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U.S. Efforts in Support of Examinations at Fukushima Daiichi - November 2022 Meeting Notes and Information Request Status

Nuclear Engineering Division

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U.S. Effort Support to Examinations at Fukushima - November 2022 Meeting Notes Meeting Notes and Information Request Status

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ABSTRACT

Information obtained from Fukushima Daiichi Nuclear Power Station (Daiichi) is required to inform future 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 2023 (FY2023) 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, 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.

Fukushima-related information and associated discussions during these meetings benefit operating, new, and advanced reactors. Significant safety insights have been and are continuing to be obtained in several areas: 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. Furthermore, Daiichi-related activities, such as code modeling improvements and analysis, testing, and new technology deployment efforts, have the potential to offer additional benefits to the operating fleet and new LWR and non-LWR designs.

U.S. evaluations of obtained examination information and input regarding future Daiichi examinations are of interest to several organizations within Japan. Since its inception, the U.S. has provided consensus input for high priority time-sequenced examination tasks and supporting research activities. In their Mid-to-Long-term Examination Plan for 1F investigations, TEPCO included all remaining U.S. consensus information requests and additional information requests they identified. TEPCO periodically provides reports on the status of these requests (reflecting D&D priorities, new insights from investigations, and new technologies that become available). Hence, U.S. experts agreed that it was appropriate for TEPCO to track and prioritize these information requests as D&D progresses. U.S. experts will continue to review and comment on the information obtained from examinations and, as needed, provide additional details and relevant background material to support future examinations. As documented in this report, several other items, such as additional details on information requests pertaining to ex-vessel examinations, relevant references from prior research, additional documents to provide insights regarding recent investigation findings, and reviews of recently released documents, were agreed to during the FY2023 meeting.

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ACRONYMS AND ABBREVIATIONS

ACE	Advanced Containment Experiments
ADS	Automatic Depressurization System
AET	Advanced Engineering Training
AFW	Auxiliary Feed Water
ALPS	Advanced Liquid Processing System
AM	Accident Management and Prevention
AMUG	Asian MELCOR User Group
ANL	Argonne National Laboratory
AR	Augmented Reality
ARC-F	Analysis of Information from Reactor Buildings and Containment Vessels in Fukushima Daiichi Nuclear Power Station
ATF	Accident Tolerant Fuel
BSAF	Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant
BWXT	BWX-Technologies
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CAMS	Containment Atmosphere Monitoring System
CBT	Computer Based Training
CCI	Core Concrete Interaction
CD	Compact Disc
CLADS	Collaborative Laboratories for Advanced Decommissioning Science
CRD	Control Rod Drive
CSARP	Cooperative Severe Accident Research Program
CUW	Reactor Water CleanUp
CV	Control Voltage
CVD	Cold-Vacuum Drying
DAEC	Duane Arnold Energy Center
Daiichi	Fukushima Daiichi Nuclear Power Station
D&D	Decontamination and Decommissioning
DOE	Department Of Energy
DOE-NE	Department of Energy Office of Nuclear Energy
dP	Differential Pressure
DW or D/W	Drywell
EMUG	European MELCOR User Group
EPG	Emergency Planning Guideline
EPRI	Electric Power Research Institute
ESTER	Experiments on Source Term for delayed Releases
FAI	Fauske and Associates, LLC

FACE	Fukushima Daiichi NPS Accident Information Collection and Evaluation
FDADA	Database for grasping the in-core situation of Fukushima Daiichi Power Station
FRAnDLi	Fukushima Daiichi Radwaste Analytical Data Library
FE-SEM	Field Emission Scanning Electron Microscopy
FGMSP	First Generation Magnox Storage Ponds
FLEX	Diverse and Flexible Coping Strategies (for accident mitigation)
FNAA	Fukushima Nuclear Accident Archive
FP	Fission Product
FT-IR	Fourier Transform Infrared Spectroscopy
FY	Fiscal Year
GOTHIC	Generation Of Thermal Hydraulic Information for Containments
HEPA	High Efficiency Particulate Air
HMD	Head Mounted Display
HPCI	High Pressure Coolant Injection
HVAC	Heating, Ventilation, and Air Conditioning
HYMERES	Hydrogen Mitigation Experiments for REactor Safety
IAE	Institute of Applied Energy
IC	Isolation Condenser
ICP	Inductively Coupled Plasma
ICP-MS	Inductively Coupled Plasma - Mass Spectrometry
INL	Idaho National Laboratory
INPO	Institute of Nuclear Power Operations
IRID	International Research Institute for Nuclear Decommissioning
iRIS	Integrated Radiation Imaging System
IRM	Intermediate Range Monitor
IRSN	Institut de Radioprotection et de Sûreté Nucléaire
JAEA	Japan Atomic Energy Agency
KKNPS	Kashiwazaki-Kariwa Nuclear Power Station
LCS	Limestone-Common Sand (Concrete)
LEISAN	Large-scale Equipment for Investigation of Severe Accidents in Nuclear reactors
LERF	Large Early Release Frequency
LIBS	Laser Induced Breakdown Spectroscopy
LIDAR	LIght Detection And Ranging
LOOP	Loss of Offsite Power
LWR	Light Water Reactor
LWRS	Light Water Reactor Sustainability
MAAP	Modular Accident Analysis Program
MACE	Melt Attack and Coolability Experiments
MCCI	Molten Core Concrete Interaction

