule

	2020				2021
	9	10	11	12	
Equipment manufacturing and mock-up					
Investigation of deposits inside X-6 penetration		▽10/28 Contact investigation ▽10/30 3D scanning			
Permanent monitoring instrumentation removal			▽Beginning of Nov. Instrumentation removal ※Restoration of monitoring instruments after completion of investigation and trial removal work		
Installation of spray device			X-53 expansion		Spray device installation
Enclosure installation X-6 hatch opening deposits removal Arm equipment installation					
Internal investigation and trial debris retrieval					

Sample analyses

- Samples taken from the surface (smear) of AWJ device for access route construction and filters from gas management system of Unit 1 sent for analysis in Ibaraki (Oct 2020) – high U, Pu content not expected in these samples
- First samples obtained in Unit 1 and Unit 2 PCV investigations will be prepared for shipping to JAEA facilities in gloveboxes within 1F site
- JAEA analysis/research facility within the 1F site currently under construction

Evaluation item (examples)	Analysis item and method (examples)	Use of analysis results (examples)
Property 1) Fuel-based objects • composition of (U, Zr)O ₂ • content rate of Gd 2) Structural material based objects • (Fe oxidizing, content of B)	Composition: SEM-EDS, TEM-EDS, ICP-MS Microstructure: TEM-EDS, Electron diffraction	• Formation mechanism is hypothesized from the microstructure of samples. ⇒ Fuel debris removal, container/storage facility design • Content rates of Gd and B are important for subcriticality control
FP distribution (concentration of Cs, Sr)	FP composition: Radiation analysis, ICP-MS Burn-up: TIMS (quantitative Nd148)	• Initial amount of FP generated is estimated from burn-up rate to evaluate remaining percentage of FP. ⇒ Container/storage facility design
Dose rate	Radiation analysis	• Used to reduce exposure during fuel debris handling
Machinability	Vickers hardness etc.	• Used to design cutting jig

- SEM: Scanning Electron Microscope, EDS: Energy Dispersive X-ray Spectroscopy, TEM: Transmission Electron Microscope, ICP-MS: Inductively Coupled Plasma Mass Spectrometry, TIMS: Thermal Ionization Mass Spectrometry

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C.2.2.3. 1F3 Behavior during RCIC Operation

Reactor Safety Technology Expert Panel Forensics Meeting

Fukushima Daiichi unit 3 plant behavior during RCIC operation

2020/11/10

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1. Introduction

1

- JNRA pointed out that TEPCO can not explain the unit 3 plant behavior during RCIC operation in NRA investigation meeting.
- In that period, RPV pressure kept higher than SRV reset value and less than SRV activation value. However, it seems there are some cyclic SRV discharges with small range.

- The operation mode of RCIC was quite far from normal operation. In order to save DC battery consumption, both of RPV injection line and return line to CST were lined up at that time. Therefore, there is no way to know how much water was injected to RPV.
- We conducted the RELAP calculation to capture the qualitative characteristics during RCIC operation.

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2. Record of the RCIC operation

Description in the shift hand-off note

2011/3/11

RCIC quick start/L-8 trip

RCIC quick start/L-8 trip	
15:06/15:25	RCICクイックスタート/L-8トリップ
RCIC start up RPV water level S/C water level	
16:03	RCIC「起動」 Rx水位: -500mm S/Cレベル: 150mm
	RCICミニフロー弁「閉」 RCIC minimum flow valve "close"
16:13	HPCI、RCIC 水源切替防止リフト
16:16	RCIC注入開始 Rx水位: -900mm
RCIC injection start RPV water level	

2011/3/12

象	11:36	RCIC トリップ	炉水位: +200mm (W)
発		RCIC trip	RPV water level(wide range)

S/C water level is increasing because minimum flow valve is "open".

https://www.tepco.co.jp/nu/fukushima-np/plant-data/f1_4_Nisshi3_4.pdf

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3. Way of RCIC operation by unit 3 operator

[Securing reactor water level via RCIC]

- Operators implemented measures to avoid automatic shutdown due to high reactor water level and to conserve batteries needed for operation control. This was done to avoid battery use due to RCIC startup/shutdown and to secure stable reactor water levels.

- One of the measures to prevent RCIC automatic shutdown due to high reactor water level was assembling a line via operation of the RCIC control panel to pass water through both the reactor injection line and test line (used for periodic function tests, loops from the water source Condensate Storage Tank (CST) back to the CST), performed while monitoring reactor water level from the Main Control Room (MCR). This set the scope of water level adjustment and secured water level.

RCIC control panel (taken at a later date)

- Two personnel were assigned to reactor water level monitoring, while another two were assigned to RCIC operation. Under this framework, both parties reported conditions to each other while working. Preparations toward smooth HPCI startup as the next injection measure following RCIC shutdown were carried out (e.g., HPCI control panel operation switches tagged).

- In order to limit valve and Flow Indication Controller (FIC) operation as a battery conservation measure, flowrate was set via test line valve openness adjustment and FIC, to ensure gradual changes in reactor water level. The method used involved repeatedly changing flowrate setting values (within approx. 75% to 100% of standard flowrate (25.2L/s) each time the reactor water level neared the upper or lower limits of water level adjustment scope.

RCIC FIC (taken at a later date)

RCIC and HPCI reactor injection line overview

2012/06/20 TEPCO 1F accident investigation report Attachment 2 P113,114

https://www.tepco.co.jp/en/press/corp-com/release/betu12_e/images/120620e0101.pdf

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4. Decay heat

5. Expected SRV opening mechanism under lower RPV pressure

After SBO, no additional N2 supply to SRV N2 accumulator.

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Relief valve mode:
If $P_p + P_A < P_N + P_R$, SRV open

→In case RPV pressure NOT reach reset pressure, SRV open/close depending on the RPV pressure.

In any cases, because of the low vapor generation, it is expected that SRV will close before full open due to RPV pressure decrease.

Safety valve mode
If $P_A < P_R$, SRV open

→In case of spring temperature increase, SRV open lower than set point due to decrease of Young's module.

6. Calculation results by using RELAP

Note

- We conducted tuned calculation to follow RPV pressure trend.
- Large pressure decreases of around 0.3 to 2MPa are due to RCIC water injection.
- We set higher steam bleed flow rate than design as continuous steam flow from RPV.
- Even employing the assumption of much steam consumption by RCIC, cyclic small discharge was required to reproduce the RPV pressure trend.

TEPCO

6. Calculation results by using RELAP (energy balance)

TEPCO

7. Summary

- We studied the 1F3 RPV pressure behavior during RCIC operation.
- Decay heat in the period was not so large. But, it is difficult to reproduce RPV pressure behavior without additional steam discharge.
- It is highly possible that the small amount of steam was cyclically discharged to S/C through SRV, which was not fully opened.
- The calculation results shows the possibility of large RPV pressure decrease due to RCIC cold water injection.
 - There is a tendency that pressure decrease was over estimated comparing to measured RPV pressure. It is supposed that such behavior is due to the characteristics of the system code which consists of node and junction.
- It is expected that 1F3 RPV pressure behavior during RCIC operation can be explained by half open SRV activation depending on the RPV pressure and RCIC water injection under particular operation employed during 1F accident.

C.2.3. Japan Atomic Energy Agency (JAEA) Activities

CLADS R&D activities on decommissioning for Fukushima Daiichi NPS

November 10-12, 2020

Collaborative Laboratories for Advanced Decommissioning Science (CLADS)
Japan Atomic Energy Agency (JAEA)

International Research Institute for Nuclear Decommissioning (IRID)

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.

1. Outline of CLADS R&D activities (T.Washiya)
2. 1F accident progression (M.Kurata)
3. Fuel debris characterization (S.Koyama)
4. Preparation of 1F fuel debris analysis (S.Koyama)
5. OECD/NEA Projects;
 - PreADES (A.Nakayoshi)
 - ARC-F (A.Nakayoshi)
 - TCOFF (M.Kurata)

1. Outline of CLADS R&D activities

2. 1F accident progression (recent key issues)

3. Fuel debris characterization

4. Preparation of 1F fuel debris analysis

5. OECD/NEA Projects;

- PreADES
- ARC-F
- TCOFF

2

3

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Waste Management Division

DL: Y. Koma

Remote System and Sensing Technology Division

DL: I. Wakaida

Fukushima Environmental Evaluation Research Division

DL: K. Iijima

Fukushima Environmental Monitoring Division

DL: T. Magara

DL: Division Leader

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Fuel Debris Characterization

- ◆ Characterization and Analysis on Fuel Debris
- ◆ Behavior Analysis of Radioactive Micro-particles
- ◆ Dose Evaluation and Material Accountancy
- ◆ R&D on Hydrogen Safety Engineering

Simulated MCCI product

Analysis on Radioactive Particles

Evaluation of dose rate distribution

H2 generation Mechanism

Accident Progression and Damaged Core Evaluation

- ◆ Accident Progression and Degradation Modeling
- ◆ Fission Product Chemistry
- ◆ R&D on Corrosion Prediction and Mitigation

Control blade and sim. fuel bundle degradation

Possible effects on corrosion

Image of reactor after the accident

Waste Management

- ◆ Collecting of physical/chemical data on various waste
- ◆ R&D on treatment and disposal for solid radioactive wastes
 - Stabilizing slurry in the multi-nuclide removal equipment
 - Corrosion of vessel from cesium absorption apparatus
 - Development of safety technology for storing waste

Sampling of waste specimen

Example of simulated conditioned waste

Remote System and Sensing Technology

- ◆ R&D for Radiation Imaging, 3D Visualization Tech.
 - Measuring the spatial distribution of radiation in the high dose-rate environment by Compton Camera
 - Development of portable Gamma-Ray Imagers
- ◆ R&D on various analytical technologies

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Radiation Imaging and Instrument Group

- Portable Gamma-Ray Imagers** (Compton Cam., etc.)
 - To measure the spatial distribution of radiation in the high dose-rate environment.
- Remote radiation monitoring by using a drone**
 - Compton Camera
 - 3D model created by a laser scanner
 - Reconstruction image of radiation
- Detection of Gamma-ray and Neutron Fluxes from Fuel Debris in the High Dose-Rate Environment** (MSGC, High-energy gamma-ray Det., etc.)
 - To measure neutrons and high-energy gamma-rays caused by fuel debris.

Remote Technology and Robotics Group

- Environment Data Collection Technology** (Simultaneous Localization Mapping (SLAM), High Permeability Laser in the Contaminated Water, etc.)
- Environment Data Management Technology** (Environment Model Construction, Relational Spatiotemporal Database)
- Environment Data Utilization Technology** (Integration with Virtual Reality I/F and Simulators)

Data Visualization Technology Group

- 3-D Visualization Technology of Radiation Source Distribution**
- Inverse Problem Analysis on the Contamination Source Information Obtained from Monitoring Data**
- Radiation Risk Evaluation for the Reduction of Workers' Exposure** (Visualization by a 3D Image)

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Compton camera

Luminous → Detects all incident light → Shielding is required to have directivity → The weight has increased and the transport device has also become larger.

Measurement → Detects contaminated point by using a linear optical fiber type scintillator

Compton camera diagram: Radiation source → γ -ray → Scatter → Absorber. Angle θ is shown. Labels: Compton scatter, Photoelectric absorption.

Ensures directivity without shielding

Plastic scintillation fiber (PSF) diagram: Radiation (γ, β , neutron) → PSF → PMT or MPPC → PA. Includes Start signal, Stop signal, Delay, and Time-of-Flight circuit. Label: Convert time difference to electrical signal.

2D visualization of radiation distribution diagram: Scan the surface → PSF → 2D visualization.

Compton camera photograph: y-ray sensor, Signal processing, Optical camera.

7

Detect hotspots in the environment with a drone

Remote radiation imaging system that combines drones and LiDAR in collaboration with a local company.

Compton camera

Ref.) Japan Atomic Energy Agency, Chiyoda Technol Corp., Press release. Rapid three-dimensional visualization of radiation distribution in the environment from the sky —Development of a drone-mounted camera system— May 9, 2019

3D Measurement inside 1F Building

The Compton camera was mounted on a crawler type robot, and contaminated distribution is visualized in 3D by using 3D image processing on camera image.

Ref.) Y. Sato, Y. Terasaka, W. Utsugi et. al., Radiation imaging using a compact Compton camera mounted on a crawler robot inside reactor buildings of Fukushima Daiichi Nuclear Power Station, Journal of Nuclear Science and Technology, 56, pp. 801-808, (2019).

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Confirmed decontamination effect of new decontamination technology

Detection of leak points in contaminated water tanks

Ref) Y. Sanada, T. Yamada, Y. Sato, et. al., "Application of plastic scintillation fiber measurement technology to contaminated water management at Fukushima Daiichi Nuclear Power Station", DOI:10.11484/JAEA-Research 2016-011 (2016) (in Japanese)

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Radiation monitor for drainage channel of 1F on-site

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1. Outline of CLADS R&D activities

2. 1F accident progression (recent key issues)

- Debris behaviour in lower plenum (impact of core energy)
- Control blade degradation (formation of metallic debris)

3. Fuel debris characterization

4. Preparation of 1F fuel debris analysis

5. OECD/NEA Projects;

- PreADES
- ARC-F
- TCOFF

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- Core energy significantly affects accident progression, i.e., core material melting, slumping (relocation to lower head), heat transfer, RPV boundary failure and core material relocation to pedestal.
- Two typical situations expected in case of low/high core material energy at slumping are illustrated in Figs. (a) and (b) below respectively.
- Evaluation of core material energy for Units 2 and -3 showed a remarkable difference in core material energy and suggested these units could have had the different characteristics as shown in Figs (a) and (b) respectively.

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CLADS-MADE : Mock-up Assembly Degradation tests

Background:

- 1F Unit 2 was damaged and the metallic debris and top tie plate were found in the PCV – the mechanism of such RPV damage is not completely understood;

Objectives:

- **Debris propagation features** – study the metallic debris formation and propagation at the very first step of the 1F core degradation;
- **Materials science aspects** – investigate, depending on the previous history and accident conditions, the impact of the metallic debris on the core damage;
- **Impact of the environment** – study the core materials interactions under steam-rich and steam-starved conditions;
- **Uncover the fundamental mechanisms** – understand the transport of B during the accident and possible interactions.

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LEISAN : Large-scale Equipment for Investigation of Severe Accidents in Nuclear reactors

- Different decay heat at the core uncover $\Rightarrow$ Different heating rate
- Difference in the water level decrease $\Rightarrow$ Different amount of steam in the core
- Unknown behavior of neutron absorber (B_4C) $\Rightarrow$ Influence on debris characteristics

Simulant BWR fuel assembly

100 mm

1.2 m

Experimental condition necessary to consider 1F accident

- 0.3~1.0 K/s heating rate
- Variable steam mass flow rate
- Axial temp. gradient (ca. 500 K/m)

Other features:

- In-situ observation window (video camera installed)
- Off-gas analysis

Control blade degradation test facility

Before

After

Test material (Top)

Test material (Lower)

Test material degraded at the top ran down and clogged between channel boxes

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Top view
Height: 1200 mm

Oxidized control blade sheath directly fell down

Control blade sheath

Side view (upper)
Height: 1150-1200 mm

Molten control blade relocated and no reaction with preoxidized channel box

Side view (middle-height)
Height: 740-790 mm

Control blade melted at the core top, solidified at the lower core and formed the channel blockage

In-situ observation enhanced understanding of the:

- temperature of the control blade degradation onset;
- relocation behavior of the melt and solid debris;
- solidification temperature;

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Control blade degradation (upper part)

【Test condition】

Steam starvation condition (1F Unit 2 condition)

- Initial oxide layer: ~ 50 μm
- Heat-up rate: 0.3 K/s
- Steam flow rate: ~40 g/min

Non-uniform solidification and blockage (lower part)

Metallic melt directly fell down

- The surface oxide layer formed on Zircaloy during the normal operation limited the reaction between control blade and channel box;
 - **Control blade alone degraded and relocated non-uniformly;**
- ⇒ Possibility of a path to BWR drainage-type accident scenario!

*A. Pshenichnikov, et al., Journal of Nuclear Science and Technology, vol. 56, issue 5, 2019, p.440-453.

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1. Outline of CLADS R&D activities
2. 1F accident progression (recent key issues)
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5. OECD/NEA Projects;
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 - ARC-F
 - TCOFF

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2. Characterization of 1F fuel debris

IRID

Ex-vessel : MCCI products

MCCI products
• $(Zr,U)O_2$ -Cubic/Tetragonal
• $(Zr,U)SiO_4$
• SiO_2 , etc.

Example : VW-U1 sample (Upper crust)

Estimated results using previous vulcano-MCCI samples (JAEA-CEA collaboration)

Ref) Takano, Onozawa, Suzuki, Obata, "Fuel Debris Properties (27 A): (9) Simulated Fuel Debris and TMI-2 Debris Microhardness", Atomic Energy Society of Japan 19
2015 Autumn Meeting G39 (2015) (in Japanese)

(1) Sintering

- Same as fuel pellet fabrication
- Mixture, Crush, Pelletize, Sintering (Ar, ~1730°C)
- Additive: Gd_2O_3 , Fe_2O_3 , CaO
- With control oxygen potential

(2) Arc dissolution

- Entire uniform heating
- Required metal rich conditions
- Simulated Core Debris, MCCI product
- ~5 g / sample

(3) Heating by light collection

- Concentrating light as heat source
- Erosion under thermal gradient conditions
- Various atmosphere gas
- Small scale MCCI: Melting of $(\text{U,Zr})\text{O}_2$

Ref.) M.Takano, et.al., Characterization of solidified melt among materials of UO_2 fuel and B_4C control blade, J. Nucl. Sci. Technol. 51 (2014) 859-875.

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➤ Structure and hardness depending on the cooling rate during solidification

Ref.) M.Takano, J.Onozawa, M.Suzuki, et.al., "Fuel Debris Properties (27A): (9) Simulated Fuel Debris and TMI-2 Debris Microhardness", Atomic Energy Society of Japan 2015 Autumn Meeting G39 (2015) (in Japanese)
M.Takano, T. Nishi and N.Shirasu, Characterization of solidified melt among materials of UO_2 fuel and B_4C control blade, Journal of Nuclear Science and Technology, vol. 51, 2014, pp. 859-875.

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- ◆ Characteristics of the MCCI products “heterogeneous product” was obtained with the large-scale test.
- ◆ The appropriate selection for defueling tools was suggested based on this information.

Ref.) A. Nakayoshi, H. Ikeuchi, T. Kitagaki, et al., KNOWLEDGE OBTAINED FROM DISMANTLING OF LARGE-SCALE MCCI EXPERIMENT PRODUCTS FOR DECOMMISSIONING OF FUKUSHIMA DAIICHI NUCLEAR POWER STATION, International Topical Workshop on Fukushima Decommissioning Research, FDR 2019-1013, FDR 2019-1033, May 24-26, 2019, J-Village, Naraha, Fukushima, Japan

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- Confinement of radioactive fine particles generated in the fuel debris retrieval process is necessary.
- Migration behavior of radioactive fine particles in air and water is required.

Ref.) E. Porcheron, S. Bourroux, S. Peillon, et al., STUDY OF THE INFLUENCE OF CUTTING TOOLS ON AEROSOL EMISSION IN THE FRAMEWORK OF FUKUSHIMA DAIICHI FUEL DEBRIS RETRIEVAL, International Topical Workshop on Fukushima Decommissioning Research, FDR 2019-1033, May 24-26, 2019, J-Village, Naraha, Fukushima, Japan

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Alteration of fuel debris

- **Secondary product** (yellow) was confirmed about 10 years after the ChNPP accident.
 - It would be **change to dust** through dissolution in moisture.
 - Activation energy for oxidation of UO_2 .
 - ✓ in water: 21-42 kJ/mol
 - ✓ In air: ~109 kJ/mol
- ⇒ water(moisture) is important factor

Stability of fuel debris in 1F

Estimating mid- and long-term stability of fuel debris to reduce potential risk in 1F decommissioning

Secondary product on surface of FCM^[3]

Process of fuel debris decay by microorganisms

[1] R. V. Arutyunyan et al., Nuclear fuel in the «Shelter» encasement of the Chernobyl NPP, M.: Nauka, 2010

[2] <http://www.jsme.or.jp/pes/Research/A-TS08-08/index02.html>

[3] B.E.Burakov et al., Secondary uranium minerals on the surface of Chernobyl «LAVA», Mat. Res. Soc. Symp. Proc. Vol. 465 (1997)

1. Outline of CLADS R&D activities
2. 1F accident progression (recent key issues)
3. Fuel debris characterization
- 4. Preparation of 1F fuel debris analysis**
5. OECD/NEA Projects;
 - PreADES
 - ARC-F
 - TCOFF

- ◆ Internal investigation in PCV is conducting by TEPCO and IRID with using robots system, and valuable information has been obtained.
- ◆ Small scale debris sampling will be planed in 2021 to confirm conditions in the PCV.
- ◆ Characterization of fuel debris properties are necessary to the debris retrieval.
- ◆ JAEA is preparing the debris analysis in the hot cell facilities in Ibaraki prefecture.

Image of the fuel debris location

Unit 2 PCV investigation by TEPCO (2019/2/13)

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JAEA is constructing "Radioactive Material Analysis and Research Facility" adjacent to Fukushima Daiichi site.

Administrative building (2018.3~)

This facility has office space, meeting rooms for researchers, and mock-up apparatus for training.

Radioactive Material Analysis and Research Facility Laboratory-2 : (Under design)

This facility is dedicated hot laboratory for radioactive analysis, chemical and mechanical characterization on fuel debris and etc..

Radioactive Material Analysis and Research Facility Laboratory-1 : (Under construction)

This laboratory is designed for radioactive analysis of low and medium level radioactive rubbles and secondary wastes.

Radioactive Material Analysis and Research Facility

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< Floor Layout >

Purpose:

- Radioactive analysis of low and medium level radioactive rubbles and secondary wastes.
- It obtains technical prospects regarding treatment / disposal measures and their safety

Analysis target :

- Rubble and Secondary Waste from water treatment, and not handle nuclear materials.
- Very small amount of hazardous material or that may contain hazardous material.
- Handling dose rate, dimensions, mass
 - (A) Low-dose rate materials
1 mSv / h or less
 - (A) Medium-dose rate materials
1Sv / h or less

【出典】特定原子力施設放射性廃棄物規制検討会(第4回)資料1-1より抜粋、加工。
Ref.) Specified nuclear facility Radioactive Waste Regulation Study Group (4th), Document 1-1

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< Floor Layout >

Purpose:

- Radioactive analysis, chemical and mechanical characterization on fuel debris and etc.
- It contributes to examination of fuel debris retrieval process.

Design condition:

- Analysis target: Fuel debris, etc.
- Shape: powder, grains, mass, sludge, etc.
- Accepted amount: 5 kg or less
- Number of acceptances: ca.12 times / y

【出典】特定原子力施設監視・評価検討会(第82回)資料1より抜粋、加工。
Ref.) Specified Nuclear Facility Monitoring / Evaluation Study Group (82nd), Document 1

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2. 分析成果の反映

2.1. 概要

【出典】特定原子力施設監視・評価検討会(第82回)資料1より抜粋、加工。
Ref.) Specified Nuclear Facility Monitoring / Evaluation Study Group (82nd), Document 1

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[illegible]

LADS
Collaborative Laboratories for Advanced Distributed Sensing

- **PreADES**
- ARC-F
- TCOFF

ANL-20/79

* **PreADES** (*Preparatory Study on Analysis of Fuel Debris*) **International Joint Research Project**

- The Preparatory Study on Analysis of Fuel Debris (PreADES) project is led by **JAEA (Japan)**. It aims to:
 - collect information for improving knowledge and methodologies for fuel debris characterization
 - identify the needs for fuel debris analysis
 - prepare a future international R&D framework on fuel debris analysis.
- Status:
 - July 2017- Dec 2021 (extended)
 - 7 countries; 15 partners (all NEA, EC-JRC)
 - 5th meeting held in February 2020

Debris location (Figure source: TEPCO)

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- **Task 1: Joint study on fuel debris characterization**
 - Discuss about fuel debris conditions location in the 1F unit 1-3
(Ref. from BSAF-2, 1F, TMI-2, CheNPP, SA studies etc.)
 - Discuss about debris properties in the **location**
Properties : **Micro** and **Macro** scopes view points
- **Task 2: Identify needs and discuss future fuel debris analysis**
 - Discuss needs for analysis from;
Criticality, Confinement, Cooling function, Radiation exposure, etc.
 - Discuss major issues for sampling;
Analysis methods, Facility functions, Transportation, Accounting, etc.
- **Task 3: Planning future international R&D framework using actual fuel debris from Fukushima Daiichi.**

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1) Figure of debris' location

- Image of debris' location in reactor
* Provided by TEPCO

2) Characteristic table for micro

-Characterisation of debris from **microscopic** point of view
(Mainly, properties of single-phase material)

Category	Density	Vickers hardness	Elastic modulus	Fracture toughness	Thermal conductivity	Specific heat	Melting point	Behaviour at high temperature	Stability
Materials/Phases	(g/cm ³)	(GPa)	(GPa)	(MPa·m ^{1/2})	(W/mK)	(J/g·K)	(°C)		
(Zr,U)SO ₄	4.8	10	110	2	6	150	1500	(H)	
UO ₂ -H ₂ O	10	10	110	2	6	150	1500	(H)	
Metal									There are no data of Zr
Zr-2	7	8	110	2	6	150	1500		
U-3Hf	7	8	110	2	6	150	1500		
U-3Zr	7	8	110	2	6	150	1500		
Fe (Zr)	7	8	110	2	6	150	1500		
Others									
B ₂ O ₃									
ZrO ₂									
Fe ₂ O ₃									

3) Characteristic table for macro

-Characterisation of debris from **macroscopic** point of view
(Mainly, features of multi-phases material/mixture)

Category	Density	Vickers hardness	Elastic modulus	Fracture toughness	Thermal conductivity	Specific heat	Melting point	Behaviour at high temperature	Stability
Materials/Phases	(g/cm ³)	(GPa)	(GPa)	(MPa·m ^{1/2})	(W/mK)	(J/g·K)	(°C)		
(Zr,U)SO ₄	4.8	10	110	2	6	150	1500	(H)	
UO ₂ -H ₂ O	10	10	110	2	6	150	1500	(H)	
Metal									There are no data of Zr
Zr-2	7	8	110	2	6	150	1500		
U-3Hf	7	8	110	2	6	150	1500		
U-3Zr	7	8	110	2	6	150	1500		
Fe (Zr)	7	8	110	2	6	150	1500		
Others									
B ₂ O ₃									
ZrO ₂									
Fe ₂ O ₃									

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

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Criticality control

Location of sample		Significance of property for CC in each phase. High: 5 to Low: 1					
Region	Zone	Why (Objective / Motivation) Material properties, etc.	Decommissioning				SA
			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

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■ Joint task force for formation process & characteristic 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 characterization.
- PreADES launched Joint task force together with ARC-F & TCOFF at last meeting in February 2020.
 - ➡CEA/Cadarache;Saclay, CIEMAT, CNL, CRIEPI, KAERI, KTH, SNL, JAEA announced their participation.
- Currently, each organization is evaluating 1F sample analysis results
- Research results of each organization will be discussed at final meeting of PreADES on December 2020.

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1. Outline of CLADS R&D activities
2. 1F accident progression (recent key issues)
3. Fuel debris characterization
4. Preparation of 1F fuel debris analysis
- 5. OECD/NEA Projects;**

- PreADES
- **ARC-F**
- TCOFF

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* ARC-F

(*Analysis of Information from Reactor Buildings and Containment Vessels of Fukushima Daiichi Nuclear Power Station*)

- The project focusses on collection, compilation & management of data and information coming from examinations inside reactor buildings & containment vessels. This post-BSAF project is led by **IAEA (Japan)** with expert advice as appropriate from expert organizations.

Three main tasks:

- Refinement of analysis for accident scenarios and associated FPs
- Compilation and management of the data and information
- Discussion toward future long-term project

■ Status:

duration 3 years, January 2019- December 2021

- 12 countries; 22 partners
- Agreement finalized
- 3rd meeting held in Feb 2020

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Two Sub Tasks in Task 2

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1. Outline of CLADS R&D activities
2. 1F accident progression (recent key issues)
3. Fuel debris characterization
4. Preparation of 1F fuel debris analysis
- 5. OECD/NEA Projects;**
 - PreADES
 - ARC-F
 - **TCOFF**

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***Thermodynamic Characterization Of Fuel Debris and Fission Products based on Scenario Analysis of Severe Accident Progression at Fukushima-Daiichi NPS** (2017-2020 @NEA/NSC)

■ Background

- **Motivation:** Internationally recognised need of continuous update of materials science knowledge for fuel degradation mechanism, which applications include providing inputs to:
 - Improvement of fuel/core degradation models and FP-behaviour models, applied in **SA-codes**;
 - **Severe accident management** and **fuel debris characterization for 1F**.
- **MEXT/Japan** has proposed a 3-year joint project (2017-2019) to NEA/NSC, in order to maximise synergies with other joint projects, **TAF-ID** (@NSC), **BSAF-2**, **PreADES**, **ARC-F** (@NEA/CSNI). (Project period extended by July 2020)
- MEXT/Japan provided the entire financial funding (720 kEuros over 3 years), which accounts for project management, NEA staff support, and an international call for R&D proposals (@2018-2019).

■ Objectives

1. **Improve thermodynamic knowledge** (databases, degradation models) with a reference to SA-progression at 1F. (DB: TAF-ID, NUCLEA/MEPHISTA, HERACLES, SGTE)
 - **Benchmark calculations** under conditions experienced during 1F-accident (by referring to BSAF/BSAF2 analysis results) to identify materials science concerns of currently available thermodynamic databases and degradation models;
 - **Improvement and/or enlargement of thermodynamic databases and degradation models** through experimental and analytical knowledge, provided as in-kind contributions and achievement of TCOFF R&D call studies;
2. **Conduct joint thermodynamic evaluations** of SA-progression at in-vessel and ex-vessel phases of 1F (unit-1,2, and 3).
 - Fuel melting and molten core/fuel relocation and FP-behaviour;
 - Fuel debris chemical and phase composition;
 - Formation mechanism of materials which are being detected at 1F.

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■ Work Scopes

■ Participants

Czech: CVR
 EC: JRC-Karlsruhe
 France: CEA, IRSN
 Germany: KIT
 Japan: CRIEPI, JAEA, TITECH
 Korea: KAERI
 Netherland: Delft Univ.
 Russia: IBRAE, KRI, NITI, SPSU
 Sweden: SSM, KTH
 Switzerland: PSI
 US: NRC
 Project coordinator: NEA
 Technical advisor: JAEA

Task 1 (Corium and fuel)

- Identification of priorities on improvement / enlargement of TD on fuel debris, based on benchmark exercises
- Re-assessment/updating of thermodynamic functions and phase diagrams on corium and fuel debris
- Updating knowledge of physical chemistry of corium/MCCI, in particular on the deviations between quasi-equilibrium systems and the real phase observed in sim-tests
- Identification and improvement of decommissioning activities using the thermodynamic approach

Task 2 (Fission Products)

- Performing thermodynamic evaluations of Cs-deposits, and identification of concerns for updating related TD, including potential kinetic effects
- Updating the knowledge on long-term leaching behavior of FP
- Assessment of thermodynamic data for important radionuclides which can have a long-term radiological impacts, Sr, Ba, Pu, MA

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■ Major achievement (to be shown in TCOFF report)

(Task-1)

- Confirmation of status of materials interaction/FP-release models in SA-analysis code
- Benchmark exercise of thermodynamic databases to identify important lack of knowledge
- Improvement of fuel liquefaction model (U-Zr-O and U-Zr-Fe-O systems)
- Improvement of multi-component systems (B-containing system, concrete-containing system, etc.)
- Discussion of Fukushima-Daiichi accident progress by materials science standpoints (in particular, RPV-failure mechanism)
- Discussion of debris formation mechanism under Fukushima-Daiichi conditions

(Task-2)

- Improvement of Cs-complex, Cs-SS databases
- Improvement of semi-volatile FP systems (Sr, Ba, Mo)
- Discussion of leaching modelling

(TCOFF workshop)

- Materials Science for Severe Accident and Fukushima-Daiichi Decommissioning (July 10-12, 2019), 134 participants from 25 countries, 29 oral, 43 poster → **Expectation to TCOFF-2**

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■ Preliminary Discussion for TCOFF-2 (to be drafted by the end of 2020)

- Task-1: Prioritization of materials science topics focusing on the fuel core degradation and FP release
- Task-2: Improvement of materials science knowledge regarding fuel core degradation and FP release (conventional fuel/cladding and ATF)
- Task-3: Implementation of improved knowledge for the simulation of the accident behaviour (conventional fuel/cladding and ATF)
- Task-4: Leaching modelling
- Task-5: Training, how to use materials science modelling for the interpretation of SA phenomenology

•Potential members: TCOFF members + observers (CNL, VTT)

•Timeline (expected):

- Project agreement drafted by the end of 2020
- Determination of TCOFF-2 members (funding members + in-kind contributors)
- Signature process, in the first half of 2021

If you are interested in TCOFF-2,

contact person: Davide Costa (OECD/NEA, project coordinator) Davide.Costa@oecd-nea.org
Masaki Kurata (IAEA, project chair) Kurata.Masaki@jaea.go.jp

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Thank you for your attention.

C.3. U.S. Topic Area Presentations

C.3.1. 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 10-12, 2020

Topics

- Key questions
- Current status – November 2020
- Major observations from 2020 activities
 1. 1F3 RCS Pressure behavior during RCIC operation

Key Questions

- What visual damage has been observed in component and structures with RPV, PCV and 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?

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U.S. Efforts in Support of Examinations at Fukushima – November 2019 Meeting Notes with Updated Information Requests

ANL-19/48

January 31, 2020

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]

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**U.S. Efforts in Support of
Examinations at Fukushima –
November 2019 Meeting
Notes with Updated
Information Requests**

ANL-19/48

January 31, 2020

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

Area	IF1	IF2	IF3
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 IF1, allowing this penetration to be used for DW investigations.
- c. X-51 is an instrument pipe penetration for measuring differential pressure in IF2/IF3. The penetration is joined to the Standby Liquid Cooling (SLC) pump injection line in the DW. This penetration is designated as X-27 in IF1.
- d. X-53 is vacant for IF2 and IF3, 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 IF1. X-11 is the HPCI steam supply penetration for IF2 and IF3.

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1F3 Timeline

Event	Time	Ref
Seismic Event and Reactor Scram	0	
RCIC Starts	19 min	1,2
RCIC Trip on High Water level	38 min	2
Loss of AC Power	52 min	1,2
RCIC restarted	1.3 hr	1,2
RCIC stops	21 hrs	1,2
HPCI auto start	22 hrs	1,2
HPCI manually tripped	36 hrs	1,2
ADS	42 hrs	3

Ref 1: [Fukushima Technical Evaluation, Phase 1 – MAAP5 Analysis 1025750, April 2013](#)

Ref 2: [Fukushima Daiichi Accident Study, SAND2013-6173, July 2012](#)

Ref 3: Findings on Fukushima Dai-ichi NPP, S. Mizokami, 18th NURETH, Aug 2020

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1F3 Pressure Behavior

- RPV level control during RCIC operation
 - Operation mode to save battery power
 - Avoid cycling of pump by controlling injection via the injection and return lines

Source: Nuclear Regulatory Authority website (<https://www.nsr.go.jp/data/000325410.pdf>)

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1F3 Pressure Behavior

Source: Nuclear Regulatory Authority website (<https://www.nsr.go.jp/data/000315706.pdf>)

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1F3 Pressure Behavior

- RPV level control during RCIC operation
 - Operation mode to save battery power
 - Avoid cycling of pump by controlling injection via the injection and return lines

Source: Nuclear Regulatory Authority website (<https://www.nsr.go.jp/data/000325410.pdf>)

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1F3 Pressure Behavior

Source: Nuclear Regulatory Authority website (<https://www.nsr.go.jp/data/000315706.pdf>)

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C.3.2. Topic Area 2- Radiation Surveys and Sampling

Sandia National Laboratories

Fukushima Forensics FY21

Dose Estimates and Radiations Surveys

Nathan Andrews, Nuclear Engineer

David Luxat, Manager

SAND2020-12332 PE

Sandia National Laboratories is a multimission laboratory managed and operated by National Technology & 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.

2 New Information of Interest

- Radiation concentration and spectra evaluation in reactor buildings
- New Cs-137 measurements (in preparation of SGTS removal operations)
 - Can be used to inform knowledge fuel debris location
- Analyses of containment vent piping
- Dose and radiation measurements
 - Near bottom of vent stack (1F1/1F2)
 - Drywell head (1F1)
 - Primary containment venting pipes (1F1/1F2 and 1F3/1F4)
- Pool scrubbing experimental efforts

3 Radionuclide Concentration in Reactor Buildings

Comparison of dose rates and contamination (right)

- Unit 2 and 3 are significantly higher

Location of primary radiation source for 1F3 is the dark red line (below)

1-3号機 原子炉建屋5階オペレーションフロアの環境の相違に応じた測定方法の選択について

対象号機	1号機	2号機	3号機
シールドプラグの状態	3層とも大きなズレ 1層目に変形あり	外観上の大きなズレや損傷なし	第1層に破損 中央部に30cm陥没
5階天井、壁、柱などの破損	大規模に破損	ブローアウトパネルの脱落程度	大規模に破損
5階天井、壁、柱の汚染	汚染あり 測定への影響度不明	ほぼ一様に汚染あり 測定に影響は小さい	汚染あり 測定への影響度小
シールドプラグ表面汚染の有無	有り 除染実績なし	有り 除染実績あり 880mSv/h⇒140mSv/h (プラグ中心1.5m高さ)	有り 除染実績あり 表面5mm程度はつり 800mSv/h⇒200mSv/h (プラグ中心1.5m高さ)
表面汚染レベル(線量のみ)	平均150mSv/h(表面) 平均100mSv/h(1.2m高さ)	平均114mSv/h(1.5m高さ)	平均50mSv/h(0.5m高さ) 平均200mSv/h(5m高さ)
高線量瓦礫の有無	有り 除染なし	有り 片付け進行中	ほぼ無し
選択した測定方法	GM計数管 (東京電力)	鉛遮蔽付きAPD ・ガンマカメラ	鉛コリメータ付き 半導体検出器
測定による汚染レベルの推定値	0.1PBq	70PBq	30PBq

Source: Nuclear Regulatory Authority (<https://www.nsr.go.jp/data/000315708.pdf>)

4 Radionuclide Concentration in Reactor Buildings – 1F2 Investigation

Unit 2 refueling bay radiation photos (right)

Remote entry robot (below)

Source: Nuclear Regulatory Authority (<https://www.nsr.go.jp/data/000315708.pdf>)

5 Radionuclide Concentration in Reactor Buildings – IF1 Investigation

Robot examination photos
(right)

Shield plug dose (below)

Source: Nuclear Regulatory Authority (<https://www.nsr.go.jp/data/000315708.pdf>)

6 Insights Based-on Cs-137 Contamination Levels, IF1/IF2 SGTS

Locations examines on the
SGTS of 1F1 and 1F2.