MCO	Multi-Cannister Overpack
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
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MSSS	Magnox Swarf Storage Silo
NA	No information available or Not Applicable
NANTeL	National Academy for Nuclear Training e-Learning
NDF	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NEUP	Nuclear Energy University Program
NPP	Nuclear Power Plant
NPS	Nuclear Power Station
NRA or NRAJ	Nuclear Regulation Authority of Japan
NRC	Nuclear Regulatory Commission
OECD	Organization for Economic Cooperation and Development
O&M	Operations and Maintenance
ORNL	Oak Ridge National Laboratory
PCMQ	Predictive Capability Maturity Quantification
PCV	Primary Containment Vessel
PLR	Primary Loop Recirculation
PM	Plant Maintenance
PN	Pin Number
PreADES	Preparatory Studies for Fuel Debris Analysis
PSD	Particle Size Distribution
PSEG	Public Service Enterprise Group
PSF	Plastic Scintillation Fibers
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group
R&D	Research and Development
RB or R/B	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
SA	Severe Accident
SAIL	SA Interactive Learning
SAG	SA Guidance (for BWRs and PWRs) or SA Guideline (for BWRs)
SAMG	SA Management Guideline (for PWRs)
SAREF	Safety Research Opportunities Post-Fukushima
SBO	Station BlackOut
SC or S/C	Suppression Chamber
SEM	Scanning Electron Microscopy
SIL	Siliceous (Concrete)
SFP	Spent Fuel Pool
SGTS	Standby Gas Treatment System
SLAM	Simultaneous Localization And Mapping
SLC	Standby Liquid Cooling
SNL	Sandia National Laboratories
SPAR	Standard Plant Analysis Risk
SOARCA	State-of-the-Art Reactor Consequence Analyses
SOUL	Smart Open Universe Learning
SRM	Source Range Monitor
SRV	Safety Relief Valve
SSB	Self-Shielded Boxes
SSWICS	Small-Scale Water Ingression and Crust Strength Tests
STSC	Sludge Transport and Storage Containers
TAMU	Texas A&M University
TC	Thermocouple
TCOFF	Thermodynamic Characterization of Fuel Debris and Fission Products Based on Scenario Analysis of Severe Accident Progression at Fukushima-Daiichi Nuclear Power Station
TDAFW	Turbine Driven Auxiliary FeedWater
TEM	Transmission Electron Microscopy
TEPCO Holdings	Tokyo Electric Power Company Holdings, Inc.
TG	Thermogravity
TIP	Traversing In-core Probe
TMI-2	Three Mile Island Unit 2
T/B	Turbine Building
TTEXOB	Terry™ Turbine Expanded Operating Band
TVA	Tennessee Valley Authority
U.S.	United States
UAV	Unmanned Aerial Vehicle

VIP	Vessel and Internals Program
VR	Virtual Reality
W	Westinghouse Electric Company
WARP	Website ARchiving Project
WDPA	West reactor cavity Differential Pressure Adjustment
WDX	Wavelength Dispersive X-Ray Spectroscopy
WW or W/W	Wetwell
X-#	Designation for PCV penetration number
XRD	X-Ray Diffraction
XRF	X-Ray Florescence
1F	Fukushima Daiichi Nuclear Power Station
1F1	Daiichi Unit 1
1F2	Daiichi Unit 2
1F3	Daiichi Unit 3
1F4	Daiichi Unit 4
1D or 1-D	One-Dimensional
2D or 2-D	Two-Dimensional
3D or 3-D	Three-Dimensional

U.S. Efforts in Support of Examinations at Fukushima Daiichi - November 2022 Meeting Notes with Updated Information Requests

1. INTRODUCTION

The Great East Japan Earthquake 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 or 1F). Much is still not known about the end-state of core materials in each unit that was operating on that date. Examination information is required to inform Decontamination and Decommissioning (D&D) activities, thereby improving the ability of Tokyo Electric Power Company Holdings, Incorporated (TEPCO Holdings) to characterize potential hazards and ensure the safety of workers involved with cleanup activities. This examination information also has important implications for the safety and operation of existing and future U.S. commercial nuclear power plants.

Similar to what occurred after the accident at Three Mile Island Unit 2 (TMI-2) [1], 1F examinations offer a means to obtain prototypic severe accident data from boiling water reactors (BWRs) related to fuel heatup, cladding and other metallic structure oxidation and associated hydrogen production, fission product release and transport, and fuel/structure interactions from relocating fuel material. Examinations from Daiichi are of special interest because multiple reactors were affected and the accident signature from each reactor appears unique. In addition, these units may offer data related to the effects of saltwater addition, reactor pressure vessel (RPV) failure, containment failure, and core/concrete interactions after RPV failure. Examination results are being used to update severe accident modeling and accident management practices, thereby enhancing global light water reactor operation and safety.

1.1. Objectives and Motivation

Since 2014, the U.S. Department of Energy Office of Nuclear Energy (DOE-NE) has sponsored an effort for U.S. and Japanese experts in plant safety and operations to meet and discuss recent investigation results from the affected plants at Daiichi. This DOE Forensics Effort 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 D&D plans for 1F.
- ***Objective 2:*** Evaluate obtained information for several reasons:
  - Gain a better understanding related to events that occurred in each unit at 1F;
  - Gain insights to reduce uncertainties in predicting phenomena and equipment performance during severe accidents;
  - Provide insights beneficial to future TEPCO Holdings D&D activities;
  - Confirm and, if needed, improve guidance for severe accident prevention, mitigation, and emergency planning; and
  - Periodically, update and/or refine Objective 1 information requests.
- ***Objective 3:*** Facilitate implementation of Japan-led international research efforts to support D&D.

Results from this effort are beneficial to the U.S. and to Japan. For Japan, U.S. involvement provides an independent evaluation of inputs to D&D activities and of information obtained from 1F examinations. Such evaluations are useful because of U.S. experience in light water reactor (LWR) plant operations, reactor safety, and TMI-2 post-accident examinations and defueling. 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. For the U.S., this effort provides access to prototypic data from three BWR core melt events with distinctively different accident signatures. U.S. experts apply examination information to inform component and performance survivability assessments, enhance accident progression and source term models, update accident management strategies and associated plant staff training, and preserve severe accident capabilities. Information gained from 1F is of benefit to global nuclear reactor safety. Japan leads several post-Fukushima international programs to inform and gain insights from the international community. The DOE Forensics Effort provides a means for U.S. experts to contribute to and benefit from such international efforts.