- 14 1F2 locations
- 3 1F1 locations

Overall 1F2 has higher dose rates
and more contamination

第81回特定原子力施設監視・評価検討会 資料4-2より

測定点	10 cmでの線量当量率 mSv/h	汚染密度 Bq/cm ²
1	6	2.81E+05
2	8	8.43E+05
3	17	3.37E+06
4	26	5.90E+06
5	27	6.18E+06
6	20	5.62E+06
7	60	1.69E+07
8	150	4.21E+07
9	160	4.49E+07
10	60	1.69E+07
11	11	3.09E+06
12	4.3	9.93E+05
13	650	1.83E+08
14	400	1.12E+08
15	2	5.17E+05
16	2	4.34E+05
17	4	1.03E+06

Bq/cm²当たりの周辺線量当量率 (μSv/h per Bq/cm²)

Source: Nuclear Regulatory Authority (<https://www.nsr.go.jp/data/000319210.pdf>)

7 Insights Based-on Cs-137 Contamination Levels, IFI/IF2 SGTS

Dose rates and associated contamination levels at locations 13 and 14

Source: Nuclear Regulatory Authority (<https://www.nsr.go.jp/data/000319210.pdf>)

測定点13と14との比較

測定点の両側1.5mの合計: 0.916×10^{13} Bq

ガンマカメラ: 1.17×10^{13} Bq

測定点1と2と対応

ガンマカメラ: 測定点1対応領域 13 GBq

測定点2対応領域 51 GBq

配管中のCs-134による測定点1と測定点2の線量率がカメラから求めた放射能比と同じになる。うに下部からのBGを 5mSv/h と推定

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8 Insights Based-on Cs-137 Contamination Levels, IFI/IF2 SGTS

Top: Unit 1 Piping

Bottom: Unit 2 Piping

More contamination is present in unit 2 than in Unit 1, by an order of magnitude

排気筒直前の2号SGTS配管
垂直部 (2m): 3.07×10^{11} Bq

Source: Nuclear Regulatory Authority (<https://www.nsr.go.jp/data/000319210.pdf>)

1号機原子炉建屋近辺

垂直部 (2m): 2.21×10^9 Bq

奥の水平部 (5m): 2.90×10^{10} Bq

9 IFI WW Leakage Scenario and Associated Experiment

Insights from the BSAF Phase 2 project

DW Transient (right)

Leakage area, DF relationship (bottom-right)

Source: Nuclear Regulatory Authority (<https://www.nsr.go.jp/data/000306602.pdf>)

Fig. 15 Relationship between Overall DF and equivalent leak area

10 IFI WW Leakage Scenario and Associated Experiment

NRA-J is interested in determining why hardened vent pipe in 1F1 has higher contamination levels than 1F3.

Two causes were investigated: (1) bypassing vacuum breaker in 1F3 and (2) conditions inside the drywell were significant different at the time of venting.

- Option 1: Diagrams show vacuum breaker; leak in the vacuum breaker would bypass pool scrubbing.
- Experiments reveal possible leak paths through holes in gasket; however, actual dose measures in the areas of vacuum breaker does not indicate bypass.
- Option 2: Diagram (Left) shows fission product transport path if the RPV was not breached before venting (most likely 1F3 scenario). Diagram (Right) shows less scrubbing if RPV was breached prior to venting (most likely 1F1 scenario).

1F3: Plots of wetwell and drywell show pressure behavior between 1st and 2nd venting; pressure differences suggest RPV was intact.

Source: Nuclear Regulatory Authority (<https://www.nsr.go.jp/data/000300429.pdf>)

11 Pool Scrubbing Experiments

Experimental plan (right)

Experimental setup (below)

Source: Nuclear Regulatory Authority (<https://www.nsr.go.jp/data/000306607.pdf>)

12 Pool Scrubbing Experiments - Results

Decontamination factor (DF) as a function of the number of particles, event scenario and submerged depth within a mock suppression chamber.

Particles

Source: Nuclear Regulatory Authority (<https://www.nsr.go.jp/data/000306607.pdf>)

Site Dose Measurements

- NRA is interested in explaining why the dose rate was extremely high at the base of 1F1/1F2 vent stack. Following the accident, dose rate was 10 Sv/h

Source: Nuclear Regulatory Authority (<https://www.nsr.go.jp/data/000300439.pdf>)

Site Dose Measurements

- Site dose rate 10 days following the accident

Source: Nuclear Regulatory Authority (<https://www.nsr.go.jp/data/000300439.pdf>)

15 Entry to IFI DW Head Region

Source: Nuclear Regulatory Authority (<https://www.nsr.go.jp/data/000300424.pdf>); image courtesy of TEPCO

16 Entry to IFI DW Head Region

Source: Nuclear Regulatory Authority (<https://www.nsr.go.jp/data/000300424.pdf>); image courtesy of TEPCO

17 Entry to IFI DW Head Region

Source: Nuclear Regulatory Authority (<https://www.nsr.go.jp/data/000300424.pdf>); image courtesy of TEPCO

18 Primary Containment Venting Pipes Analysis

- NRA discussed their venting pipes evaluations; access is better to 1F1 and 1F2 exhaust pipes. NRA is performing evaluations to gain more insights about why there are areas of high and lower contamination. NRA also performing evaluations of the exhaust vent piping at 1F3 and 1F4.
- NRA also performing rupture disk evaluations. The objective is to gain insights about why the 1F2 rupture disk didn't open (as indicated by plant data). NRA is the impact of steam concentrations, suppression pool temperatures (1F3 temperature was higher) and pressures to see if they can gain insights about the effects on venting at 1F1, 1F2, and 1F3.
- Several experts wonder what caused re-pressurization at 1F3
 - SNL thought that it might be due to steam
 - TEPCO thought that it might be due to flashing.

Source: Nuclear Regulatory Authority (<https://www.nsr.go.jp/data/000292309.pdf>)

C.3.3. Topic Area 3- Debris Endstate

WE START WITH YES.

TOPIC 3 - CORE DEBRIS LOCATION EVALUATIONS

MITCH FARMER

Nuclear Science & Engineering Division
Argonne National Laboratory

Fukushima Forensics Meeting, November 10-12, 2020
Virtual Meeting (Webex)

TOPICS FOR DISCUSSION

- Coolant suspension tests (TEPCO)
 - Recent measurements for 1F3
 - Additional comments for 1F2 and implications for continued water injection
- Evaluations of Cs-137 levels to infer debris location (JNRA)
- Thoughts about Laboratory-2 capabilities (JAEA)
- Revisions to information requests

COOLANT SUSPENSION TESTS

- Coolant suspension tests have been carried out by TEPCO for Daiichi Units 1, 2, and 3 to support emergency planning.
- Unit 1: Results presented at the fall 2019 meeting and were integrated into the FY2020 report.
- Unit 2: Tests were carried out in April-May 2019 timeframe
 - Some additional comments on these tests are provided herein.
- Unit 3: Tests were recently carried out in February 2020
 - A few comments on the consistency of these results are provided herein.

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COOLANT SUSPENSION TESTS* (1F3)

- The coolant suspension test lasted 48 hours.
- This test revealed some interesting and initially unexpected results.
- As shown in the figure below, when injection was terminated, the temperatures initially increased.
- After injection was reestablished, the temperatures initially decreased, but then showed a rise to levels above that observed before injection was terminated.
- When the location of core debris and water level in the PCV are considered, these temperature rises may be due to other phenomena, such as evaporation and convection associated with injected water recontacting hot debris.

TC responses and water injection flowrate

RPV TC locations

* TEPCO: http://www.tepco.co.jp/decommission/information/committee/roadmap_progress/pdf/2020/d200227_10-i.pdf.

COOLANT SUSPENSION TESTS* (1F3)

- Similar behavior was observed based on measurements from the PCV thermocouples; see graph below.
- The decrease in temperature measured by TC TE-16-003 (T3) may be understood if one considers that it was located near the surface of the water.
- As water injection ceased, the water surface level dropped, causing the junction to rise above the water level where it was exposed to cooler air.
- When injection resumed, the water level rose and covered the junction causing the temperature to rise due to contact with warm water.

TC responses and water injection flowrate

PCV TC locations

* TEPCO: http://www.tepco.co.jp/decommission/information/committee/roadmap_progress/pdf/2020/d200227_10-j.pdf.

COOLANT SUSPENSION TESTS* (1F3)

- TEPCO found that their analytical model best matched the RPV temperature response data when they assumed that only a small fraction of the core debris (i.e., 10%) remained in the RPV.
- This prediction is consistent with the PCV examinations that indicate a very large amount of core debris (up to 3 meters deep) present in the pedestal region.

Images Courtesy of TEPCO

COOLANT SUSPENSION TESTS* (1F2)

- Two coolant suspension tests were conducted on Unit 2. These tests were as follows:
 1. STEP-1 test: Water injection reduced from 3.0 to 1.5 m³/h
 - Result: Maximum temperature increase at the RPV rose ~5 C from ~ 20 C at start of the test
 2. STEP-2 test: Water injection reduced from 3.0 m³/h to 0
 - Result: Peak temperatures on the RPV lower head rose at a rate of ~5 C/h.
- TEPCO concluded that the temperature responses were in general agreement with their model predictions.

Changes in RPV lower
surface temperature during
STEP-2 test

*Images Courtesy of TEPCO; see <https://www.iaea.org/sites/default/files/20/09/events-and-highlights-august-2020.pdf>

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COOLANT SUSPENSION TESTS (1F2)

- What else can these tests tell us? A couple points that may be worth considering are as follows:
 1. Energy conservation, and implications for parasitic heat loss from the PCV.
 2. Additional estimates of core debris mass remaining in RPV lower head based on thermal response.
 3. What if analysis: should termination of water injection be considered?

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COOLANT SUSPENSION TESTS (1F2)

1. Energy conservation considerations

- First step is to estimate decay heat load in 1F2 at current time. Decay heat level is estimated to be at ~22 kW at current time (e.g., ~ 10 years after the accident).
- Water is injected at 3 m³/h, and the temperature change of coolant is in the range of 5-6 C.
- From $\dot{Q} = \dot{m}c_p\Delta T$, the heat rejection by the coolant at the given flowrate and observed temperature rise is between 17 to 21 kW.
- Given uncertainties in the decay heat estimate, the heat rejection and decay heat estimates are deemed to be in reasonable agreement.
 - Their closeness also indicates that parasitic heat losses are minimal.
- One consideration: To reduce on-site contaminated water production, consider reducing the injection rate and let the water temperature rise a bit?
 - e.g., if flowrate is cut in half, the ΔT would still only be ~10-12 C.

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COOLANT SUSPENSION TESTS (1F2)

2. Additional estimates of core debris mass in lower head (1/2)

- Previously we had estimated* that ~ 1/3 to 1/2 of core mass remained in lower head for 1F2 based on muon tomography measurements augmented by a simple geometry argument.
- These tests provide the opportunity to estimate the mass in the lower head from a different angle.
- The TCs on the lower head provide data on the rate of temperature rise after injection is stopped, and that is related to the thermal loading as $\dot{Q} = mc_p\dot{T}$
- Assuming the (hemispherical) LH has an inner radius of 2.5 m, wall thickness of 16 cm, and a steel density of 8000 kg/m³, the lower head would weigh ~ 54 MT, and steel has a specific heat of ~490 J/kg-K.

Images
Courtesy
of TEPCO

*M. T. Farmer, 'Topic 3-Core Debris Evaluations,' Fukushima Forensics Meeting, November 29-30, 2018, Nuclear Energy Institute, 1201 F St., NW, Suite 1100, Washington, DC.

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COOLANT SUSPENSION TESTS (1F2)

2. Additional estimates of core debris mass in lower head (2/2)

- Assuming:
 1. half the core inventory is in the vessel lower head,
 2. all decay heat in the debris is transferred to the lower head,
 3. the steel specific heat capacity is $\sim 490 \text{ J/kg-C}$,
 4. the lower head can be treated adiabatically, and
 5. a lumped capacity heat transfer approach can be used to estimate the head thermal response.
- Under these assumptions, the rate of temperature rise of the lower head is calculated to be 1.5 C/hr .
 - This is ~ 3 times the actual rate of $\sim 0.5 \text{ C/hour}$.
- Conversely, if we assume that $1/3$ of the core debris is in the lower head, leaving $22/3 \sim 7 \text{ kW}$ of the decay heat therein, then the predicted rate of temperature rise would be $\sim 1.0 \text{ C/hour}$, which matches the measured value.
- This simple analysis supports the previous muon-tomography-based inference that $\sim 1/3$ to $1/2$ of the core debris remains in the lower head.

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COOLANT SUSPENSION TESTS (GENERAL)

3. What if coolant injection was terminated all together?

- After 10 years, the decay heat level in the debris has fallen to a relatively low value.
- Meanwhile, the PCV and reactor building itself offer large surface areas for heat transfer via natural convection and conduction to ambient.
- During the accident, the question arose as to how long it would take before the decay heat in the affected units would fall to the point where the heat could be dissipated passively to the environment.
- A scoping analysis* was carried out soon after the accident to provide answers to this question.

*M. T. Farmer, "Assessment of Long Term Passive Cooling Viability for Fukushima Units 1-3," M. T. Farmer, March 31, 2011.

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COOLANT SUSPENSION TESTS (GENERAL)

3. What if coolant injection was terminated all together? (Contd.)

Summary of assumptions on heat transfer resistance and boundary conditions

Res.	Description	Heat Transfer Assumptions	Heat Transfer Conductance, Watts/°C	Potential Heat Removal Rate Across Boundary at 100 °C
1	Convection and/or Conduction from In-Vessel Core Debris to RPV wall	Based on extensive in-vessel retention R&D, assume that decay can be passively dissipated through nucleate boiling on the exterior of the lower head and so heat transfer resistance is negligible.	Infinite	Infinite
2	Conduction Across RPV Wall		Infinite	Infinite
3	<i>Convection from RPV Outer Wall to Drywell Liner</i>	<i>Drywell liner surface area ~1500 m²; free convection heat transfer coefficients 5-25 W/m²-°C</i>	<i>7,500 – 37,500</i>	<i>0.75 – 3.75 MW</i>
4	Conduction Across Drywell Liner	Thickness 3.18 cm, thermal conductivity ~ 40 W/m-°C, surface area ~1500 m ²	1.9-10 ⁶	190 MW
5	Convection Across Drywell Liner Gap	Will be small if gap is plugged and can be significant if gap is filled with condensing steam or water; resistance neglected in this analysis	Infinite	Infinite
6	<i>Conduction Across Concrete Shield Wall</i>	<i>Thickness ~ 200 cm, thermal conductivity ~ 2.2 W/m-°C, surface area ~1500 m²</i>	<i>1,650</i>	<i>0.165 MW</i>
7	Convection through Building	Difficult to assess not knowing building state as well as whether or not ventilation pathways exist. Thus, this is neglected.	Infinite	Infinite
8	Conduction through Reactor Building Exterior Wall	Thickness ~ 100 cm, thermal conductivity ~ 2.2 W/m-°C, surface area ~8500 m ²	17,000	1.7 MW

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COOLANT SUSPENSION TESTS (GENERAL)

3. What if coolant injection was terminated all together? (Contd.)

Results of passive cooling analysis over the first year of the accident

- Conclusion: Not possible over this time interval

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COOLANT SUSPENSION TESTS (GENERAL)

3. What if coolant injection was terminated all together? (Contd.)

- This scoping analysis can roughly be extended to the current time, as follows.
- The surface area of the shielding concrete/drywell surface is $\sim 1500 \text{ m}^2$.
- Assuming a (pessimistically low) NC heat transfer coefficient of $5 \text{ W/m}^2\text{-K}$, then the outer surface of the shielding concrete would need to rise on average 3 C above ambient to dissipate the 22 kW of decay heat.
- Looking at conduction heat transfer from the exterior to the interior of the shielding concrete (2 m thick) would require a 13 C temperature rise.
- Again using a low NC heat transfer coefficient of $5 \text{ W/m}^2\text{-K}$ on the interior of the drywell surface (taken as 1500 m^2), an additional 3 C temperature rise would occur.
- Thus, assuming an ambient temperature of 20 C in the reactor building, the resultant temperature in the PCV to dissipate decay heat would be 36 C given the various temperature rises that are calculated above.

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COOLANT SUSPENSION TESTS (GENERAL)

3. What if coolant injection was terminated all together? (Contd.)

- Although approximate in nature, this analysis indicates that terminating water injection would not lead to excessive temperatures inside the PCV, at least for unit 2.
- If this is possible, it would likely reduce demands for on-site treatment and storage of contaminated water.
- When the time comes to enter the PCVs to start D&D, injection could always be restarted to scrub aerosols.
 - However, this would be much more effective if the water is placed where the debris is being disturbed for removal, as opposed to letting it drain through the bottom of the RPV.
- If this would warrant additional consideration, it would be useful to run TEPCO's containment heat-up models under this scenario to see what the PCV temperature might rise to.
- Also, MELCOR/MAAP/SAMPSON have detailed containment models that mock up the various heat transfer resistances cited herein; it may be worth exercising these models to explore this possibility.

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EVALUATIONS OF CS-137 LEVELS TO INFER DEBRIS LOCATION (1/3)

Source: Nuclear Regulatory Authority:
(<https://www.nrs.go.jp/data/000315708.pdf>)

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EVALUATIONS OF CS-137 LEVELS TO INFER DEBRIS LOCATION (2/3)

- Debris endstate locations inferred from measurements and calculational assumptions outlined in <https://www.nsr.go.jp/data/000315708.pdf>
 - Used measurements and images obtained from TEPCO supplemented with JNRA investigations
 - Generally, assumed Cs-137 uniformly distributed between plugs but considered damage observed in images (1F3 and 1F1)

Source: Nuclear Regulatory Authority:
(<https://www.nrs.go.jp/data/000315708.pdf>)

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EVALUATIONS OF CS-137 LEVELS TO INFER DEBRIS LOCATION (3/3)

Source: Nuclear Regulatory Authority:
(<https://www.nrs.go.jp/data/000315708.pdf>)

- Calculations consider release to the environment and reactor building water.
- Preliminary results indicate significantly higher contamination levels remain in 1F2 with smallest amount remaining in 1F1 (differences attributed to accident progression)
- Results consistent with evaluations from other sources and emphasize the importance of forensics investigations and systems analysis tools to assist D&D

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THOUGHTS ABOUT CAPABILITIES OF LABORATORY -2 (1/2)

- On-site capabilities of Laboratory 2 for analyzing fuel debris important
 - significantly simplifies needs for transportation
 - provides timely results to on-going D&D activities
- Prioritization of capabilities based on D&D needs important
 - TMI-2 and ChNPP4 experience and preliminary 1F debris examination results important for identifying required capabilities

Image Courtesy of JAEA

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THOUGHTS ABOUT LABORATORY -2 HOT CELL CAPABILITIES (2/2)

	分析ニーズ (分析データの反映度)	分析・試験項目
初期(燃料デブリの取出開始から10年程度)	燃料デブリを構成する物質の由来等の推定への反映等	組成-U/Pu含有率 組成-Zr/Fe/Cr含有率 (SUS等由来)
	取出等の技術開発に当り想定した条件の妥当性の確認 -インベントリ管理への反映等	組成-U/Pu含有率 組成-U/Pu含有率
	取出等の技術開発に当り想定した条件の妥当性の確認等	形状 (形状/塊) 、化学形態、表面状態 硬さ、じん性
	燃料デブリの取出し工程への反映等	寸法 (粒径)
	粒子状の燃料デブリの取出し工程への反映等	組成-B含有率 (B/C等由来)
	燃料デブリの取出し工程への反映等	組成-Gd含有率
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
中期(燃料デブリの取出開始後5～20年程度、当初計画範囲)	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
後期(燃料デブリの取出開始後20年～)	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)
	燃料デブリの取出し工程への反映等	組成-Si含有率 (Si/C等由来)

U.S. Comments:

- Prioritization of capabilities based on D&D important
- Investigations to understand severe accident progression and D&D needs closely related (e.g., RN distributions provide insights about structural integrity)
- Round robin evaluation of representative sample might increase international community confidence in results

- Current prioritization for debris exams (see <https://www.nsr.go.jp/data/000319207.pdf>) is:
 - Red (highest priority): Composition, shape, surface state, hardness, debris removal properties)
 - Blue: Density, porosity, heat transport properties
 - Green (lowest priority): fission product emission during heating
- May wish to consider detailed examination input in Appendix B.2 of FY20 report as Laboratory 2 priorities are finalized. Detailed requests regarding evaluations to detect MCCl, debris oxidation state, RN distribution on pedestal wall and floor, etc

Image Courtesy of JAEA;

Source: Nuclear Regulatory Authority (<https://www.nsr.go.jp/data/000319207.pdf>)

REVISIONS TO INFORMATION REQUESTS

- Effectively, none.
- However, it is emphasized to 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 would not have been expected based on the current simplistic vision of water ingress 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.

Images Courtesy of TEPCO

ACKNOWLEDGEMENTS

- Thanks to Japan for continuing to interact on this project, findings at Daiichi have provided many insights into accident management and considerably reduced knowledge gaps in this area.
- Thanks to DOE for their continued support, laboratory involvement would not be possible without it.
- Personal thanks to Dr. Rempe for leading these efforts and providing the information in a format that we can access, analyze, and assess.

C.3.4. Topic Area 4 - Combustible Gas Effects

Fukushima Forensics: Combustible Gas Effects Insights/Comments on 1F3 Hydrogen Combustion based on JNRA Evaluations

**Wison Luangdilok
H2Technology LLC
FAI**

**DOE Reactor Safety Technology Expert Panel Forensics Meeting
Virtual Meeting
November 10-12, 2020**

H2TECHNOLOGY LLC

Outline

- Was the 1F3 Explosion a Detonation?
 - Examine the Wang et al.-2017 Large Scale Testing of Fireball Explosion over a Leaked LNG Pipeline.
- What is the Reinforced Concrete Behavior in response to a Detonation Blast?
 - Examine test results from a series of Foglar et al (2013-2017) publications
 - Examine the damage of 1F3 reactor building concrete structures
- What is the Combustion Pressure in the Fireball Explosion?
- Why did the Condensation Shock appear during the 1F1 Explosion, but not during the 1F3 Explosion?
 - Characteristics of Shock Wave Propagation from Mid-Air Detonation
 - Characteristics of Shock Wave Propagation from Near-Surface Detonation

REVIEW OF RECENT DEVELOPMENT

H2TECHNOLOGY LLC

Fukushima Daiichi Accidents and Hydrogen Explosions

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Recent Analysis of H2 Generation required for 1F3 Explosion

Updated from Luangdilok, *Nuclear Eng. and Design* 362 (2020) 110536 2020

Source of H2	Amount of H2	Method or assumptions
H2 equivalent burned at 1F3	1540 ± 250 kg	Estimated from the fireball size (Luangdilok, 2020)
H2 in 1F3 RB (X3) +PCV	~1448 kg (75% H2 limit) 114~343 kg (PCV)	Assumed high-end 74% H2 concentrations in RB 5F/4F at 323K/310K, 10~30% H2 in DW at 600K/425K, 17% free volume in WW
H2 in 1F4 RB (X4)	186~206 kg	Assumed near low-end concentrations in RB: 9~10% H2 in RB 5F/4F at 300K.
H2 vented through the 1F3/1F4 common stack (X5)	$= (65/35) \times \text{H2 leaked to 1F4}$ $= 345 \sim 383 \text{ kg}$	35% of vent flow from 1F3 leaked to 1F4
Estimate of total generation of H2 equivalent by time of 1F3 explosion (~5% is CO)	2100 ~ 2400 kg	Total sum of X3+X4+X5 + H2 (in 1F3 PCV)
Theoretical amount ~2945 kg (100% oxidation)		

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Potential Sources of H2 Generation in the 1F3 Accident

Core Component	Potential source (kg)	Potential H2 generation (100% oxidation) (kg)	Potential H2 generation (100% oxidation) %
Zr in fuel cladding	29000	1272	43%
Zr in channel box	18000	789	27%
Fe in control blade	12800	641	22%
B4C in control blade	960	243	8%
Total (kg)	60760	2945	100%

Estimate of total H2 generation by time of 1F3 explosion

2100 ~ 2400 kg

Updated from Luangdilok, *Nuclear Eng. & Design* 362 (June 2020) 110536

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The Estimate of H₂ Generation is large, but not too Large.

- The BASF phase 2 study shows an encouraging result (Lind et al., NURETH-18, 2019).
- One of the analyses by VTT using MELCOR comes very close to this estimate (2420 kg MELCOR vs. 2100~2400 kg estimate).
- Experimental works at JAEA (Kurata et al. 2018) and KIT (Steinbrück, 2014) suggests that the significance of oxidation of B₄C containing melt such as B₄C-SS-Zry melt resulting from eutectic interaction of B₄C and stainless steel (SS) in the control blade and zircaloy (Zry) in the channel box.

Kurata et al. JNM 500 (2018) 119-140

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Can the 1F3 Fireball burn in a Detonation Mode?

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

Wind speed : 4-7 m/s

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

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Analysis of Combustion Pressure required for Elastic Buckling of an LNG Pipeline due to Radial Compression

Elastic buckling pressure of a uniform cylinder	Timoshenko 1961 Theory of elastic stability	$P_c = \frac{Eh^3}{4(1-\nu^2)R^3}$		
P_c	Pressure required for Elastic buckling (Pa)	6.90E+06	1 ksi = 6.895 MPa	Yield strength of X80 steel, MPa
Steel	Elasticity modulus E	2.00E+11	σ_y	Yield strength of X80 steel, ksi
Steel	Poisson ratio ν	0.3	$P_y = \sigma_y h / R$	Pressure that cause plastic yielding ksi
Pipe thickness	h (m)	0.0356		Pressure that causes plastic yielding MPa
Pipe radius	R (m)	0.711		
	Hoop stress (Pa) at buckling pressure	1.38E+08	Less than	
	Hoop stress in ksi	20.0		

Buckling pressure $P_c = 6.9$ MPa < Plastic yielding pressure $P_y = 31.3$ MPa

Buckling failure (radial deformation in a cosine-shaped buckling mode) occurs first.

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Near-Ground Detonation above the Pipeline ?

Large-Scale Testing of an Explosion from a leaked LNG Pipeline
(Wang et al., J of Loss Prev. in the Proc. Industries, 46 (2017)13-22)

X-80 steel pipeline 1.5 m underground
Pipe diameter = 1.422 m
Pipe length = 430 m

Only a detonation can generate a pressure of at least 6.9 MPa.

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Similarity between the Large-Scale LNG Pipeline Testing and the Unit-3 Explosion

Wind speed : 4-7 m/s

Temperature > 2200°C

~400 m

276,000 kg of CH4 burned

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

260-300 m

5-20 s: Stable and fully grown fireball

20-30 s: Fireball weakened and burnout

30-45 s: Height decreased and completely burned out.

1F3 Explosion

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

ANL/LWRS-19/08

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What is the Reinforced Concrete Behavior in response to a Detonation Blast?

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1F3 Damage Conditions after debris removal

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The Explosion Damage was typical of a detonation blast

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

Edited contents from: Nuclear Regulatory Authority Slide 14 of Yasui presentation, Sept 2, 2020 (<https://www.nsr.go.jp/data/000325406.pdf>)

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Testing of Reinforced Concrete subjected to a Close-Range 25kg-TNT Detonation

Constructed from Foglar et al (2013 & 2015)

Specimen	Concrete Grade*	Additional synthetic fiber	Compressive strength	Concrete tensile strength	Reinforcement steel** tensile strength	Reinforcement steel** yield strength	Fiber volume
1	C30/37	No	60.1 39 MPa	2.9 MPa	600 MPa	560 MPa	0.0
2	C30/37	Yes					0.50%
3	C55/67	No	80 55 MPa	4.07 MPa	600 MPa	560 MPa	0.0
4	C55/67	Yes					0.50%
9	C30/37	Yes	48 MPa	2.9 MPa	600 MPa	560 MPa	4.5%

Foglar et al., Construction and Building Materials, 94 (2015)536-546

Foglar & Kovar, Int. J. Impact Engineering, 59 (2013)18-29

Copyrighted images removed

TNT Mechanism (reduced)
 $C_7H_5N_3O_6 \rightarrow 1.5N_2 + 2.5H_2O + 3.5CO + 3.5C$
 $C + \frac{1}{2}O_2 \rightarrow CO$
 $CO + \frac{1}{2}O_2 \rightarrow CO_2$

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Reinforced Concrete Behavior in response to a Close-Range Detonation: Specimen No. 1

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Large concrete puncture ~ 1m opening

Spallation is worse on the bottom side.

Longitudinal deflection ~5%

Foglar & Kovar, Int. J. Impact Engineering, 59 (2013)18-29

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Reinforced Concrete Behavior in response to a Close-Range Detonation: **Specimen No. 3** (increased COMPRESSIVE strength)

Foglar & Kovar, Int. J. Impact Engineering, 59 (2013)18-29

Longitudinal deflection <5%

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Reinforced Concrete Behavior in response to a Close-Range Detonation: **Specimen No. 9** (reduced COMPRESSIVE strength)

Foglar et al., Construction and Building Materials, 94 (2015)536-546

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Blast Overpressure generated by Denotation: Measurement

Foglar & Kovar, Int. J. Impact Engineering, 59 (2013)18-29

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Blast Overpressure Propagation : Blast No. 3

Copyrighted images removed

Blast wave over pressure

$$\Delta p_+ = \frac{93}{Z} + \frac{383}{Z^2} + \frac{1,275}{Z^3} \quad (\text{kPa})$$

$$Z = \frac{R}{\sqrt[3]{W}}$$

$R \leftarrow$ distance from the blast center (m)
 $W \leftarrow$ weight of TNT (kg)

Foglar & Kovar, Int. J. Impact Engineering, 59 (2013)18-29

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Near-Field Blast Overpressure : an Estimate

- Blast Overpressure Correlations by Henrich (1979) Dynamics of Explosions and its Use

$$\Delta P = \frac{619.38}{Z} - \frac{32.62}{Z^2} + \frac{213.24}{Z^3} \quad \text{for } 0.3 < Z < 1$$

$$\Delta P = \frac{66.2}{Z} + \frac{405}{Z^2} + \frac{328.8}{Z^3} \quad \text{for } 1 \leq Z \leq 10$$

- Far-field Blast Overpressure Correlations used by Foglar & Kovar (2013)

$$\Delta P = \frac{93.2}{Z} + \frac{383}{Z^2} + \frac{1275}{Z^3}$$

- Far-field Blast Overpressure Correlations by Mill (2001)

$$\Delta P = \frac{108}{Z} - \frac{114}{Z^2} + \frac{1772}{Z^3}$$

- Far-field Blast Overpressure Correlations by Sadovskii (1952)

$$\Delta P = \frac{76}{Z} + \frac{255}{Z^2} + \frac{650}{Z^3}$$

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Near-Field Blast Overpressure : an Estimate

- Blast Overpressure Correlations by Henrich (1979) Dynamics of Explosions and its Use

$$\Delta P = \frac{619.38}{Z} - \frac{32.62}{Z^2} + \frac{213.24}{Z^3} \quad \text{for } 0.3 < Z < 1$$

A distance from the TNT charge center to the nearest side edge of the concrete beam.

R (m)	2	0.93
W (kg TNT)	25	25
Z	0.69	0.32
Overpressure (effective)	1.5 MPa (0.17 MPa)	8.25 MPa (0.89 MPa)

The level of reinforced concrete damages can only be caused by detonation.

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Detonation Wave Pressure at a Distance 2 m away in the Test

Radial Distance R from detonation center to the impact point (m)	2.00
Overpressure P in radial direction at distance R from detonation center(MPa)	1.50
Horizontal distance of impact point from the detonation center(m)	1.92
Angle of the line connecting the impact point to the detonation center Θ (degree)	15.97
Θ in radian	0.11
$\sin(\Theta)$	0.11
Detonation wave pressure against concrete surface (in vertical direction) $P \cdot \sin(\Theta)$ (MPa)	0.17
Detonation wave pressure parallel to concrete surface (in horizontal direction) $P \cdot \cos(\Theta)$ (MPa)	1.49

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Detonation Pressure Loads on the Concrete Slab during the Tests

Foglar et al., Construction and Building Materials, 94 (2015)536-546

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SIMPLIFIED ANALYSIS OF EXPLOSION PRESSURE REQUIRED to INITIALLY CRACK THE CONCRETE SLAB

Foglar & Kovar, Int. J. Impact Engineering, 59 (2013)18-29

Analysis of Reinforced Concrete Slab used in the Testing

Total area of reinforcement bars
Yield strength of steel bar
Concrete compressive strength
Length of concrete beam
Width of concrete beam
Thickness of concrete beam
Compressive layer depth, calculated.
Rebar position measured from the top of the beam, assumed.
Max. moment capacity
Max. load capacity
Max. pressure Loading
Assumed beam length under detonation pressure (m)

Method 1-Calculate the beam moment capacity for resisting tensile failure of reinforcement bars

	$M = A_s f_y (d - a/2)$
A_s, m^2	22 cm ²
f_y (MPa)	560
f_c (MPa)	60
L (m)	6
b (m)	1.5
h (m)	0.3
$a = A_s f_y / (0.85 f_c b)$ (m)	0.016
d (m)	0.25
M (N·m)	299609.1
F (N)	99869.7
P (Pa)	1.46 bar
m	3

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Reinforced Concrete Behavior in response to a Close-Range TNT Detonation

- Concrete damage occurs in the form of crack, spallation, and puncture.
- The bottom side suffers much more damage than the top blast-facing side.
- Spallation is limited to close distances to the detonation center.
- Near-surface detonation can puncture reinforced concrete slabs
- The puncture size depends on the compressive strength of the concrete, i.e., the smaller the compressive strength the larger the puncture size.

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ANALYSIS OF 1F3 CONCRETE BEAM

Unit 3 - Small Beam Analysis	Flexural Strength of reinforced concrete	Method 1-Calculate moment capacity for resisting tension failure of the beam from reinforcement properties	Method 2 - Use design shear force to obtain the moment capacity for resisting tension failure
Total area of reinforcement bars sq. M.	As. m ²	$M = A_s \cdot f_y \cdot (d - a/2)$ 0.0021	$M = V \cdot L/2$ 0.0021
Yield strength of steel bar	f_y (MPa)	560	
Concrete compressive strength	f_c (MPa)	35	
Length of concrete beam	L (m)	7.6	7.6
Width of concrete beam	b (m)	0.6	0.6
Thickness of concrete beam	h (m)	1.05	1.05
Compressive layer depth, calculated.	$a = A_s \cdot f_y / (0.85 \cdot f_c \cdot b) \text{ (m)}$	0.066	
Depth of rebar location measured from the top of the beam, assumed.	d (m)	0.92	
Strain in the rebar	$e = (d/a \cdot 0.8 - 1) \cdot 0.003$	0.031	
Moment of inertia of beam	$I = b \cdot h^3 / 12 \text{ (m}^4\text{)}$		
Design shear force capacity	V (ton)		32.1
Design moment capacity	M (N·m)	1043181.2	1195404
Design load capacity	W (N)	274521.4	314580.0
Max. pressure Loading	P (Pa)	2.2 bar	2.4 bar

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Reason for Detonation in 1F3 Explosion: Severe punctures of the floor of the 5th floor.

Based on
1. Severe puncture in the testing of RC subjected to close-range detonation,
2. Detonation damage from fireball in the large-scale LNG pipeline testing

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Edited contents from: Nuclear Regulatory Authority Slide 14 of Yasui presentation, Sept 2, 2020 (<https://www.nsr.go.jp/data/000325406.pdf>)

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Severe structural damages on the northwest side of the RB 5th floor (diagonally opposite to the southeast side where the spent fuel pool is)

- Were the damages caused by fireball detonations above the 5th floor?
- If so, why did the fireball detonation occur here rather than somewhere else?
- Or, if the damages were caused by detonations from beneath the 5th floor (i.e., 4th floor), how did it happen?

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Condensation Shock caused by Blast Under-pressure

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Condensation Shock was visible during the 1F1 Explosion

Yanez et al., Int. J. Hydrogen Energy 40 (2015) 8261-8280

- Shock wave speed of 360 m/s was estimated at distance 50~70 m.
- Shock wave overpressure of 28 kPa was estimated.

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Weather Conditions during 1F1 and 1F3 Explosions

Data taken at Onahama, a seaside location 40 miles south of 1F site

Data taken at Ohno monitoring post, an inland location 5 km west of 1F site

○気象庁 過去の気象データ検索における日ごとの値

参考

UNIT 1

小名浜 2011年3月12日 (10分ごとの値)

時分	気圧(hPa)		降水量 (mm)	気温 (°C)	相対湿度 (%)	風向・風速(m/s)				日照 時間 (分)
	現地	海面				平均	風向	最大瞬間	風向	
1530	1021.5	1022.2	x	8.8	59	5.4	南	7.9	南	10

UNIT 3

小名浜 2011年3月14日 (10分ごとの値)

時分	気圧(hPa)		降水量 (mm)	気温 (°C)	相対湿度 (%)	風向・風速(m/s)				日照 時間 (分)
	現地	海面				平均	風向	最大瞬間	風向	
1100	1014.0	1014.6	x	11.7	71	3.9	南	6.4	南	10

Temperature

Relative humidity

	大野 (モニタリングポスト)			温度、湿度	風向・風速
	Unit 1	Unit 3			
3/12	15:30	9.3°C 47.1%	15:30	E	4.2 m/s
	15:40	9.4°C 44.8%	15:40	ESE	3.7 m/s
3/13	Unit 3				
3/14	11:00	18.9°C 14.1%	11:00	W	7.0 m/s
	15:40	18.9°C 14.7%	11:10	WSW	5.9 m/s
3/15	06:10	6.3°C 71.4%	06:10	NNW	2.9 m/s
	06:20	6.3°C 73.1%	06:20	NNW	3.0 m/s

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Yasui, NRA Presentation, Sep. 2, 2020, Slides 36 & 33 <https://www.nsr.go.jp/data/000325406.pdf>

The 1F3 Explosion Mechanism is Different.

- Not a major premixed combustion like in 1F1.
- Most hydrogen (with some CO) was so rich and non-flammable (not enough O₂) at onset. Except for initial burning (probably near or below the floor).
- Fireball entrained significant amount of concrete dust and drew surrounding air to keep the explosion going.
- **It likely involved near-surface detonations.**
- Building structures interfered with blast wave propagation.
- (Hence) No appearance of condensation shock.

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Concluding Remarks: Important Questions to ask

- Recent analysis suggests the possibility that a large amount of unmixed hydrogen gases had been accumulating at extremely rich concentrations in the reactor building including the 5th floor and the 4th floor prior to the explosion in 1F3.
- The 1F3 explosion and its damages were one of many possibilities the combustion of an extremely large amount of hydrogen can lead to.
- The explosion processes could occur differently with different levels of damages to the reactor building if the location and timing of ignition, which we don't know, are different.
- Why were severe damages tilted toward the NW side of the RB away from the spent fuel pool area?
- Would there be certain explosion scenarios that lead to severe damages of the SW side where the spent fuel pool is located?

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C.3.5. Topic Area 5 - Operations and Maintenance

C.3.5.1. Terry™ Turbopump Testing 2020 Update

Terry Turbopump Testing 2020 Update

Randy Bunt (Southern Nuclear)
Consortium Chairman
October 2020

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Presentation Content

- Overview
- Milestones Update
- Active Milestone

2

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.