Since its inception, key findings and recommendations are documented in annual reports and other publications.[2 through 13] As documented in these publications, the U.S. has already gained significant safety benefit from the information obtained from the affected 1F units to reduce uncertainties in BWR severe accident progression and implement safety enhancements for BWRs, pressurized water reactors (PWRs), and future nuclear power plant designs. As uncertainties in modeling the 1F events 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.

1.2. Approach

The approach developed to ensure that objectives outlined in Section 1.1 are achieved relies primarily on expert panel meetings. Industry, university, and national laboratory experts participate in this process. U.S. Nuclear Regulatory Commission (NRC), U.S. DOE, TEPCO Holdings, Japan Nuclear Regulation Authority (NRA or NRAJ), Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF), and Japan Atomic Energy Agency (JAEA) experts also attend and inform participants during these meetings.

1.2.1. Objective 1 Activities

To complete Objective 1, expert panel meetings initially focused on developing a report during Fiscal Year (FY) 2015 with a prioritized initial list of information of interest to U.S. stakeholders.[9] Special attention was devoted to documenting why such information is important and how it will be used to benefit the U.S. nuclear enterprise.

Most information needs 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. Organized in tables per location [e.g.,

* 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.

the reactor building (RB), the primary containment vessel (PCV), and the RPV], information need documentation identified applicable units and other relevant factors (e.g., how information should be obtained, why it is needed, its expected use or benefits, and when it should be obtained).

1.2.2. 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.[14] 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.

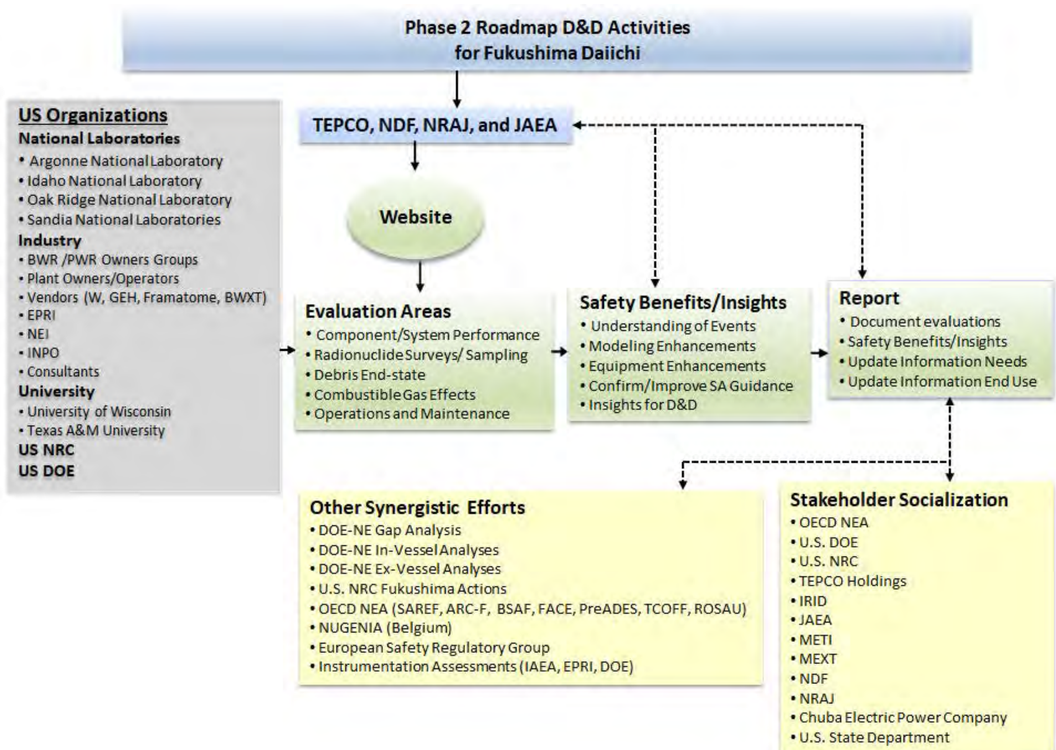

Figure 1-1. Objective 2 activities (organizations and programs defined in list of acronyms and abbreviations)

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 (for which it was deemed that important lessons could be learned to enhance reactor safety as well as plant operations and maintenance):

- 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 by TEPCO Holdings[15] 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 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 Holdings reports documenting unconfirmed and unresolved issues also receive special attention in the forensics effort.[16 through 22] Websites created by organizations within Japan, such as the Institute of Applied Energy (IAE)[23], TEPCO Holdings[24], JAEA [25], and METI [26], are important references for this effort. In addition, as discussed in Appendix B of Reference [5], a website for U.S. Forensics Effort participants has been developed to archive key references used by U.S. experts to complete these evaluations. At the time this FY2023 report was completed, there were nearly 800 references archived on this website (a significant increase over the 230 references archived at the time that Reference [5] was published).

Forensics evaluations have led to several types of safety benefits and insights. As shown in Figure 1-1, U.S. experts have prepared 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 [5 through 9] 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. During the FY2023 meeting, US experts decided to revise the process regarding consensus information requests (see Section 2.5).

1.2.3. 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. Results from this U.S. effort support several aspects of these synergistic efforts. Presentations at annual forensics meetings provide updates on these synergistic activities.

1.3. Report Objectives and Organization

As noted above, this FY 2023 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 FY2023 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 the agenda for the November 2022 U.S. Forensics Meeting. Appendix B lists examination information requests and provides tables with additional details for selected information requests. Appendix C includes presentations from participants wishing to include them in this publication, and Appendix D provides supporting information for Topic Areas 3 discussions.