Image courtesy of SNL; originally published in SAND2017-10773

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Terry Turbine Testing Overview (Cont)

- The goal of the International Consortium is to provide long-term oversight of the Terry Turbine Expanded Operating Band Project (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.
- Milestones 1-7, Plan development, basic science, individual component testing, small scale testing, low pressure testing, self regulation simulation, closeout
- 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, USDOE, 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.

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Milestones – Update

- Milestone 1 – Complete
- Milestone 2 – Complete
- Milestone 3 – Complete
- Milestone 4 – Complete, reports pending
- Milestone 5 – Stopped
- Milestone 6 – Stopped (Being pursued as limited scope in 5/6 Hybrid)
- Milestone 5/6 Hybrid – 40-50% Complete, testing and reports pending
- Milestone 7 – Project Closeout

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Active Milestone: Brief Description

Milestone 5/6 Hybrid:

- Due to reduced funding across the Consortium entities (beginning in the Fall of 2019), there was a reduction in scope and deliverables, which eliminated the full-scale testing planned in Milestone 5 and 6 (MS-5/6). Because of this change in scope, the following gaps remain: full-scale steam test data, full-scale duration test with steam, self-regulation full-scale, and impact of steam quality.
- Milestone 5/6 Hybrid is to obtain confidence in scaling factor (MS5) and self-regulation (MS6). This will be accomplished through testing at Texas A&M University (TAMU) during a 3-4-month period utilizing previously designed testing stations and utility experience.
- Data has been collected from the utilities in which GS Terry turbines were used under a variety of inlet steam pressures during injections for greater than four hours and compared to steam testing of the ZS-1.
- EPRI Performed data analysis and provided to the Team. The data has good correlation when an appropriate correction factor is applied. There is a specific area when additional data will enhance the confidence of the correction factor.
- Schedule:
 - Scaling factor confidence tests complete – 4th Qtr 2020
 - Self-regulating air tests complete – 1st Qtr 2021

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Assessment of Plant data to TAMU Data

- Good correlation of test data and plant data when using a correction factor based on nozzle differences is applied to the TAMU test data.
- ZS-1 steam-water/air-water testing is desired to provide additional data correlation with plant data for the green cloud area.

Questions and Comments

Reference Material

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Basic Science Modeling

Unexpected RCIC Behavior in Fukushima Daiichi Unit 2

- Before Fukushima accidents, it was generally considered in SBO analyses that loss of the DC power would result in flooding the steam line and turbine, where it is assumed that turbine would then be disabled.
- This behavior, however, was not observed in the accidents, where the Unit 2 RCIC functioned for nearly three days, much longer than the assumed 4-6 hours battery life.

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Basic Science Modeling

Two-Phase RCIC System Test Model

- The goal is to create a Fukushima-accidents-like scenario, with a reasonable self-regulating behavior appearing in the simulation.
- Periodic two-phase condition given at the turbine inlet.
- The turbine inlet pressure is 7.5 MPa and the turbine outlet pressure is 193 kPa.
- The simulation was run for two more cycles of dry-low quality steam two phase flow flood stages.

Terry turbine RCIC system test model

Courtesy of INL

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System Modeling

Courtesy of IAE

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System Modeling

Physical understanding1: Work by flashing

No. 8

- In the case of 1F2, reactor water level would reach the main steam line and subcooled water would have flowed into RCIC turbine.
- Under this condition, work by flashing would be generated in a nozzle due to pressure decrease in turbine system.
- A turbine system using hot water is already put to practical use, so it is highly possible RCIC has continuously operated based on the mechanism.

Courtesy of IAE

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System Modeling

SAMPSON modeling

No. 13

Courtesy of IAE

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Oil Temperature Testing

Conclusion

- Oil Facility
 - 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.

Oil Test Results – 1st Test

- Oil temperature held at 325°F (163°C) for 72hr
 - 2250mL of Mobil DTE 732
 - Noticeable color change
 - Thermal expansion of oil

Oil Test Results – 2nd Test

- Oil temperature held at 250°F (121°C) for 72hr
 - 2250mL of Mobil DTE 732
 - Oil at temp for full 72 hrs
 - Viscosity measured online
 - Color change less noticeable than in 1st test

Courtesy of TAMU

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Nozzle Free Jet Testing

Single-Phase Normal Impinging Jet H = 10 mm

Jet pressure 26 psig Impinging on a surface Terry Turbine nozzle

Updated Experimental Facility

Process flow diagram of the T122 experimental facility single phase free jet tests.

Single-Phase Normal Impinging Jet H = 10 mm

Jet pressure 26 psig Mean velocity vector fields, velocity magnitude contour

A 400-mil throat ID Terry nozzle and elbow connector. The outlet is a rounded square of 9/16 inch.

Courtesy of TAMU

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Turbine Testing

Operating Conditions

Operating Conditions for current tests:
Suction Pressure psia: 30, 50, 70, 80, 90, 100, 110
Speed: 400-3600 rpm
Air Fraction %: 100, 90, 80, 70, 60, 50, 40, 30, 25, 20

Two Phase Flow- 70 PSIA Inlet Pressure

□ Terry turbine performance is consistently degraded due to increased water fraction in the two-phase mixture

Conclusions

- The performance of ZS-1 terry turbine under single-phase (100% air) and two-phase (air and water) flow mixtures is being investigated.
- Single-phase flow data revealed deviation in the performance for 30 psia inlet pressure compared to that expected from affinity laws. Improved performance with higher suction pressures is possibly indicative of a changed expansion process across the inlet nozzle.
- Systematic degradation in the turbine performance was observed as a function of two-phase flow. The best efficiency point is moving towards the right to a higher turbine speed. Overall efficiency is improved with an increase in suction pressure, indicating reduced momentum losses associated with two-phase flow interaction.

Schematic of Flow Loop for Z1 Turbine Test

Open loop multiphase flow facility

- Liquid: 0-100 gpm
- Compressed air supply
 - Unit 1: 2800 CFM @ 120 psi
 - Unit 2: 950 CFM @ 350 psi
- Electro-Pneumatic Valves to Control Water and Air Flow Rate
- Heater installed to Preheat the Air

Courtesy of TAMU

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Turbine Testing

Experimental Setup GS-2

Flow Visualization

- Mid-low air mass fraction (30%)
- Annular-Churn transition flow regime
- Water mostly suspended in air
- Increasing pressure...

Flow Visualization

- 70 psia
- A) Thin annular layer along pipe walls
- B) Water flowing along walls and in the air stream
- C) Water suspended in air stream
- D) More water flowing suspended in air stream

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

Two-Phase Performance

Courtesy of TAMU

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Turbine Testing

ZS-1 Turbopump Steam - Water Tests

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- **Coupled with Dyno:**
 - Turbine steady-state response
 - Torque vs. speed for given steam pressure, quality, and flow
 - Turbine step response
 - Time constant for step change in flow conditions
- **Coupled with RCIC Pump analog**
 - Turbopump steady-state response
 - Pump output vs. backpressure for given steam pressure, quality, and flow
 - Turbopump step response
 - Time constant for step change in flow conditions
- Provide an experimental basis for improved CFD and system level modeling efforts.

Courtesy of TAMU

C.3.5.2. Terry™ Turbopump Testing 2020 Update (Backup Material)

Milestones 3, 4 and 5/6 Hybrid Testing Updates

TEXAS A&M★ENGINEERING

Karen Vierow Kirkland
Texas A&M University, College Station, TX USA

Stakeholders Meeting of the RISA Pathway for the LWRS Program
Virtual Meeting, October 13, 2020

Presentation Overview

TEXAS A&M★ENGINEERING

- Overview of Testing at Texas A&M
 - Milestone 3
 - Milestone 4
 - Milestones 5/6 Hybrid
- Conclusions

Overview of Testing at Texas A&M

TEXAS A&M
ENGINEERING

- Milestones 3 and 4
 - Testing status is compared against the latest version of the test plan.
 - Terry Turbopump Expanded Operating Band Full-Scale Component and Basic Science Detailed Test Plan – Final, SAND2017-XXXX, Dec. 2016.
- For each test series, the following information is provided:
 - Objectives
 - Testing Status
 - Information Dissemination
 - Photos of facilities and/or figures showing the type of results produced
- Milestones 5 and 6 Hybrid
 - Progress to date is summarized.
 - Tasks and objectives were defined in spring 2020 discussions.

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Milestone 3 Free Jet Testing

TEXAS A&M
ENGINEERING

- Free Jet Testing
 - Objective
 - Develop a body of knowledge regarding the realistic outcomes of two-phase flow past a GS-series Terry turbine nozzle.
 - Testing status
 - The tests were conducted in Dr. Yassin Hassan's Thermal-Hydraulic Research Laboratory (Nuclear Engineering)
 - **Testing has been completed.**
 - Information Dissemination
 - Results documented in the Year 3 deliverable to the IAE.
 - Raw data transferred to the IAE in January 2020.

Milestone 3 Free Jet Testing

TEXAS A&M
ENGINEERING

CAD design of TTIJ experimental facility

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Milestone 3 Free Jet Testing

TEXAS A&M
ENGINEERING

- Experimental results were obtained from the velocity measurements of impinging jets at the distances of
 - 10 mm
 - 20 mm
 - 30 mm
- PIV techniques were employed.

Results of jets impinging on a flat plate at a distance of 10 mm

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Milestone 3 GS-series Governor & Trip/Throttle Valve Testing

TEXAS A&M
ENGINEERING

- GS-series Governor & Trip/Throttle Valve Testing
 - Objective
 - Develop a body of knowledge regarding the realistic outcomes of GS-series Terry turbine governor valve and trip/throttle valve behavior under low steam pressure operations.
 - Testing status
 - Tests were conducted in Dr. Abhay Patil's Turbomachinery Laboratory (Mechanical Engineering) for air and air-water tests.
 - **Testing has been completed.**

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Milestone 3 GS-series Governor & Trip/Throttle Valve Testing

TEXAS A&M
ENGINEERING

- GS-series Governor & Trip/Throttle Valve Testing
 - Information Dissemination
 - Within the Advisory Council
 - Results documented in the Year 3 deliverable to the IAE.
 - Raw data transferred to the IAE in January 2020.
 - Public documents
 - Vandervort, J., Two-phase Operational Performance of a Terry GS-2 Steam Turbine Using Air & Water As Working Fluids, MS Thesis, TAMU, Dept. of Mechanical Engineering, May 2020.
 - Vandervort, J., G. Lukasik, B. Ayyildiz, M. Solom, A. Delgado, K. Vierow Kirkland, A. Patil, "Performance Evaluation of a Terry GS-2 Steam Impulse Turbine with Air-Water Mixtures", submitted for journal publication in Sept. 2020.

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Milestone 3 GS-series Governor & Trip/Throttle Valve Testing

TEXAS A&M
ENGINEERING

- Established database for single-phase air and two-phase air-water tests.
- Found that turbine follows established affinity laws with correction for two-phase flow.

Crystal River-3 Terry Turbine (GS-2)
used for Milestone 3 testing

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Milestone 3 GS-series Governor & Trip/Throttle Valve Testing

TEXAS A&M
ENGINEERING

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Milestone 3 Lube Oil Testing

TEXAS A&M
ENGINEERING

- Lube Oil Testing
 - Objective
 - Develop a body of knowledge regarding the realistic performance of GS-series Terry turbopump lubrication oil behavior under BDBE conditions.
 - Testing status
 - Tests were conducted in Dr. Karen Kirkland's Laboratory for Nuclear Heat Transfer Systems (Nuclear Engineering).
 - Test matrix was reduced from the test plan.
 - Outside of the scope of the test plan, chemical analysis of the oil was performed by Dr. Hisham A. Nasr-El-Din, Professor & John Holt Edgar Endowed Chair, TAMU Petroleum Engineering
 - Testing has been completed.

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Milestone 3 Lube Oil Testing

TEXAS A&M
ENGINEERING

- Lube Oil Testing Test Plan Matrix
 - Reduced matrix was agreed upon by the Advisory Council in April 2019.
 - Kirkland's lab degraded oil for 72 hours at:
 - 250F, no water
 - 200F, 10% water by volume
 - Dr. Nasr-El-Din's laboratory degraded oil for 72 hours at:
 - 77F, 160, 180, 250, 310, and 390 F, all without water

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Milestone 3 Lube Oil Testing

TEXAS A&M
ENGINEERING

- Lube Oil Testing
 - Information Dissemination
 - Within the Advisory Council
 - Results documented in the Year 3 deliverable to the IAE.
 - Raw data produced by Dr. Kirkland's lab transferred to the IAE in January 2020.
 - Reports of Petroleum Engineering analysis, upon request.
 - Public documents
 - D. Peck, Turbine Oil Degradation and Its Effects on Performance of a Terry Turbine in Nuclear Applications, MS Thesis, TAMU, Dept. of Nuclear Engineering, May 2019.
 - Gage Green, Effect of Degraded Oil on Terry Turbine Bearing Performance at High Temperature, MS Thesis, TAMU, Dept. of Nuclear Engineering, May 2020.
 - Intention to prepare a journal submission.

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Milestone 3 ZS-1 Bearing Tests

TEXAS A&M
ENGINEERING

- Bearing Testing (ZS-1 bearings)
 - Objective
 - Develop a body of knowledge regarding the realistic performance of ZS-1 Terry turbopump bearings with lubrication oil behavior under BDBE conditions.
 - Both ZS-1 and GS-2 tests are in the test plan.
 - Testing status
 - ZS-1 tests were conducted in Dr. Karen Kirkland's Laboratory for Nuclear Heat Transfer Systems (Nuclear Engineering).
 - Reduced matrix was agreed upon by the Advisory Council in April 2019.
 - Kirkland's lab ran 6-hour tests at 3,000 rpm and 1,000 rpm at 250F.
 - No alternating speed test.
 - Testing has been completed.

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Milestone 3 ZS-1 Bearing Tests

TEXAS A&M
ENGINEERING

ZS-1 Bearing Test Facility

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Milestone 3 GS-2 Bearing Tests

TEXAS A&M
ENGINEERING

- Bearing Testing (GS-2 bearings)
 - Objective
 - Develop a body of knowledge regarding the realistic performance of GS-2 Terry turbopump bearings with lubrication oil behavior under BDBE conditions.
 - Both ZS-1 and GS-2 tests are in the test plan.
 - Testing status
 - GS-2 tests were conducted in Dr. Karen Kirkland's Laboratory for Nuclear Heat Transfer Systems (Nuclear Engineering).
 - Testing has been completed.

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Milestone 3 GS-2 Bearing Testing

TEXAS A&M
ENGINEERING

GS-2 Bearing
Test Facility

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Milestone 3 ZS-1 and GS-2 Bearing Testing

TEXAS A&M
ENGINEERING

- Bearing Testing
 - Information Dissemination
 - Within the Advisory Council
 - Results for ZS-1 bearings documented in the Year 3 deliverable to the IAE.
 - Raw data for the ZS-1 transferred to the IAE in January 2020.
 - Results for GS-2 bearings presented at EPRI-hosted web meeting in June 2020.
 - Public documents
 - Gage Green, Effect of Degraded Oil on Terry Turbine Bearing Performance at High Temperature, MS Thesis, TAMU, Dept. of Nuclear Engineering, May 2020.
 - Thomas Freyman's MS thesis (expected in Fall 2020)
 - Intention to prepare a journal submission for each turbine's data set.

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Milestone 4 ZS-1 Turbine Testing

TEXAS A&M
ENGINEERING

- ZS-1 Turbo-pump Testing
 - Objective
 - Develop a body of knowledge regarding the realistic outcomes of Terry turbopump performance.
 - Testing Status
 - Tests were conducted in Dr. Abhay Patil's Turbomachinery Laboratory (Mechanical Engineering) for air and air-water tests.
 - Testing has been completed.
 - Tests will be conducted in Dr. Karen Kirkland's Laboratory for Nuclear Heat Transfer Systems (Nuclear Engineering) for steam and steam-water tests.
 - Will remove pump from ZS-1 turbine and replace with Dynamometer for most of the tests.
 - Testing has begun.

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Milestone 3 ZS-1 Turbine Testing

TEXAS A&M
ENGINEERING

Facility Setup
for Air-water
Tests in the
Turbolab.

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Milestone 4 ZS-1 Turbine Testing

TEXAS A&M
ENGINEERING

Dimensionless Analysis- Affinity Laws

Flow coefficient $\Rightarrow \Phi = \frac{Q}{\omega D_s^3}$

Head coefficient $\Rightarrow \frac{n}{(1-n)p_{inlet}} \left[\left(\frac{p_{outlet}}{p_{inlet}} \right)^{\frac{n-1}{n}} - 1 \right] = \frac{\rho D_s^2 \omega^2}{\rho D_s^2 \omega^2}$

Power coefficient $\Rightarrow \Pi = \frac{\omega \Gamma}{\rho \omega^3 D_s^5}$

Efficiency $\Rightarrow \eta = \frac{\omega \Gamma}{\rho g Q H} = \frac{\Pi}{\phi \psi}$

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Milestone 4 ZS-1 Turbine Testing

TEXAS A&M
ENGINEERING

Affinity Laws Modified to Include the Two Phase Flow

* IP 110 psia, AirF 100% ♦ IP 110 psi, AirF 90% ▲ IP 110 psi, AirF 80% • IP 110 psi, AirF 70% ▲ IP 110 psi, AirF 60%
 □ IP 110 psi, AirF 50% ● IP 110 psi, AirF 40% × IP 110 psi, AirF 30% ○ IP 110 psi, AirF 25% + IP 110 psi, AirF 20%
 - IP 110 psi, AirF 15% - IP 110 psi, AirF 10% ♦ IP 110 psi, AirF 5%

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Milestone 4 ZS-1 Turbine Testing

TEXAS A&M
ENGINEERING

- ZS-1 Turbo-pump Testing
 - Information Dissemination (of air and air-water tests)
 - Within the Advisory Council
 - Testing documented in the Year 3 deliverable to the IAE.
 - Raw data transferred to the IAE in spring 2020.
 - Public documents
 - Patil, A., S. Sundar, Y. Wang, M. Solom, K. Vierow Kirkland, G. Morrison, “Characterization of Steam Impulse Turbine for Two-Phase Flow”, *International Journal of Heat and Fluid Flow*, Vol. 79, 2019, paper 108439.
 - A. Patil, Y. Wang, M. Solom, A. Alfandi, S. Sundara, K. Vierow Kirkland, G. Morrison “Two-phase Operation of a Terry Steam Turbine using Air and Water Mixtures as Working Fluids”, *Applied Thermal Engineering*, Vol. 165, paper 114567, 2020.

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Milestones 5/6 Hybrid

TEXAS A&M
ENGINEERING

- Milestone 5
 - Using the dimensionless parameters proposed by Abhay et al. in Milestone 3 testing.
 - Dr. Kirkland’s lab evaluated NHTS Laboratory test parameters that would enable benchmarking of the small-scale steam-water data to full-scale steam-water plant operation.
 - Small-scale steam-water tests with the ZS-1 Terry turbine are being conducted with conditions as close as possible to those in selected full-scale plant operational test points.
 - These tests correspond to R. Bunt’s discussion of gaining confidence in scaling factor.
 - Tests expected to be complete in the calendar year.

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Milestones 5/6 Hybrid

TEXAS A&M
ENGINEERING

- Milestone 6
 - Full-scale air-water tests with the Crystal River GS-2 Terry turbine will be conducted to observe self-regulation behaviors.
 - Dr. Delgado in Texas A&M's Turbomachinery Lab is designing a flow loop for self-regulation testing.
 - Operating conditions and test parameters required to conduct the experiments have been identified.
 - Equipment is currently being procured and the facility is being constructed.
 - Testing expected by first quarter of 2021.

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Conclusions

TEXAS A&M
ENGINEERING

- Milestones 3, 4 and 5/6 hybrid were summarized.
 - For each tests series, the following was provided
 - Objectives
 - Testing Status
 - Information Dissemination
 - Photos of test setups and summary of results.
 - Most test series have been completed.
 - Progress on uncompleted Milestones 5 and 6 hybrid test series was summarized.
 - No major problems
 - Milestones 5/6 hybrid are in progress.
 - All data are expected to be available in the next few months.

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C.3.5.3. Related Topics

U.S. DEPARTMENT OF
ENERGY

Nuclear Energy

Operations and Maintenance: Related Topics

U.S. DEPARTMENT OF
ENERGY

Nuclear Energy

Operations and Maintenance: Related Topics

- **Lessons learned from the Accident and Decontamination and Decommission efforts can be valuable to the Nuclear Industry, both operating and decommissioning units.**
 - The EPA recently compiled similar information on specific topics
- **Worldwide Nuclear Plants have made safety improvements to operate post-Fukushima accidents.**
 - Safety improvements are a requirement for the Japanese Nuclear fleet to restart. Some of these requirements are quite substantial.

OE from D&D Activities

- EPA Report Illustrative Example

3

Lessons Learned

EPA Report Example Illustrative of Desired Area 5 Lessons Learned Report

Graphic courtesy EPA

EPA Report Objectives and Motivation:

- Japan developing and deploying new technologies for surveying, characterization, stabilization, decontamination, and waste minimization
- Report systematically compares new methods deployed in Japan with available US options:
 - Provides findings about technical effectiveness (efficiency, cost, waste generation, schedule, and safety)
 - Identifies areas requiring additional research and testing prior to recommending use in U.S.

Forensics FY20 Report:

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.

This recommendation should be implemented in FY21

4

Technical Effectiveness considers Efficiency, Cost, Waste Generation, Schedule, and Safety

Survey Area	Small < 1 km ²	Local > 1 km ²	Semi-Regional > 100 km ²	Regional >1000 km ²
Option	Micro unmanned aerial vehicle (UAV)	Unmanned helicopter	Unmanned airplane	Manned helicopter
Altitude	< 10 m	~ 50 m	~ 150 m	~300 m
Features	Allows focused surveys, e.g., above urban areas or in forests; under development	Higher resolution mapping available	Allows remote controlled long-time flight (e.g., six hours); under development	Standardized methodology available for efficient regional surveys

Increasing:
cost, altitude, fuel, range, maintenance, pilot qualifications, ground support

Graphics courtesy EPA 5

Systematically Compares Options for Forest Land Decontamination

Decontamination Technology		Removal of Litter and Humus Manually or by Mechanical Digging (Flat Ground)	Removal of Litter and Humus Manually or by Mechanical Digging (Slopes)	Removal of Litter, Humus and Topsoil by Mechanical Digging (Flat Ground)	Trees	
					High Pressure Washing Trunk	Branch Pruning (Lower Trunk)
Distribution of radioactivity in evergreen forest (September 2011) ^A		44 – 84%			Trunks: 1 – 3%	Branches and leaves: 14 – 53%
Percentage dose reduction ^B (at 1 cm)		60 – 80%	60 – 80%	60 – 80%	~ 30%	5 – 30%
Volume of decontamination waste generated (L/m ²)		20-90	20-90	100-200	< 1,000 per tree	270 (non-compacted waste volume)
Secondary contamination		N/A	N/A	N/A	Water infiltration to soil	Foliage from branches to forest floor
Effects on surrounding environments		Possibility of causing erosion on slopes				
Decontamination speed (one person day)		50 m ²	30 m ²	40 m ²	8 trees	40 m ²
Direct implementation cost Japanese Yen (JPY)/m ² for area > 1,000 m ²)		530	760	890	3,390	580
Overall evaluation	Deciduous forests	Highly effective	Highly effective	Effective	Limited effect	not applicable
	Evergreen forests	Highly effective	Highly effective	Effective	Limited effect	Effective

For branch pruning of trees, the dose rate was measured at a height 1 m above the ground; for other techniques, the dose rate was measured at a height of 1 cm from the ground. The techniques are climate- and season-dependent.

A: Approximately half of the radioactivity was found to be contained in the trees, mainly on the branches and leaves. Branch trimming was confined to the lower parts of trees.

B: Percentage dose reductions were calculated using the values measured before and after decontamination. In most cases, branch pruning was carried out simultaneously with forest floor cleanup (e.g., litter removal), therefore dose reduction was a composite of multiple methods and could not be separately estimated.

Graphic courtesy EPA 6

Systematically Compares Options for Agricultural Land Decontamination

Decontamination Technology	Thin Layer Soil Stripping/ Mowing (Manual or Mechanical Hammer Knife)	Mechanical Digger (Stripping Thickness 5 cm)	Application of Solidification Agent and Collection by Mechanical Digger	Reversal Tillage (Tractor and Plowing) (~ 25-50 cm)	Interchanging Topsoil with Subsoil (Mechanical Digger) (~ 45 cm)
Dose rate reduction (at 1 m)	~ 70%	65 - 95%	~ 40-70%	65 - 80%	~ 65%
Volume of removed soil	Actual volume to be removed	Actual volume to be removed and overbreak ^A	Actual volume to be removed and overbreak ^A	-	-
Secondary contamination	-	-	-	-	-
Area decontaminated (m ² /person day)	70	90	50	1100	100
Application conditions	Effective for thin stripping Flat ground only Cannot use when ground is frozen	Must have sufficient load bearing capacity Cannot strip to a depth of less than 5 cm	Must have sufficient load bearing capacity; Cannot use when standing water is present or ground is frozen; 1 week required for solidification	For low level contaminated soil	For low level contaminated soil
Direct implementation cost JPY/m ² (area > 1000 m ²)	690	560	880	33	310
Overall evaluation	Effective	Effective	Moderately effective ^B	Effective	Highly effective

A: The overbreak is the excess soil removed around the area that was decontaminated, due to the precision of machine operation.

B: The time required for both application of the solidification agent and meeting the correct soil conditions reduces the overall evaluation of this technique.

Graphic courtesy EPA

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Systematically Compares Options for Decontaminating Concrete

Decontamination Method	Dust Collection Sander (Concrete)	Ultra-High-Pressure Water Jet (150-240 MPa)	High-Pressure Water Jet (10-50 MPa)	Shot Blasting
Count rate reduction	60 - 80%	~ 80%	20 - 70%	~ 90%
	(Depends on number of applications)	(Depends on pressure)		(Depends on shot density)
Decontamination waste generated (L/m ²)	Concrete debris ~ 1	Concrete debris ~ 3	Sludge 0.02 - 0.04	Concrete debris ~ 3
Secondary contamination	None. Blast material collected via suction.	As almost all wash water is collected, virtually no secondary contamination occurs.		Dust is collected via suction but some fine material may be lost.
Area decontaminated (m ² / person day)	10	80	50	170
Application condition	Inefficient for use on large areas. Surface must be dry.	Not applicable for areas such as corners where access is limited. Can't use for vertical surfaces.	Should be applied carefully to prevent scattering	No corners or narrow sections. Difficult to apply to vertical surfaces. Surface must be dry.
Direct implementation cost JPY/m ² (area > 1,000 m ²)	1,940	1,150	960	480
Overall evaluation	Effective	Effective	Effective	Effective

MPa: megapascals

Graphic courtesy EPA

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Systematically Compares Options for Road Decontamination

Decontamination Method	Water-Jet Vehicle ^A	High-Pressure Water Jet ^A (10–20 MPa)	Ultra-High Pressure Water ^A (240 MPa)	Shot Blasting ^B	Surface Stripping ^C
Secondary contamination	As almost all wash water is collected, virtually no secondary contamination occurs.			Dust is collected via suction but some fine material may be lost.	
Application conditions	Best for smooth surfaces that are not distorted or damaged.	Best for smooth surfaces that are not distorted or damaged. Roadside drain lids can also be washed		Best for smooth surfaces that are not distorted or damaged. Dry surface	
Area decontaminated (m ² per person day)	1,000	50	80	170	145
Direct implementation cost (JPY/m ²) for area > 1,000 m ²	150	960	1,150	480	390
Overall evaluation	Count rate reduction depends on the pressure. Effectiveness largely depends on road surface conditions and depth of Cs penetration.	Effectiveness largely depends on road surface conditions and depth of Cs penetration.	Count rate reduction depends on the pressure. Highly effective but causes damage and therefore should only be used on highly contaminated roads.	Count rate reduction depends on the blasting density. Highly effective but causes damage and therefore should be used only on highly contaminated roads.	Highly effective but causes damage and therefore should be used only on highly contaminated roads.

^AThese vehicles were equipped with water collection devices to minimize secondary contamination.
MPa = megaPascals.

^BThese vehicles had a collection system for blast materials and therefore secondary contamination was minimal. The little secondary contamination that was produced was manually collected after blasting was completed.

^CSecondary contamination was minimal as above.

^DThe uneven condition and/or cracks in roads may reduce decontamination effectiveness.

Graphic courtesy EPA 9

Waste Generation Important in Considering Road Decontamination Options

Figure 2-12 Panel	Technique	Area Decontaminated (m ² , one person day)	Waste Volume Generated (L/m ²)	Waste Type	Collection Type and Rate	Decon Factor; Gamma Dose Rate Reduction	Direct Implementation Cost (Yen/m ² ; \$/ft ²)* for Areas > 1000 m ²
A	Street sweeping	3,500	1 – 1.5	Soil, road dust, vegetation	N/A	1 – 2;	10; 0.01
B	Ride-on sweeping	1,750				0 – 45%	20, 0.01
C	Water-jet vehicle	1,000	30 – 40	Sludge	Vehicle 50 – 70%	1 – 3;	150; 0.11
D	Manual high-pressure water washing	50			Vacuum 100%	0 – 70%	960; 0.71
E	Hydro-blast ultra-high pressure water washing	80	3	Road dust, water	Vacuum absorption 100%	1 – 3;	1,150; 0.85
F	Dry-ice blasting	70	2	Road dust	N/A	0 – 65%	2 – 15;
G	Sand-blasting	5	20	Road dust, sand		40 – 95	1,310; 0.97
H	Medium-scale shot-blasting (iron balls)	170 – 270	3	Concrete, asphalt dust, iron shot		2.5 – 10;	4,190; 3.09
I	Large-scale shot-blasting (iron balls)			170		Road dust, iron balls	60 – 90%
J	Asphalt planing/shaving	150	8 (@5 mm thickness)	Asphalt		3 – 23;	480; 0.35
K	Mechanical digger asphalt removal	26	150			60 – 95%	390; 0.29
L	Top-soil removal from unpaved road or soft-shoulder	90	20 – 50	Gravel, soil		22;	3 – 10;
					95	70 – 90%	560; 0.41

*Internal Revenue Service's 2015 Average Exchange Rates for Converting Foreign Currencies into U.S. Dollars (125.911 Yen per Dollar) was used for conversion.

Graphic courtesy EPA

- What other topics are of interest?
- How is the OE going to be obtained?
- Who is the main audience, scientist, model developers, engineers, end users?
- What forum should any information be published?
- Should it be promoted by a recognized group, EPRI, NEI, INPO?

September 2010

FY 2012 OMB Budget Briefing

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Topic 5 Lessons Learned

Example of Japanese Requirements and Expectations for Restart:

- Kashiwazaki-Kariwa 6/7 Restart Status

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Kashiwazaki-Kariwa 6/7 Restart Status

- Two ABWRs started in 1995 (KK6)/1996 (KK7).
- Although not affected by March 2011 GEJE, shutdown since March 2012 (KK6) and August 2011 (KK7).
- In December 2017, JNRA announced that KK6/7 met Japan's revised regulatory requirements, as specified in Article 43-3-9 of the Nuclear Reactor Regulation Act.
- In June 2020, TEPCO/TOSHPIBA formed Joint Venture Company to implement required safety measures
- Completion of required safety measure construction expected to be December 2020.
- Final inspections to be completed by April 2021; restart also requires approval from local community.

Graphic courtesy TEPCO

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Wide Range of Safety Enhancements Implemented (continued)

Severe Accident Countermeasures (effectiveness evaluations)

- Additional High-Pressure Alternative Cooling (HPAC) System pump
- 42 fire pump trucks on high elevation locations for LPCI
- Alternate circulation cooling system (to reduce reliance on venting)
- Improved depressurization and venting capabilities (alternate relief valves, additional N₂ cylinders, iodine filters)
- Remote and field-operated venting system with filtration
- Hydrogen monitors (in PCV and RB), hydrogen recombiners (RB), and water level measurement sensors (SFPP)
- Enhanced backup power capabilities (turbine-generator trucks, high voltage power trucks, and batteries)
- Enhanced long-term cooling (water reservoir and retention dam)

H₂ recombiners

T-G trucks

Water reservoir

Graphic courtesy TEPCO

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Wide Range of Safety Enhancements Implemented

Pipe supports

Improved piping seals

Leak-tight doors

Leak-tight covers

Higher elevation battery charging stations

Natural Disaster Countermeasures

- Seismic Events - Additional supports and reinforcements for piping and electrical conduits
- Flooding - New 15 m high seawall (maximum tsunami height 6 m)
- Flooding - Watertight doors, waterproofed penetrations, and enhanced protection for batteries
- Fire - Firebreaks between power station and forest

Firebreaks

Graphic courtesy TEPCO

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Wide Range of Safety Enhancements Implemented (Continued)

Procedures and Emergency Response Measures

- Implemented Incident Command System (US system for emergency response)
- Increased staffing
- Revised procedures (identify ~ 50 main parameters, with alternative and auxiliary parameters to use when primary sensors unavailable)
- Increased drills and training

Graphic courtesy TEPCO

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Effectiveness Evaluations confirm Adequacy of New Measures

Peak PCV pressures and temperatures significant lowered by alternate cooling system and venting

Graphic courtesy TEPCO

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C.4. System Analysis Code Updates and Other Related Activities

C.4.1. MAAP and Related EPRI Activities

Related EPRI Activities

Fukushima Uncertainty Evaluations and SAMG Research

Kelli Voelsing
EPRI RSM Program Manager

Reactor Safety Technology Expert Panel Forensics Meeting
November 12, 2020

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EPRI's Engagement in International Severe Accident Research

- Provide technical input based on EPRI's history of safety analysis and severe accident research
- Data and lessons learned to support ongoing enhancement and validation of MAAP
- Serve as a bridge to incorporate new lessons learned into Severe Accident Guidance used at nuclear power plants
 - Take results from laboratory research and generate practical guidance and models in EPRI codes
 - Engage with PWR and BWR Owner's Groups to inform/enhance operator training on SAMGs
- Engage to understand the diverse international interests in this area

International Severe Accident Collaboration

■ Phenomenology Projects

- Preparatory Study on Analysis of Fuel Debris (PreADES)
- Analysis of Information from Reactor Buildings and Containment Vessels of Fukushima Daiichi Nuclear Power Station (ARC-F)

■ Uncertainty Projects

- Reduction of Severe Accident Uncertainties (ROSAU)
 - Expected to provide additional validation data for MAAP's ex-vessel debris models
- Management and Uncertainties of Severe Accidents (MUSA)
 - EPRI role in MUSA is to extend the uncertainty work for BWRs performed in support of Fukushima to PWRs

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Key MAAP Improvements from Fukushima Lessons-Learned

- Detailed BWR primary system thermal hydraulic models
- Generalized containment fluid stratification model
- Reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) turbine performance models
 - RCIC model of self-regulating operational mode seen at 1F2
- Ex-vessel relocation and debris coolability modeling improvements
- New corium spreading and molten core concrete interaction erosion models

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Ongoing MAAP Development

- MAAPv5.06 being developed in parallel with MAAP6 modernization efforts
- Objective of MAAPv5.06 is to meet critical member needs ahead of modernized MAAP code
 - Small modular reactor modeling capabilities
 - Controlled enhancements (in-core pool scrubbing model, X-Quenchers, etc.)
- MAAP6 development includes significant progress on converted C++ code
 - Runtime and accuracy both seeing close agreement with MAAPv5.05 production code
 - Continued work on automated testing suite, modernized user interface, and user documentation
- Re-structured MAAP6 code will allow for more rapid deployment of new models and to meet industry needs (e.g., accident tolerant fuel and Gen III+ designs, etc.)

Fukushima Uncertainty Evaluations

Uncertainty Analysis Methodology

- MAAP Uncertainty methodology was developed for Fukushima assessments:
 - Define baseline event scenarios (i.e., best estimate)
 - Identify and parameterize uncertainties related to scenario definition
 - Identify uncertain severe accident modeling parameters
 - Develop uncertainty model for uncertain parameters
 - Conduct sensitivity analyses to identify importance of various parameters to severe accident progression
 - Conduct uncertainty analyses with more limited set of parameters varied across their probability distributions

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Parameter Uncertainty Models

- Probability distributions assigned to parameters based on what is believed to be their “correct” value
 - “Correct” meaning the value that leads to most accurate simulation of a particular event
 - Accuracy of probability distributions directly related to the state of knowledge for a given parameter
- Fukushima uncertainty assessments applied EPRI methodology to range of MAAP modeling parameters applicable to Mark I BWR designs
 - Approach designed to be extensible to all PWR and BWR modeling parameters in the MAAP code

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Fukushima Uncertainty Evaluation Scope

- Uncertainty evaluations for Fukushima Daiichi Units 2 (1F2) and 3 (1F3) developed as inputs to best estimate scenario for each reactor
 - This scenario then used to refine debris location results based on observed data
- Best estimate representation relies upon detailed understanding of the MAAP5 in-vessel and ex-vessel modeling relative to the inferred accident sequence progression at 1F2 and 1F3
- Due to the uncertainty in accident progression timing, MAAP results were compared to observed data (i.e., containment pressure excursions) to identify key modeling assumptions required to match observed behaviors
 - Analysis identified two key modeling assumptions for 1F2 and three assumptions for 1F3

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1F3 RPV Lower Head Breach Representation

- MAAP5 tends to estimate relatively rapid relocation event
 - Leads to development of containment challenge soon after vessel breach
 - May not be representative of actual event
- Although likely that lower head failure initiated around 55 hours into event, containment pressure response not representative
- Therefore, best estimate MAAP scenario assumes later lower head breach time around 60 hours, which aligns with observed containment pressurization.