2. FY2023 EXPERT PANEL MEETING HIGHLIGHTS

This section highlights presentations and discussions that occurred during the FY2023 Expert Panel meeting for the U.S. DOE sponsored Forensics Effort. Information discussed during the meeting is organized into five subsections: introductory U.S. presentations (Section 2.1); new information presented by Japanese organizations (Section 2.2); U.S. topic area evaluations (Section 2.3); U.S. systems analysis codes development and application activities (Section 2.4); and 1F forensics examination information request updates (Section 2.5).

2.1. Introductory U.S. Presentations

After a brief welcome by John Butler, who represented the Nuclear Energy Institute that provided the facility in which this meeting was conducted, there were three introductory presentations: an overview of the meeting agenda by Joy Rempe, Technical Lead for the DOE-sponsored forensics effort; an overview of the DOE-sponsored efforts to support forensics examinations by Damian Peko, the U.S. DOE program manager; and an overview of relevant U.S. NRC activities by Hossein Esmaili, the U.S. NRC Office of Nuclear Regulatory Research (RES) Fuel and Source Term Code Development Branch Manager.

2.1.1. Department of Energy

In her presentation (see Appendix C.1.1.1), Dr. Rempe observed that the FY2023 meeting was conducted as a hybrid meeting, allowing in-person and virtual participation. She reviewed the meeting agenda and provided a link where participants could access presentation materials. As indicated in Appendix A.1, presentations by experts participating virtually from Japan were scheduled early on the first day (to accommodate time zone differences). As indicated in Appendix A.2, approximately 60 experts in reactor safety and/or plant operations participated in this meeting. Dr. Rempe concluded her presentation by summarizing the status of U.S. consensus examination information requests and offering suggestions for future U.S. activities regarding these requests. Additional discussion and actions taken on this topic are found in Section 2.5.

In his presentation (Appendix C.1.1.2), Mr. Peko reviewed the DOE Forensic Effort approach and how results from this collaboration continue to benefit the U.S. and Japan. In addition to updated assessments of the potential hazards associated with external events, industry increased the equipment available to respond to beyond design basis events and improved operator guidance and training to respond to beyond design basis events. Fukushima-related efforts continue to improve our understanding of the accident progressions in each unit and the performance of structures, systems, and components during these accidents. As new information becomes available, the U.S. continues to benchmark (and revise, as warranted) models in systems analysis codes, such as the Electric Power Research Institute (EPRI)-sponsored Modular Accident Analysis Program (MAAP) code [27] and the NRC-sponsored Methods for Estimation of Leakages and Consequences of Releases (MELCOR) code, [28] and containment fission product transport codes, such as the EPRI-sponsored Generation Of Thermal Hydraulic Information for Containments (GOTHIC) code.[29] Because the safe and economic operation of the existing fleet is essential for acceptance of new reactor designs, he noted that DOE continues to value insights gained from evaluations of new information obtained from 1F related to BWR system performance and severe accident phenomena. Mr. Peko also emphasized the importance of other related DOE-supported efforts, such as the Boiling Water Reactor Owners Group (BWROG)-led Terry™ Turbine Expanded Operating Band Project (TTEXOB) and international Japan-led Organization for Economic Cooperation and Development Nuclear Energy Agency (OECD-NEA) projects. Finally, he noted the DOE and Ministry of Education, Culture, Sports, Science and Technology (MEXT) are exploring funding options to launch a new effort to develop and deploy new technologies deployed at Daiichi that may reduce plant maintenance costs and personnel exposures.(see Sections 2.2.2 and 2.3.5 for additional discussion on this topic).

2.1.2. U.S. Nuclear Regulatory Commission

Dr. Esmaili provided an overview of NRC-sponsored severe accident and source term research activities. As emphasized in his presentation (see Appendix C.1.2), these research activities provide important input to several agency regulatory activities. Dr. Esmaili reviewed the status of MELCOR development and application activities (emphasizing code modernization efforts and tasks to expand the code to simulate the performance of non-LWRs). He also reviewed other on-going international experimental and analysis programs related to code development and applications, including Japan-led post Fukushima activities [e.g., the Benchmark Study of the Accident at Fukushima (BSAF), Preparatory Study on Analysis of Fuel Debris (PreADES), Thermodynamic Characterization Of Fuel debris and Fission products based on scenario analysis of severe accident progression at Fukushima-Daiichi Nuclear Power Station (TCOFF), Analysis of Information from Reactor Buildings and Containment Vessels of Fukushima (ARC-F), and Fukushima Accident Information Collection & Evaluation (FACE) projects], the Institut de Radioprotection et de Sûreté Nucléaire (IRSN)-led Experiments on Source Term for Delayed Releases (ESTER program), and the U.S. NRC-led Reduction Of Severe Accident Uncertainties (ROSAU) program. Finally, he emphasized the importance of knowledge management in these activities, noting NRC efforts to preserve publicly-available information related to TMI-2 post-accident investigations and D&D activities can be accessed using CDs and websites.[30]

2.1.3. Summary

During the last decade, the U.S. nuclear enterprise has and continues to use Fukushima insights to enhance the safety of the operating fleet. The DOE-led U.S. Forensics Effort Expert Panel Meeting is an important avenue to gain these insights, which are important for operating LWRs, as well as advanced LWRs, small modular LWRs, and non-LWRs. In-person attendance, by representatives from Japanese and U.S. organizations, at the Expert Panel Meeting increased this year. Participants noticed that in-person attendance increased the benefit from this interaction (There was improved knowledge transfer, increased requests for additional information, and offers to provide information). Nevertheless, because of resource limitations, it is important to continue providing a virtual options to allow broader meeting attendance.

Introductory presentations and discussions led to two 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 Daiichi examinations that can be used to validate (and as needed enhance) accident management strategies as well as to reduce uncertainties in systems analysis codes.

Recommendation: Although knowledge transfer and interactions were improved with in-person attendance, U.S. Forensics Expert Panel Meetings should continue to include options for in-person and virtual participation.