Observed containment pressure response used to inform best estimate RPV lower head breach timing in MAAP

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BWR SAMG Research

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BWR SAMG Uncertainty Studies

Objectives coordinated with BWROG Since 2015:

- Identify relative importance of severe accident:
 - Phenomena
 - Decisions
 - Actions
- Compare with respect to mitigating releases
- Used to inform the implementation of BWROG Rev 4 Emergency Procedures / Severe Accident Guidelines (EPG/SAG) at the utilities

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BWR SAMG Uncertainty Study Approach

- Use severe accident analysis accident signatures to probe SAMG actions
- Address a wide-range of accident signatures and accident mitigation strategies
- Use existing latest severe accident guidance as part of the simulation
- Will take into account :
 - Potential failures of operator actions
 - Inaccurate diagnosis of plant conditions
 - Uncertainties in the severe accident simulation
- Optimize the implementation of SAMGs
 - Identify actions and decisions that should be emphasized in training and drills
 - Also identify those that can be covered in less detail
 - Inform the development of drills consistent with severe accident research

Leverages insights gained from EPRI MAAP/MELCOR Crosswalk and Fukushima

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SAMG Uncertainty Study Outcomes

- Aligned the core damage and containment event trees to best reflect EPG/SAG guidance
- Flow rates into containment adjusted to be consistent with current NEI guidance
- Over 10000 simulations completed
- Machine learning applied to this research
- Graphical macro-visualizations developed

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SAMG Uncertainty Study Outcomes Machine Learning Applications – Finding Important Actions

- Initial analysis using “random forest” algorithm show primary containment venting as the most important decision point in containment event tree
 - Venting pressures are dependent on whether injection 1 hour after core damage is successful
- Influential actions are determined from machine learning theory:
 1. Blowdown on MSCRWL*
 2. External injection w/i 1 hour after core damage
 3. Vent at PCPL, PSP, or low pressure
- If operators successfully complete these three most influential actions, Csl release to the environment will be between 0 and 5 kg (< 6% initial Csl)

* Minimum Steam Cooling RPV Water Level

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SAMG Uncertainty Study Outcomes Example Visualization

If operators successfully complete all three actions, MAAP results in a likely small Csl release

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C.4.2. MELCOR Updates and Related SNL Activities

MELCOR Update

Lucas Albright, Nathan Andrews, David L. Luxat
Severe Accident Modeling/Analysis Department

Sandia National Laboratories is a multi-mission laboratory managed and operated by National Technology & 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

2 MELCOR – Technology-Neutral Nuclear Safety Modeling and Simulation

Nuclear Reactor System Applications		
Safety/Risk Assessment	Regulatory	Design/Operational Support
<ul style="list-style-type: none"> Technology-neutral <ul style="list-style-type: none"> Experimental Naval Advanced LWRs Advanced Non-LWRs Accident forensics (Fukushima, TMI) Probabilistic risk assessment 	<ul style="list-style-type: none"> License amendments Risk-informed regulation Design certification (e.g., NuScale) Vulnerability studies Emergency preparedness Emergency Planning Zone Analysis 	<ul style="list-style-type: none"> Design analysis scoping calculations Training simulators
Non-Reactor Applications		
Fusion	Spent Fuel	Facility Safety
<ul style="list-style-type: none"> Neutron beam injectors Li loop LOFA transient analysis ITER cryostat modeling He-cooled pebble test blanket (H³) 	<ul style="list-style-type: none"> Risk studies Multi-unit accidents Dry storage 	<ul style="list-style-type: none"> Leak path factor calculations DOE safety toolbox codes DOE nuclear facilities (Pantex, Hanford, Los Alamos, Savannah River Site)

MELCOR Modernization

Evolve MELCOR to support flexible evolution of physical modeling

- Motivated by insights from Fukushima Daiichi
- Enable flexible evolution of modeling to extrapolate beyond limited data and assess impact of non-parametric uncertainty
- Support U.S. national priorities to enable nuclear energy innovation and modernization

Improvements in code structure designed to strengthen

- Performance – shorter run times and more robust calculation
- Numerics – reduced numerical variance, improved accuracy, and improved characterization of uncertainty
- Capability – Removal of limitations that hinder modeling of some effects
- Maintainability – more extensible framework to allow code to adapt to new reactor designs, probe impact of evolution in state-of-knowledge, flexibility in future code/model development

Maintain currently validated physics

- Objective is not to add new physics models
- Evolve architecture to support, as needed, flexible addition/evolution of models

Multi-year effort

- Generalized Numerical Solution Engine Module
- MELCOR hydrodynamic modeling
- MELCOR In-Vessel Core Damage Progression
- Ex-Vessel Damage Progression
- Fission Product Release and Transport Modeling
- Rationalization of Current MELCOR Code Base with Modernized MELCOR Code Base / Finalization of the Modernized Code

MELCOR Architectural Evolution

Traditional MELCOR code has integrated separate codes into an integral plant response system

- Numerical functionality interwoven with physical models (packages)
- Communication between packages relatively complex

5

MELCOR Hydrodynamics Modeling – Motivation to Evolve Control Volume Hydrodynamics

Fukushima simulations have pointed to need for enhancements to hydrodynamic solution scheme

Reformulation of the hydrodynamic equation set and numerical solution scheme

- Introduce state-of-the-art residual-based solver
- Enhance overall robustness and efficiency of numerical solution

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MELCOR Hydrodynamics Modeling – Enhancements to Control Volume Hydrodynamics

Refactorization of CVH package interfaces to pass all CVH package Control Volume and Flow Path state variable data through calling list

- Code has been refactored to be “stateless”
- Development of well-defined interfaces to disentangle use of functionality from implementation
 - More flexibly swap in and out different implementations of physics or numerics

Revised and additional metrics

- Calculation of scaled residual vector enables check of solution goodness after the convergence criteria based on relative change is satisfied
- Scaled convergence metrics. For example, check on $\alpha_{j,\varphi}^n v_{j,\varphi}^n$ (the scaled quantity of interest) instead of $v_{j,\varphi}^n$ (unscaled velocity)
- Eliminates false positives (solution actually bad) and false negatives (solution actually good) that lead to code failures and/or code thrashing

MELCOR In-Vessel Core Damage Progression

Generalization of the code architecture

- Reduce numerical variability
- Support flexible injection of new physics representations of degradation processes
- Expand scope of uncertainty that can be probed for insights to inform decision-making

Candling model

- MELCOR moves melting material down a component surface (candle) in a single time step
- Such modeling limits oxidation and heat transfer for material in motion

Material interactions

- Enhancements to enable code to more flexibly represent new material systems
- Critical in evaluation of high-temperature material response range of advanced nuclear energy technologies
- Support generalized treatment of fission product speciation in novel systems
- Rationalize material system with ex-vessel (CAV) package

Lower head structure

- The lower head model to be rewritten to improve the numerical solution of the equations to better account for melting at the interior boundary

COR component objects and restructuring of COR database

- Allow templated creation of component objects
- Carry-over properties such as oxidation, hold-up, number of surfaces in contact with CVH, etc. for new components
- Enhance user flexibility to define COR component attributes for specific design needs

Ex-Vessel Damage Progression

Code numerics

- Separation of numerical solution and physical modeling functionality
- Introduction of a residual-based solver

Generalized geometry

- A key component of this effort also involves generalization of how ex-vessel geometries are represented
 - For example, the reactor cavity structures below the reactor vessel
- Significantly enhanced functionality to support greater severe accident modeling realism
 - Critical to evaluating efforts to credit insights from Fukushima Daiichi accident

Replace Vanessa with a generalized Gibbs free energy minimizer

9 Fission Product Release and Transport Modeling

Numerics enhancement

- The numerical solution and physical modeling functionality will be separated
- Introduction of a residual-based solver
- Modernization of the underlying radionuclide transport equations (in the MAEROS model)

Enhancement of RN class representations

- Generalized to facilitate more flexible addition of new classes
- The ability to clone radionuclide classes from an existing class
- Generalization of chemical interactions
 - Enable modeling radionuclides undergoing chemical reactions
 - Enable transformations between radionuclide classes

10 MELCOR Representation of Water Level Measurement

MELCOR has flexible control volume representation for hydrodynamic systems

Control function implementation of water level measurement from Fuel Zone A

- Captures calibration of instrumentation and error approximation

Calculation scheme

- Solve for variable leg and reference leg pressures at DP sensor
- Use simulated DP value to determine water at calibration conditions

Enhancements to Terry Turbine Modeling

Previous work added mechanistic RCIC models to MELCOR (2015-2017)

- Homologous pump model
- Terry turbine pressure and velocity stage models
- Rigid shaft model (torque-inertia equation)
- Miscellaneous features including optional sub-models that facilitate use of Texas A&M experimental data
 - Extra user-specified turbine and pump torque terms (control function capability)
 - Sensitivity coefficients for windage torque and driving torque multiplier
 - Turbine-only operation mode (no pump coupling)

Current work focuses on the use and implementation of mechanistic RCIC models

- Generic BWR demonstration input deck to show
 - MELCOR capabilities for modeling RCIC intervention in one or more selected severe accident scenarios
 - MELCOR as a tool for risk-informed decision making within the context of PRA, SA, and/or UQ type studies
- Support for Texas A&M University experimental program
 - MELCOR models of experimental facilities (e.g. GS-2 air facility, ZS-1 turbine/dynamometer facility)
 - Consultation with Texas A&M Nuclear Heat Transfer Systems Lab for ongoing experimental program

MELCOR RCIC System Modeling and Analysis

Generic BWR demonstration input deck modeling goals include:

- Demonstration of systems-level RCIC model configuration and operation for users' benefit
- Learning lessons about code performance, system response, and accident progression
- Production of a generally useful end product for the MELCOR user base (regulators, academia, industry)

Primary modeling emphasis: capture thermal-hydraulic response of the RCIC turbopump system

- Start with a scram and 2-hr run-up to DC power loss
- Eventually, RCIC enters a self-regulating mode
 - Initial over-speed when governor valve goes full-open
 - Self-regulation observed (without over-speed) thereafter

To investigate moving forward:

- Model sensitivities
- Types/forms of turbine losses (experimental insights)
- High-consequence, low-knowledge parameters

Severe Accident Uncertainty – Evolution toward Non-Parametric Uncertainty Assessment

Developing insights

- Severe accident progression – what is/are the end-states, the releases, the risks?
- Model form error – Are the model assumptions accurate? Can they be improved?
- Model biases – Are the models imposing inappropriate, non-physical, or otherwise incorrect structure on accident progression simulations?
- Unknown unknowns – What are we missing?

Expansion of the uncertainty space domain

- Inclusion of other forms of uncertainty is a more “complete” representation of reality
- Gross bifurcations may emerge (due to model differences, modelling gaps, etc.)

MELCOR Material Interactions Models

Eutectics Model (EUT Model)

- Eutectic reactions causing early failure of components
 - zirconium-based fuel cladding and inconel grid spacers (ZR/INC); 1400 K
 - B₄C and steel control blade cladding (SS/B₄C) ; 1520 K
- Composition-dependent mixture properties
 - Five pseudo-binary phase diagrams as surrogates for the complex, multi-component systems
 - Mixture solidus is evaluated as the mole-weighted combination of the solidus temperatures of every binary combination of material pairs in the mixture
 - Other properties of mixtures are defined as the mass-weighted averages of constituent properties
- Chemical dissolution of solids
 - MELCOR uses the experimental correlations reported by Hofmann, *et al.*, (1989)
 - Parabolic rates are an upper bound for dissolution and is used for all mixtures, including mixtures with a low Zr content for which the predictions become more inaccurate

Interactive Material Model (INT Model)

- Liquefaction temperature modification (MP_PRC: ZRO2-INT, UO2-INT; Relocation Temperature)
 - Approximates the effect of materials interactions on debris phase changes during core degradation
 - Uses alternate material classes – ZRO2-INT, UO2-INT, and B4C-INT – with identical characteristics to the pure ZRO2, UO2, and B4C material classes, respectively
- Secondary material transport
 - Transports unmolten “secondary materials” by candelung molten materials
 - Simulates both the dissolution of UO2 by molten Zr and relocation of oxide shell debris carried by candelung materials

Severe Accident Uncertainty – Evolution toward Non-Parametric Uncertainty Assessment

Not a “best-estimate” uncertainty analysis – not attempting to quantify the uncertainty in a traditional sense

Identify the underlying biases of each model through an “exploratory” uncertainty analysis

- Not using “best-estimate” distributions of input parameters or attempting to establish “best-estimate” distributions of FOMs
- Uniform distributions are utilized to promote coverage of the uncertainty space and perform a “blind” comparison of models
- Removal of a priori biases on input and result distributions to investigate model form bias

Uncertainty Assessment Simulations

- Model interactions and uncertainties
- Simulation length: 25 hours
- 2400 simulations (1200 each) perturbing all uncertain parameters

Uncertainty Assessment – In-Core Damage Progression

Core component intact mass fractions at 4.5 hours (left) and 6.5 hours (right)

Maximum component temperatures

Maximum fuel temperature and relocation temperature

Uncertainty Assessment – Debris Slumping and Beyond

Steam dome temperature and debris mass in the lower plenum at 5.5 hours

Drywell pressure and in-vessel hydrogen generation at 10.3 hours

Kernel density estimates are a useful visual aid for inspecting result clustering

Bifurcations in the dataset can indicate alternate accident progression pathways or transitions between core damage states

Thank you for your attention

APPENDIX D. Additional Evaluation of TEPCO Coolant Suspension Tests on Units 1, 2, and 3

During the U.S. discussions on Topic 3 (see Section 2.3.3), Dr. Farmer provided several implications from 1F1, 1F2, and 1F3 coolant suspension tests (tests in which water injection to the RPV was suspended to gain insights regarding RPV and PCV thermal response and water leakage locations and size). To supplement this information, Dr. Farmer developed this appendix.

Coolant suspension tests have been carried out by TEPCO for F1, 1F2, and 1F3 to support emergency planning. The results for 1F1 were presented at the Fall 2019 meeting and were integrated into the Forensics Effort FY2020 report.[2] For 1F2, tests were carried out in the April to May 2019 time-frame and some additional comments on these tests are provided in this appendix. Finally, 1F3 tests were carried out in February 2020, and a few comments on the consistency of these results are also provided here. The 1F3 results are discussed first.

D.1. 1F3 Coolant Suspension Test Results and Discussion

The coolant suspension test at 1F3 lasted 48 hours [82]. This test revealed some interesting and initially unexpected results. As shown in Figure D-1, when injection was terminated, the temperatures initially increased. After injection was reestablished, the temperatures initially decreased, but then showed a rise to levels above that observed before injection was terminated. When the location of core debris and water level in the PCV are considered, these temperature rises may be due to other phenomena, such as evaporation and convection associated with injected water re-contacting hot debris.

Figure D-1. Thermocouple responses and coolant injection flowrate (left) and RPV thermocouple locations (right) for 1F3 coolant injection suspension test. (Courtesy of TEPCO Holdings [104])

Similar behavior was observed in measurements from the PCV thermocouples; see Figure D-2. The decrease in temperature measured by thermocouple TE-16-003 (T3) may be understood if one considers that this thermocouple was located near the surface of the water. As water injection ceased, the water sur-

face level dropped, causing the junction to rise above the water level and be exposed to cooler air. When injection resumed, the water level rose and covered the junction causing the temperature to rise due to contact with warm water.

Figure D-2. Thermocouple responses and coolant injection flowrate (left) and PCV thermocouple locations (right) for 1F3 coolant injection suspension test. (Courtesy of TEPCO Holdings [104])

TEPCO found that their analytical model best matched the RPV temperature response data when they assumed that only a small fraction of the core debris (i.e., 10%) remained in the RPV. This prediction is consistent with the PCV examinations that indicate a very large amount of core debris (up to 3 meters deep) is present in the pedestal region; see Figure D-3 [3].

Figure D-3. Debris locations in 1F3 pedestal region (Courtesy of TEPCO Holdings [105])

From an accident management viewpoint, these results illustrate the importance of evaluating instrument response in terms of localized conditions (e.g., whether or not the instrument is submerged in water) that can significantly affect their readings.

D.2. 1F2 Coolant Suspension Test Results and Discussion

Two coolant injection suspension tests were conducted at 1F2.[106] The operational characteristics of these tests are summarized as follows:

- STEP-1 test: Water injection reduced from 3.0 to 1.5 m³/h

Principal result: maximum temperature increase of the RPV was ~5 °C from ~ 20 °C at the start of the test

- STEP-2 test: Water injection reduced from 3.0 m³/h to zero

Principal result: maximum temperatures on the RPV lower head rose at a rate of ~5 °C/h

It was concluded the temperature responses were in general agreement with model predictions.[106]

These results have been further analyzed in an attempt to gain additional insights. Three points that are considered are as follows:

- Energy conservation, and implications for parasitic heat loss from the PCV as well as the level of water injection required to maintain adequately low temperatures within the PCV
- Additional estimates of core debris mass remaining in RPV lower head based on thermal response
- Evaluation of the consequences of reducing or eliminating water injection into the PCV

To address the first point, an estimate of the decay heat load is needed for 1F2 when the coolant suspension test was conducted. The ANSI/ANS 5.1 decay heat standard [107] is used to obtain this estimate. Knowing the reactor power level of 2380 MWt and assuming 300 Effective Full Power Days (EFPDs) of irradiation time when the accident occurred, then the current (i.e., ~ 10 years after the accident) decay heat level is estimated to be at ~22 kW (see Figure D-4).

Figure D-4. Estimated decay heat in 1F2 applying information in [107]

As noted above, the water volumetric injection flowrate is initially 3 m³/h, which corresponds to a mass injection flowrate, $\dot{m}$, of ~3000 kg/hr. The temperature rise, ΔT , of the coolant as it passes through the PCV, ΔT , is in the range of 5-6 °C. From the fluid flow energy balance equation

$$\dot{Q} = \dot{m}c_p\Delta T \quad (D-1)$$

the heat rejection, $\dot{Q}$, by the coolant at the given flowrate and observed temperature rise is between 17 to 21 kW. Given uncertainties in the decay heat estimate, the heat rejection and decay heat estimates are in reasonable agreement. Their closeness also indicates that parasitic heat losses are minimal.

One key consideration for current on-site operations is the extent of contaminated water production due to ongoing cooling of the three units by water injection. It is noted that if the injection flowrate were cut in half, the temperature rise of the coolant as it passes through containment would still only be ~10-12 °C, which seems acceptable based on a thermodynamic viewpoint.

With respect to the second point, in Appendix G of the Forensics Effort FY2019 report[3], it is estimated that ~ 1/3 to 1/2 of the core mass remained in lower head for 1F2 based on muon tomography measurements augmented by a simple geometry argument (see Figure D-5). These coolant injection tests provide the opportunity to estimate the mass in the lower head from a different viewpoint. In particular, the thermocouples on the lower head provide data on the rate of temperature rise after injection is stopped, and this can be related to the thermal loading through Equation (D-1).

Figure D-5. Images of 1F2 obtained using muon tomography with scintillation detectors (Courtesy of TEPCO Holdings [108])

If we assume the (hemispherical) lower head has an inner radius of 2.5 m with a wall thickness of 16 cm and the steel density is 8000 kg/m³-°C, then the mass of the lower head is estimated to be ~ 54 MT. We further assume that: i) half of the core inventory is in the vessel lower head, ii) all decay heat in the debris is transferred to the lower head, iii) the steel specific heat capacity is ~490 J/kg-°C, iv) the lower head can be treated adiabatically, and v) a lumped capacity heat transfer approach can be used to estimate the head thermal response. Under this set of assumptions, the rate of temperature rise of the lower head is calculated to be 1.5 °C/hr from Equation (D-1). This is noted to be ~ 3 times the measured temperature rise rate of ~0.5 °C/hour.

If we revise assumption i) to 1/3 of the core debris remaining in the lower head, this would leave ~7 kW of decay heat therein. With this amount of heating, the predicted rate of temperature rise would be ~ 1 °C/hour, which is closer to the measured value. This simple analysis thus supports the previous muon-tomography-based inference that 1/3 to 1/2 of the core debris remains in the lower head.

Regarding the third point, as discussed above, the decay heat level in the debris in 1F2 (and 1F3) has fallen to a relatively low value estimated to be ~ 22 kW at the current time. Meanwhile, the PCV and reactor building offer large surface areas to dissipate heat via natural convection and conduction. During the accident, the question arose as to how long it would take before the decay heat in the affected units would fall to the point where the heat could be dissipated passively in this manner. A scoping analysis was carried out at that time to address this question.[84] The various heat transfer resistances for dissipating decay heat from within the PCV to the outside environment are quantitatively evaluated in Table D-1. The two most limiting heat transfer resistances are highlighted in red within this table.

Table D-1. Summary of assumptions on heat transfer resistance and boundary conditions invoked in [84]; assumptions based on the state-of-knowledge approximately one week into the accident.