2.2. New Information from Japan

Four Japanese organizations (NDF, JAEA, TEPCO, and NRAJ) provided detailed presentations with significant new information and important insights.

2.2.1. NDF

In his presentation (Appendix C.2.2) Hiroyuki Ito provided an overview of the NDF 2022 Strategic Plan for decommissioning of Fukushima, which was released in October 2022.[31] Mr. Ito's presentation emphasizes the holistic risk reduction approach NDF uses to coordinate efforts by various organizations participating in 1F D&D, such as TEPCO, JAEA, and the International Research Institute for Nuclear Decommissioning (IRID). The NDF 2022 Strategic Plan continues to focus on four aspects of 1F D&D:

- Evaluation and implementation of candidate methods and technologies for characterizing, processing, and disposing of radioactive wastes
- Issues related to near-term debris retrieval and selecting methods for expanded debris retrieval
- Strategies for discharging Advanced Liquid Processing System (ALPS)-treated water
- Strategies for removing fuel from spent fuel pools

The NDF holistic risk reduction approach considers the magnitude of radiation sources on the site (including the fuel debris within the PCV, the fuel within the spent fuel pools [SFPs], and contaminated water), the potential for its release (considering the effects of degradation of containment barriers due to factors such as aging or future seismic events), schedule, and resources. As new examination information is obtained, the step-by-step aspect of this NDF approach recognizes that uncertainties are reduced and updates are required. The strategic plan also continues to take actions to improve project management of organizations implementing D&D and to fund research and development (R&D) of new technologies required for D&D. Finally, the strategic plan notes the importance of international cooperation activities (e.g., identifying lessons from international experience, communicating future plans, and disseminating obtained information) and interactions with local communities and governments (e.g., taking actions to increase their participation and to revitalize affected communities).

During this presentation, Mr. Ito identified specific examples taken to reduce site hazards. For example, replacing flanged tanks with welded tanks reduces radiological risk from water leakage and manpower surveillance requirements. He also highlighted recent 1F1 investigations showing the exposed PCV rebar near the pedestal opening (and on-going assessments regarding the integrity of the 1F1 PCV). Plans for future examination and debris retrieval were also highlighted. In addition, Mr. Ito discussed the systematic step-by-step approach, which includes mock-up tests, to identify possible improvements for upcoming 1F2 examinations. At this time, he observed that several options are under consideration for 1F3 debris retrieval, including dry and submersion methods. No decision has yet been made regarding the method that will be selected. He also discussed the number of required debris hot-cell analyses, which will be conducted at the new JAEA facilities built at Okuma and at other hot laboratories in the Ibaraki area prefecture. These analyses will be used to reduce the anticipated volume of debris that must be stored as high level waste. However, there are concerns regarding the capacity of the available facilities and skilled workers to conduct such analyses. Research to develop remote analysis technologies that could reduce the number of required hot cell evaluations is underway. As part of this effort, an independent technology development effort by TEPCO Holdings, combined with a new Company ("Toso Mirai Technology Company"), was established in October 2022.

In subsequent post meeting interactions,[32] US participants noted difficulties they'd experienced related to managing the information obtained from the TMI-2 post-accident investigations and cleanup effort (e.g., data preservation for knowledge transfer in a project involving participation from multiple generations, websites becoming extinct, and obtaining and correlating information obtained from multiple organizations). NDF responded they were aware of the importance of consolidating knowledge on the 1F accident and decommissioning. In fact, consideration of fuel debris retrieval methods and radioactive waste storage and management is a high priority for NDF. Collected information is currently provided in English by TEPCO Holdings and JAEA through forums, such as the DOE Forensics Meeting and Japan-led OECD projects (e.g., PreADES, ARC-F, and FACE). The Website ARchiving Project (WARP) was started by Japan's National Diet Library to ensure that no information is lost.[33] Currently, information is divided into multiple sites (based on implementing agency budgets and projects). In their response, NDF suggested that participants of the DOE Forensics project consider the following websites:[23,24,25,26]

- Database for grasping the in-core situation of Fukushima Daiichi Power Station (FDADA) by TEPCO Holdings; https://fdada-plus.info/database/en/ [Click on [debrisWiki] to access the page for fuel debris study by JAEA and TEPCO];
- Fukushima Daiichi Radwaste Analytical Data Library (FRAnDLi) by JAEA; https://frandli-db.jaea.go.jp/FRAnDLi/index.php?country=e;
- Fukushima Nuclear Accident Archive (FNAA) by JAEA:Search for Reports Related to the Fukushima Daiichi Nuclear Power Plant Accident https://f-archive.jaea.go.jp/index.php?locale=en https://www.jaea.go.jp/atomic_portal/jaea_channel/28/; and
- Progress of decommissioning of Fukushima Daiichi Nuclear Power Station: https://www.meti.go.jp/english/earthquake/nuclear/decommissioning/index.html.

2.2.2. JAEA

During this meeting, JAEA presentations (see Appendix C.2.1) emphasized three topics: an update on recent 1F sample analysis by Hirotomo Ikeuchi; an update on large-scale experimental research related to debris formation by Yuji Nagae; and an update on the capabilities of iRIS (integrated Radiation Imaging System) by Yuki Sato.

Dr. Ikeuchi reviewed results from analyses of samples obtained from 1F, focusing on JAEA analysis of small 1F2 samples obtained during Japan's FY2021.† Using methods, such as Scanning Electron Microscopy (SEM), Transmission Electron Microscopy (TEM), and X-Ray Diffraction (XRD) studies, the material compositions and metallurgical phases of samples were characterized to gain insights about their origin, including the temperature ranges and oxidation conditions they experienced. Samples are either swipes or small grab samples from the X6 penetration and from the upper RPV well in the PCV. Discussions after this presentation focused on uncertainties in analysis results. While analytical methods have improved over the years, these samples are extremely small and their origin is generally unknown. The TMI-2 experience emphasized that it was difficult to form conclusions about accident progression from such samples.

In his presentation, Dr. Nagae described JAEA tests on control blade material and lower head penetrations in BWR RPVs. These tests, which are conducted in a new facility called LEISAN (Large-scale

† Japan's FY2021 started on April 1, 2021 and ended on March 31, 2022.

Equipment for Investigation of Severe Accidents in Nuclear reactors), focus on learning about the behavior of these metallic components during a severe accident (and the potential for low temperature eutectic formation). The LEISAN facility includes a fairly large, well-instrumented furnace capable of high heating rates, variable rate steam introduction, and measurement of aerosols and gases formed during the degradations process. In addition to gaining melting and relocation insights, JAEA will use LEISAN to characterize debris blockage formation and mechanisms, such as melting, creep failure and eutectic interactions, that could lead to RPV lower head failure.

In discussions following this presentation, participants emphasized the importance of this new facility, which can be used to rebuild Japanese testing capabilities. Participants offered several suggestions, such as: assessing test results using a systems analysis code; adding structural loading effects on creep failure in combination with material liquefaction; testing other components (e.g., BWR drain line materials, BWR channel boxes); testing other materials /components of interest to Accident Tolerant Fuel (ATF) programs; and considering insights from the OECD TMI-2 Vessel Investigation Program. To facilitate future JAEA investigations, U.S participants provided information on several of these topics (e.g., the NRC Lower Head Failure Program [34], the OECD TMI-2 Vessel Investigation Program Margin-to-Failure Calculations [35,36,37], and the OECD and NRC Sandia National Laboratories (SNL) lower head failure programs [38,39]). After the meeting, JAEA provided a more detailed response to suggestions and questions regarding the LESIAN facility [40]. In their response, JAEA expressed interest in considering the drain line penetration as a vessel failure pathway and the effects of non-uniform debris relocation on vessel failure. Prior to testing these additional phenomena, however, JAEA indicated they will be performing computational fluid dynamic analyses to assess the effects of such phenomena. In addition, JAEA noted LESIAN facility limitations will preclude considering structural loading effects on creep failure. However, such effects will be considered analytically. Finally, with respect to use of a systems analysis code, JAEA observed that their current objective is to develop a new model based on LESIAN facility data and then implement the new model into systems analysis codes.

In the third JAEA presentation, Dr. Sato provided an update on JAEA efforts to enhance the capabilities of iRIS, a remote method, based on Compton camera technology, for measuring and displaying contamination levels in three-dimensional easy-to-understand maps with dose rate information overlaid on photographic images. JAEA development of this technology has focused on compact lightweight Compton cameras, allowing measurements to be obtained by robots or a worker using a small backpack. In his presentation, Dr. Sato included videos illustrating how workers can easily detect hot spots and minimize their exposure in future work activities with the virtual reality capabilities of this system (In response to a request from DOE, JAEA provided access to these videos [40]). Currently, the system is used for detecting and displaying gamma radiation, but JAEA plans to expand the systems so that it may also be used to detect beta and alpha radiation. During the discussion, questions were raised about practical limitations currently associated with the system (because of interest in using it for normal plant operations and maintenance activities). It was observed that there are some resolution issues for hot spot detection in situations where many hot spots or high radiation background levels occur. In addition, it was observed that it is difficult to access some locations with the iRIS at this time. JAEA indicated that additional research is underway to improve resolution and increase the field-of-view for this system. There continues to be considerable enthusiasm about this effort by U.S. participants; several U.S. participants request that a joint U.S./Japan effort be launched to deploy these technologies for operations and maintenance (O&M) activities. Additional discussion on this topic may be found in Section 2.3.5.

2.2.3. TEPCO Holdings

During the first meeting day, TEPCO Holdings experts provided presentations (see Appendix C.2.3) on several topics: recent 1F1 PCV investigation findings and plans for future investigations by Michael Cibula; recent 1F1 investigation results by Kenji Owada; results from experimental investigations of materials within the containment that could generate combustible gases by Shinya Mizokami; and an overview of the recently released document, “6th Progress Report on the Investigation and Examination of Unconfirmed and Unresolved Issues related to the 1F Accidents”, by Shinya Mizokami. In addition, Kenji Owada provided an update of TEPCO’s Mid- and Long-Term Investigation Plan on the second meeting day.

In his presentation (see Appendix C.2.3.1), Dr. Cibula provided an overview of 1F1 PCV post-accident observations, from 2011 through recent images obtained using well-instrumented submersible Remote Operated Vehicle (ROVs), developed by IRID. Images (and dimensions) of structures within the PCV prior to the accident were included to facilitate understanding of the observed damage. Of special interest were recently obtained images of: ‘shelves’ and ‘lavacicles’ (Slides 8 and 29-37); exposed rebar with minor deformation (Slide 9); the height distribution of deposits (Slides 12-15); glittery deposits (Slides 16-17); and bulky and eggshell deposits (Slides 21-28). Tables speculating possible origins and compositions of observed sediment were also provided (Slides 18, 21, 23, 25, 27, 29, 34, and 38). The questions and discussions on this presentation were deferred until after an NRA presentation (Section 2.2.4), which dovetailed topics discussed by Dr. Cibula.

In his first presentation (see Appendix C.2.3.2), Dr. Owada summarized recent findings from recent 1F1, 1F2, and 1F3 investigations and plans for future near-term investigations. Combustible gas investigations (Slides 2 through 5) indicate that hydrogen generated during the accident had transferred from the PCV into other locations within each unit, and TEPCO Holdings is considering the impact of this finding on future D&D activities. In addition, Dr. Owada provided results from recent investigations of the 1F2 fuel handing machine remote control room, the 1F1/1F2 upper floor radiation dose surveys, the 1F2 shield plug, the 1F2 reactor building drawdown efforts, and the 1F1 through 1F4 Standby Gas Treatment System (SGTS) room investigations.