Number	Description	Heat Transfer Assumption	Heat Transfer Conductance, W/°C	Potential Heat Removal Rate across Boundary at 100 °C; MW
1	Convection and/or conduction from in-vessel core debris to RPV wall	Based on extensive in-vessel retention R&D, assume decay heat can be passively dissipated through nuclear boiling on the exterior of RPV lower head (so heat transfer resistance is negligible)	Infinite	Infinite
2	Conduction across RPV wall		Infinite	Infinite
3	<i>Convection from RPV outer wall to drywell liner</i>	<i>Drywell liner surface area ~ 1500 m³; free convection heat transfer coefficient 5-25 W/m²-°C</i>	<i>7,500 -37,500</i>	<i>0.75 - 3.75</i>
4	Conduction across drywell liner	Thickness= 3.18 cm; thermal conductivity ~ 40 W/m°C; surface area ~1500 m ³	1.9 x 10 ⁶	190
5	Convection across drywell liner gap	Small, if gap is plugged, but significant if gap is filled with condensing steam or water; resistance assumed negligible	Infinite	Infinite
6	<i>Conduction across concrete shield wall</i>	<i>Thickness ~ 200 cm; thermal conductivity ~ 2.2 W/m-°C; surface area ~1500 m³</i>	<i>1,650</i>	<i>0.165</i>
7	Convection through reactor building	Difficult to assess not knowing building state and availability of ventilation pathways	Infinite	Infinite
8	Conduction through reactor building wall	Thickness ~100 cm; thermal conductivity ~ 2.2 W/m°C; surface area ~8500 m ³	17,000	1.7

Results of this early analysis are summarized in Figure D-6. This graph shows the predicted decay heat level in the three affected units (using [107]). In addition, bounding estimates of the heat rejection rate from the PCVs are plotted (as dashed lines) due to the combined effects of the heat transfer resistances between the PCV and the outside of the environment. The conclusion at that time was that it was not possible to cool the core debris without continued water injection for a number of years.

This scoping analysis can roughly be extended to the current time, as follows. The surface area of the shielding concrete/drywell surface is ~ 1500 m². Assuming a (pessimistically low) natural convection heat

Figure D-6. Comparison of decay heat levels in 1F1, 1F2, and 1F3 to bounding estimates of passive heat rejection rates from the PCV to the outside environment (based on information in [84])

transfer coefficient of $5 \text{ W/m}^2\text{-K}$, then the outer surface of the shielding concrete would need to rise on average 3°C above ambient to dissipate the 22 kW of decay heat. In addition, conduction heat transfer from the exterior to the interior of the shielding concrete (2 m thick) would require an additional 13°C temperature rise to dissipate decay heat across the available surface area. Thus, assuming an ambient temperature of 20°C in the reactor building, the resultant temperature in the PCV to dissipate decay heat would be 36°C given the above-calculated various temperature rises.

Although approximate in nature, this analysis indicates that terminating water injection would not lead to excessive temperatures inside the PCV, at least for 1F2. If this is possible, it would likely reduce demands for on-site treatment and storage of contaminated water. When the time comes to enter the PCVs to start D&D, injection could be restarted to scrub aerosols. However, this would be much more effective if the water is placed where the debris is being disturbed for removal, as opposed to letting it drain through the bottom of the RPV. If this warrants additional consideration, it would be useful for TEPCO to apply more detailed containment heat-up models to obtain more precise estimates of PCV temperature increases.

It is noted that the system-level severe accident codes (e.g., MAAP, MELCOR, and SAMPSON) have detailed containment models for applying the various heat transfer resistances cited herein; it may be worth exercising such models to explore this possibility.

APPENDIX E. 1F1 and 1F3 Combustion Event Evaluations

Investigations by the Nuclear Regulation Authority of Japan (NRAJ) are providing several new insights regarding the combustion events observed at 1F1 and 1F3. These insights are based on detailed evaluations of seismometer data, observations from higher resolution videos of the explosions, and field investigations at 1F3. As part of their efforts, NRAJ considered information in U.S. Forensics Effort reports.[2 through 4] In his presentation (see Section 2.2.1 and Appendix C.2.1), Mr. Yasui, NRAJ, expressed interest in future NRAJ interactions with U.S. experts on their findings. In particular, he expressed interest in collaborating on insights related to combustible gas effects.

As documented in [2 through 4] and in Section 2.3.4, U.S. discussions on Topic 4 did not lead to a consensus opinion about several aspects of the combustion events that occurred at 1F. Since the FY2021 Forensics Panel meeting, additional information was released by NRAJ [94 through 97]. In addition, the lead for Topic Area 4 discussion, Dr. Wison Luangdilok, provided additional information to support his presentation in Appendix C.3.4.[98] This appendix summarizes this information and related findings and recommendations. It discusses evaluations provided by NRAJ for each type of information: seismometer data; satellite and high resolution video images; and damage observed in field surveys. Recognizing there are significant uncertainties in available information, this appendix focuses on reviewing information to identify, and where possible reduce, uncertainties affecting accident management strategies.

Several U.S. experts (Drs. Sud Basu, Michael Corradini, Randall Gauntt, Wison Luangdilok, and Chan Paik) reviewed a draft of this appendix. Comments by this review panel were considered in the version included in this report.

E.1. Seismometer Data

Recent NRAJ evaluations of seismometer data [94,97] provide important insights about the accelerations and energy imparted to the ground during the explosions at 1F1, 1F3, and 1F4. Furthermore, seismometer data provide input regarding the magnitude and number of events that occurred at 1F3.

E.1.1. NRAJ Seismometer Data Evaluations

Figure E-1 shows the locations of seismometer at 1F and the peak accelerations recorded at each location at the time of the explosion. Comparisons of the waveforms recorded during the 1F1 and 1F3 combustion events (see Location C in Figure E-2) support that the 1F3 explosion was a multi-stage event. Using three-dimensional seismometer data and location-specific soil damping measurements, NRAJ applied a model to estimate the energy ratio imparted by explosions that occurred at 1F1, 1F3, and 1F4. Preliminary results (shown in Figure E-3) indicate the ratio of peak accelerations for 1F1/1F3 is approximately 2.1 and for 1F3/1F4 is approximately 10. Assuming the energy ratio is proportion to the square of the peak acceleration amplitude, the ratio for the energy imparted to the ground by the 1F1/1F3 and 1F3/1F4 explosions are 4 and 100, respectively.

Locations of Seismic Observation Recorders within the Fukushima Dai-ichi Nuclear Power Station

Maximum acceleration recorded by the seismic observation recorder at the occurrence of the explosion
at Unit 1, Unit 3 and Unit 4 R/B (Gal)

	Unit 1 Explosion March 12 at 15:36			Unit 3 Explosion March 14 at 11:01			Unit 4 Explosion March 15 at 6:12		
	South-north direction	East-west direction	Vertical direction	South-north direction	East-west direction	Vertical direction	South-north direction	East-west direction	Vertical direction
Observation point A	81	52	120	14	15	21	2	2	2
Observation point B	284	129	138	45	18	28	4	3	3
Observation point C	2,320※	2,392※	1,956	115	158	490	11	9	6
Observation point D	102	91	231	36	51	173	6	7	11
Observation point E	39	22	26	26	24	30	5	5	11

※This value is only for information as it was out of range (2,000Gal) of the sensor. 単位

Figure E-1. Locations of and peak accelerations measured by 1F seismometers (Courtesy of NRAJ [67]; Appendix C.2.1, Slide 26)

Figure E-2. Vibration records from 1F seismometers (Courtesy of NRAJ [67]; Appendix C.2.1, Slide 27)

Source: Nuclear Regulation Authority (<https://www.nsr.go.jp/data/000337867.pdf>)

Figure E-3. Comparison of measured maximum accelerations measured for combustions at 1F (Courtesy of NRAJ [94])

E.1.2. Findings and Recommendations

Although video images suggest forces exerted during the 1F3 combustion events were significantly larger, recent NRAJ evaluations of seismometer data indicate the 1F1 explosion imparted larger energies to the ground. Furthermore, NRAJ analysis of seismometer data suggest the 1F3 explosion was a multi-stage event with two stages of equal magnitude. Insights from estimated ratios of the energy imparted by the 1F1 and 1F3 explosions differ from insights inferred from visual images (the 1F3 plume is significantly larger than the 1F1 plume; Appendix C.3.4 and [98] suggest the 1F3 plume size and shape is indicative of a detonation event). Seismometer data, however, only provide an indication of the energy imparted to the ground. Nevertheless, NRAJ assessment results indicate multi-directional transmission and localized effects should be considered when assessing the energy associated with combustion events.

One review panel member, Dr. Chan Paik, commented that additional insights might be obtained if it were hypothesized that the 1F1 and 1F3 combustion events were both detonation events. He observed results from prior tests [109] suggest the 1F3 multi-stage combustion event could have occurred with higher initial gas temperatures than the 1F1 event. Prior test data also suggest that if there were equal hydrogen concentrations of gas present, the detonation pressure for the 1F3 multi-stage combustion might be lower than the detonation pressure for the 1F1 combustion event. This would account for the difference in measured peak accelerations between the 1F1 and 1F3 combustion events (the peak 1F1 acceleration being roughly a factor of 2 higher than the peak 1F3 acceleration).

Another panel member, Dr. Wison Luangdilok, observed that in addition to the topography of the area, soil damping effects, the relative locations of the explosion center and the seismometers, and the impact of structural barriers to blast wave propagation on the seismometer readings, may be important. It may also be important to consider differences in the 1F1 and 1F3 building designs and their impact on acting as barriers to blast wave propagation and measured accelerations.[110] Above the operating floor (5th floor), the 1F1 reactor building was a steel frame structure covered with steel plates; this framework was not designed to withstand internal pressure. The 1F3 reactor building was a reinforced concrete structure; the resistance to

pressure of the walls on its operating floor was much stronger.[111] The force imparted on 1F3 5th floor structures, which experienced significant damage, may have weakened the pressure wave and reduced accelerations detected by seismometers near 1F3.

E.2. Satellite and High Resolution Video Images

Evaluations of images obtained from a satellite and recently obtained high resolution video images provide additional insights about differences in the 1F1 and 1F3 explosions. As documented in Section 2.2.1 and Appendix C.2.1, NRAJ is working with Nippon Television Network Corporation to obtain high resolution images of the videos obtained by Fukushima Central Television (FCT) of the 1F1 and 1F3 explosions. Although the final videos were not available at the time this report was written, the ‘interim’ 1F3 image in Figure E-4 illustrates the potential benefit of this effort. In conjunction with this effort, NRAJ is evaluating enhanced images to gain insights about the combustion events and conditions within each building affecting these events. This section reviews insights from these NRAJ efforts [94,95] supplemented by information provided in Appendix C.3.4 and [98].

Source: Nuclear Regulation Authority (<https://www.nsr.go.jp/data/000334168.pdf>)

Figure E-4. Interim effort illustrating enhanced image obtained for 1F3 on March 14, 2011 (11:01) (Courtesy of NRAJ [95])

E.2.1. Combustion Event Insights

Evaluations of the enhanced video images led NRAJ to conclude 1F3 experienced a multi-stage event. NRAJ defined four key steps based on observed phenomena, such as building deformations, flame images, debris ejection events, and rising smoke. These enhanced images are being used to estimate the location, duration, and extent (e.g., plume size and shape, size of debris ejected with the plume, etc.) of phenomena occurring for each stage of the event. Reference [94] provides a more recent update for information shown in Slide 24 of Appendix C.2.1.

The NRAJ review of seismometer data supported their multi-stage hypothesis (see Appendix E.1). As observed in [98], this hypothesis is similar to the multi-mode explosion hypothesis, first advanced in the 2018 U.S. Forensics Report.[4] At that time, however, U.S. experts relied on un-enhanced images produced by FCT.

E.2.2. Localized Effects Impacting Combustion Events

To gain insights about differences in the 1F1 and 1F3 explosions, NRAJ evaluated the flame colors observed in high resolution videos and gases emitted from 1F3 in a satellite image. During the 1F1 and 1F3 explosions, the emitted flame was yellow-orange with no clear difference in color. NRAJ concluded this yellow-orange color suggests the presence of not only hydrogen but also flammable organic compounds. The brighter 1F3 flame, typically, indicates higher concentrations of flammable organic compounds. However, a satellite photograph immediately after the hydrogen explosions shows steam being released from the 1F3 shield plug, suggesting steam was present inside the reactor building at the time of the hydrogen explosion. Thus, NRAJ investigations concluded the reactor building contained hydrogen, flammable organic compounds and steam. Images in high-resolution videos indicate some combustion phenomena were localized, suggesting gas concentrations varied at different locations within the reactor building.

E.2.3. Findings and Recommendations

As emphasized in [98, 99], there is substantial experimental evidence relating the flame color to the type of fuel, the flame temperature, and gas concentration impurities. The review panel noted the potential for organic materials to contribute to combustible gas generation has not been considered in prior evaluations. However, panel members also observed that any conclusions regarding reactants based on flame colors must also consider ‘real world’ conditions in the reactor building prior to the combustion event. For example, the presence of sodium within the reactor building could also produce a yellow colored flame.

Uncertainties in the approach presented by Dr. Luangdilok (Appendix C.3.4) were also discussed by the review panel. This approach estimates the hydrogen mass based on the observed size of the fireball. Lacking detailed knowledge, it is reasonable to estimate the fireball size based on the smoke cloud. However, the mass of required hydrogen is very sensitive to the fireball diameter ($d_{fireball}$); it is approximately proportional to the cube of the fireball diameter ($d_{fireball}^3$). If the diameter of the fireball is reduced to 2/3 of the estimated 100-120 m, the mass of hydrogen required (1540 kg based on Slide 5 in Appendix C.3.4) is reduced to ~ 30% (~ 450 kg). It is recommended that U.S. and Japanese investigations continue to focus on how much combustible gas is required to produce the observed 1F3 smoke cloud diameter and height. In particular, theoretical calculations of a momentum driven jet and buoyancy driven plume calculations might be beneficial in such investigations.

Several panel members emphasized that large uncertainties remain in systems analysis code predictions of the mass of hydrogen generated in the core and the containment. There are also large uncertainties in predicting the timing of vessel failure and the contribution of MCCI to combustible gas generation. Some of these uncertainties might be reduced if the mass of combustible gas required to produce the 1F3 combustion can be estimated. Because accurate predictions of combustible gas generation are important for severe accident management, the panel also noted the importance of proposed 1F examinations efforts within the RPV (see Appendix B.1) and on-going ROSAU tests (see Appendices C.1.2 and C.4.1).

E.3. Damage Observed in Field Surveys

As documented in Section 2.2.1 and Appendix C.2.1, NRAJ completed field surveys, obtaining photographs and measurements of damage within the affected 1F units. During September and December 2020, NRAJ conducted two field investigations at 1F3.[94,96] Surveys, as illustrated by example images shown

in Figure E-5, revealed there was considerable rubble and damage (cracks and displacements) to several beams and walls on the third floor, but no significant damage to equipment. Examinations also indicated damage is localized. At some locations where walls were lost, no damage was observed on nearby equipment, such as control panels. NRA is performing evaluations to estimate the magnitude of the pressure pulse associated with the 1F3 explosion, using the measured beam displacements from field surveys and explosion timing and location information inferred from high resolution videos (see Appendix E.2). This section summarizes preliminary results from these evaluations, supplemented by information provided in Appendix C.3.4 and [98].

Figure E-5. Representative images obtained during 1F3 field survey; Beam 7816 (left) and Beam 7815 (right) (Courtesy of NRAJ [67]; Appendix C.2.1, Slide 8)

E.3.1. Pressures Required to Induce Observed Damage

Using beam design information (e.g., material, shape, boundary conditions), displacements measurements obtained during field surveys, and explosion durations inferred from high resolution videos, NRAJ is estimating the pressure pulses required to induce measured deformations and observed damage. As an example, NRAJ reported results for Beam 7816 in [94, 96]. Preliminary results from their model, which required many simplifying assumptions, such as the force was applied uniformly and the observed deformation is symmetric, indicate observed displacements (between 2 and 27 mm) could be induced if a pressure pulse (between 100 to 500 kPa) is applied for 20 to 100 ms (high resolution images indicate 30 to 50 ms as the duration of the ‘prior’ or initial explosion initiating on the 4th floor of the 1F 3 reactor building; see Section E.2.1).

To gain insights regarding the minimum hydrogen concentrations required to induce the observed damage, NRAJ completed an evaluation assuming the explosion occurred in a closed volume. For the Beam 7816 case, it was predicted hydrogen concentrations of approximately 5% could generate a 500 kPa pressure pulse.

NRAJ also completed evaluations, again applying the non-realistic closed volume assumption, to determine if the pressure pulses required to induce damage observed at other 1F3 locations could be classi-

fied as a detonation.[94] For several locations, preliminary results indicate the estimated pressure pulses might be classified as a detonation. However, field surveys indicate the associated damage each of these cases was localized. NRAJ also emphasized these conservative evaluations were preliminary; they have not conducted a systematic evaluation on this topic.

E.3.2. Findings and Recommendations

The NRAJ damage evaluation approach is appropriate, especially their efforts to consider multiple sources of information (field surveys and video images). Because of uncertainties in available input data, the use of a model with simplifying assumptions is appropriate for developing initial estimates for pressure pulses to induce observed damage.

Although Appendix C.3.4 and [98] support the hypothesis of the second explosion observed at 1F3 might be a detonation, there is too much uncertainty in available information for a definitive conclusion. Furthermore, it is not clear that such a determination is important with respect to future accident management strategies, the primary focus of the U.S. Forensics Effort. It is important for Topic Area 4 investigations to remain focused on identifying and reducing, where possible, uncertainties that impact accident management strategies.

E.4. Summary

As discussed within this appendix, new information related to the explosions observed at 1F1 and 1F3 have led to several findings and recommendations. This appendix lists these new insights and recommendations.

Insight: NRAJ and U.S. evaluations of the explosions observed at 1F1 and 1F3 are providing new insights with respect to accident progression at these units.

There are uncertainties in predicting combustible gas generation, mixing, and transmission. NRAJ evaluations indicate the importance of considering localized conditions (temperatures, gas concentrations, humidities, etc.) and multi-directional transmission effects.

Insight: Experts are unable to reach a consensus about the hypothesis of a detonation occurring at 1F3, but this is not significant with respect to accident management.

Recommendation: Topic Area 4 investigations should remain focused on identifying and reducing, where possible, uncertainties that impact accident management strategies.

There are uncertainties in estimating the conditions existing at various locations within 1F3. Although estimates of the forces exerted by 1F3 explosions are important for safely completing D&D, U.S. forensics evaluations should focus on obtaining insights affecting accident management strategies, such as the ability to predict combustible gas generation. Hence, U.S. experts emphasize the importance of proposed 1F examinations efforts within the RPV (see Appendix B.1) and on-going ROSAU tests (see Appendices C.1.2 and C.4.1).

Recommendation: Representatives from the NRAJ and the U.S. Forensics Effort should continue to communicate as new information related to combustible gas generation and ignition becomes available.

Recent NRAJ evaluations have provided several important new insights, including the following:

- Evaluations of seismometer data indicate the energy exerted by the 1F1 explosion imparted similar or higher forces to the ground.
- Evaluations of high resolution images support the hypothesis that 1F3 experienced a multi-stage explosion.
- Damage assessments also emphasize the importance of localized effects. Damage observed in some components suggests smaller, localized, detonations may be possible; but other nearby components were undamaged.

These NRAJ evaluations, however, are preliminary. Models applied to extract insights invoke many simplifying assumptions. NRAJ is actively discussing new information and insights with stakeholders in Japan. Representatives from NRAJ participating in the Forensics Panel workshop indicated U.S. participation was also valuable, citing the 1F3 explosion hypothesis identified and documented in prior Forensics Effort reports by the Topic Area 4 lead, Dr. Wison Luangdilok. Because of the insights being gained from NRAJ investigations, U.S. participants agreed that continued interactions with NRAJ were of interest.

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