In his first presentation (see Appendix C.2.3.3), Dr. Mizokami reported recent results from TEPCO heating tests of BWR cabling [e.g., control voltage (CV), pin number (PN), coaxial], paints (e.g., inorganic, epoxy), and insulator materials (e.g., Urethane, Polyimide) samples. The objective of these tests is to quantify sources of combustible gases, not currently considered in systems analysis codes, that might have been generated during the 1F accidents. The test setup allows components to be heated in flowing hydrogen or steam. Tests conducted at 200 °C for 24 hours did not detect any flammable gas generation. At higher temperatures (up to 1000 °C), tests conducted in hydrogen produced carbon, hydrogen sulfide (in many cases), as well as numerous short chain hydrocarbons (methane, ethane, etc.). At these temperatures, results indicate more sample mass loss occurred and larger amounts of combustible gas were generated (CO_2 being the predominant gas generated) in tests conducted in steam than in tests conducted in hydrogen. In addition to uncertainties in the containment atmospheric conditions, the range of obtained values from these tests emphasizes there is still considerable uncertainty associated with the effort to quantify the amounts of combustible gas generated. Future testing will consider other materials present within the PCV (e.g., inorganic zinc paint, organic zinc paint, and nylon braided silicon-containing cable).

In his second presentation (see Appendix C.2.3.4), Dr. Mizokami provided an overview of the contents of the report, "6th Progress Report on the Investigation and Examination of Unconfirmed and Unresolved Issues on the Development Mechanism of the Fukushima Daiichi Nuclear Accident",[22] which was released by TEPCO Holdings on November 10, 2022.‡ This report provides updates on the following topics:

- Debris endstate diagram
- The cause for the high radiation dose rate observed in the southeast area of the first floor of the 1F1 reactor building
- Why high dose rates were not observed in the 1F2 auxiliary cooling water system
- Decrease in containment pressure in 1F2 on the morning of March 15
- Behavior of Suppression Chamber (S/C) pressure gauge in 1F2 after 21:00 on March 14
- Re-evaluation of the method to estimate core damage ratio in Mark I PCV
- Study on the water level in the 1F3 pressure S/C
- 1F3 accident progression after reactor depressurization
- Plant conditions during 1F3 Reactor Core Isolation Cooling (RCIC) operation
- Investigation of accident conditions based on sample analysis results

During this presentation, Dr. Mizokami observed that the only changes to the debris endstate diagram were decreases in the water level (due to recent seismic events). Dr. Mizokami also described recent efforts to reduce differences between time-dependent predictions of core damage based on dose rates measured by the S/C and Drywell (D/W) Containment Atmosphere Monitoring System (CAMS). These time-dependent core damage maps, known as 'core damage ratio evaluation maps', are used by operators during an accident. After power was restored on March 14, 2011, CAMS measurements were about one order of magnitude lower in the S/C. Although the trends in dose rate/damage curves in both maps were correct, it was found that the damage was under-predicted using data from the S/C CAMS detectors (e.g., although 100% core damage had occurred, evaluation maps indicated 55% core damage for 1F1, 35% core damage for 1F2, and 30% core damage for 1F3). Errors in core damage estimates were attributed to not properly considering several factors, such as the shielding by the S/C wall and the position of the CAMS detector. Improved analyses are reducing these inconsistencies. Dr. Mizokami emphasized that insights from this activity are being used to confirm there is no need to improve operator guidance of the operating fleet. For example, at the Kashiwazaki Kariwa Nuclear Power Units 6 and 7, detectors are positioned inside the PCV penetration and core damage ratio evaluation maps are no longer used by operators during an accident. In closing, Dr. Mizokami requested that, after this 6th progress report is translated, U.S. experts review it and provide comments. During the discussion, U.S. experts agreed to this request.

Dr. Owada's second presentation (see Appendix C.2.3.5), which was presented on the second day of this meeting as part of the information request update/revision discussion (Section 2.5), provided an update of TEPCO's Mid-and-Long-Term Plan for 1F investigations. His presentation demonstrated that TEPCO Holdings is continuing to address information requests to facilitate D&D, to improve the global nuclear community's understanding of how the 1F accidents progressed, and to improve safety of the operating fleet. TEPCO's plan includes all remaining US information requests and additional information requests identified by TEPCO (see Appendix B). Similar to the approach used by the U.S., TEPCO's plan groups information requests according to the location of the examination and the D&D schedule for obtaining the

‡ Release is currently just in Japanese; an English translation is forthcoming.

data. The grouping is updated as D&D progresses. His presentation not only includes a schedule for obtaining each information request (Slides 4-7 and 17) but also images depicting the location where the examination will occur (Slides 8 and 19-22). Additional discussion on this topic is found in Section 2.5.

As noted above, some of the topics presented in TEPCO Holdings presentations are closely related to other presentations and discussions. Additional insights, action items, and recommendations pertaining to the presentations by Drs. Cibula and Owada are found in Sections 2.2.4, 2.3.3, and 2.5. As noted in these sections, discussions after each of these presentations emphasized the importance of continued efforts to monitor and evaluate information obtained from the affected reactors at Daiichi. In addition, these discussions emphasize the benefits of in-person participation (e.g., several action items were requested by, and subsequently addressed by, meeting participants).

2.2.4. NRAJ

In his presentation (Appendix C.2.4), Masaya Yasui focused on two items: (1) damage observed in recent 1F1 PCV pedestal investigations (see Section 2.2.3); and (2) Cs-137 transport during a severe accident (as inferred from recent SGTS line and shield plug measurements).

For Item (1), he emphasized that 1F1 PCV investigations were not only to provide insights about accident progression, but were also needed to address NRAJ concerns related to the integrity of the PCV and its ability to support the RPV. Mr. Yasui emphasized several characteristics of recent TEPCO Holdings/IRID investigations, such as the increased height of debris (a 'mound') near the pedestal opening (suggesting that debris spreading did not occur), the loss of concrete around apparently-intact rebar on both sides of the opening, the morphology of formed crusts (porosity, smooth underside surfaces, bubble-like images on some surfaces, crusts, and shelves), and the observed damage to remaining structures and concrete. His presentation lists several major questions (and provided preliminary NRAJ thoughts on these questions):

1. Why the debris dropped from the RPV did not spread out?
2. How was the concrete part alone of the pedestal wall damaged?
3. How was the "crust" formed?

With respect to the first question, Mr. Yasui observed that the observed mounds in 1F1 as well as 1F3 (as high as 3 m for 1F3) could be related to the debris metallic content and higher viscosities /slower velocities associated with materials that relocate at lower temperatures. For the second question, he described possible options that might have led to the observed damage, including concrete degradation due to high temperature exposure, heat shock, silicone extraction, and phase changes. Heating tests on concrete samples are being performed to gain insights on this question (see Slide 16). Last, for the third question, he noted that none of the currently proposed options ('normal' crust formation as described in Slide 12, hydrothermal reactions as described in Slide 13, and a shell remaining after swelling due to gas production and release as described in Slide 14) fully explain the observed phenomena and welcomed thoughts from the international community. NRAJ expects more information from upcoming investigations and sampling analysis to address these questions (and related topics such as damage to the pedestal exterior wall).

There was considerable discussion regarding Item (1) and the information in the related presentation by Dr. Cibula (see Appendix C.2.3.1). Several clarifications were provided regarding the status of on-going investigations, details regarding component damage not mentioned in the slides, materials present in 'early-vintage' 1F1 Mark I PCVs, and potential effects from saltwater addition. Several theories, such as ash generation that could lead to observed crusts and shelves, were offered to account for possible observations. Of particular interest were comments by Dr. Farmer, during this and his subsequent presentation (see Section 2.3.3), indicating that some images shown in these 1F1 PCV investigations were consistent with images from prior tests investigating thermal-induced concrete degradation and molten core concrete interaction (MCCI). Several sources of publicly-available information, associated with prior U.S. and international research activities on these topics, were identified and provided to Japan after the meeting.

Participants concurred it was important for Japan to use additional information, such as images, sample analysis results, and separate effects testing data information, to address the three NRAJ questions. During discussions on this topic, it was observed that the lack of spreading was important for reducing uncertainties related to ex-vessel accident progression phenomena for the operating fleet but also for advanced LWR and non-LWR designs that rely on melting spreading and core catchers to mitigate accidents. In particular, participants emphasized the importance of data to discern whether observed phenomena are consistent with information from prior thermal-induced concrete degradation and MCCI tests (especially data related to ex-vessel debris coolability).

For Item (2), Mr. Yasui emphasized recent investigations on the impact of condensation and condensed water location on cesium transport within the SGTS lines and the shield plug. In the case of the shield plug, analyses indicate that elastic deformation along with condensation / water location assumptions are leading to predictions that better match measured contamination levels. A supplementary presentation [41] was referenced to provide additional details about measured contamination levels. Insights from these NRAJ calculations to consider elastic deformation of the shield plug were of interest because of synergistic U.S. activities described in Section 2.4.1.

2.2.5. Summary

These presentations emphasized how Fukushima-related information and discussions of this information at DOE Forensics Expert Panel meetings continue to benefit the U.S. operating fleet as well as new LWR and non-LWR design efforts. These interactions also emphasize how U.S. experience in severe accident research and plant operations can benefit future Daiichi D&D activities.

In addition to focusing on waste processing and disposal, debris retrieval, and treated water release, the 2022 NDF Strategic Plan emphasizes the systematic holistic risk reduction approach, which includes learning from international experience and sharing information and lessons learned. Discussions indicate that NDF is well-aware of (and is addressing) lessons learned from prior U.S. experience from TMI-2 post-accident investigations and cleanup activities. An impressive suite of sample examination techniques is being used to characterize 1F samples. However, U.S. participants cautioned that the TMI-2 experience indicated that it is difficult to make extrapolations about accident progression from a limited number of samples, especially when samples are small in size and when details about the sample's origin are not well characterized. Participants were impressed with the well-instrumented, LEISAN facility, noting that it can be used to rebuild Japanese testing capabilities and gain 1F insights. In addition, participants were impressed with JAEA progress on developing new technologies for D&D. TEPCO and NRAJ 1F investigations, especially from within the 1F1 PCV, sparked considerable discussion. In several cases, U.S. participants identified (and subsequently provided) sources of publicly-available information that might be useful to future JAEA, TEPCO, and NRAJ investigations.

These presentations and associated discussions led to the following 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 can be used to validate (and as needed, enhance) accident management strategies as well as to reduce uncertainties in systems analysis codes.

Recommendation: JAEA should consider expanding the LESIAN test program by evaluating other LWR vessel components of interest and the impact of test results on ATF fuel implementation efforts.

Recommendation: Additional efforts should be devoted to facilitate deployment of new D&D technologies from 1F for routine O&M activities. The U.S. should expedite efforts to launch a U.S./Japan effort to deploy new D&D technologies for routine O&M activities.

Recommendation: Japan should ensure that new images, sample analysis results, and separate effects testing data information address the three NRAJ questions related to 1F1 PCV investigations. In particular, it is important to obtain data related to the hypothesis that observed phenomena are consistent with information from prior thermal-induced concrete degradation and MCCI tests (especially data related to ex-vessel debris coolability).

Results presented in several presentations (e.g., recent forensics investigation data, images obtained from 1F1 PCV investigations, and combustible gas testing results) are discussed further in subsequent U.S. Topic Area presentations (see Section 2.3).

2.3. Topic Area Evaluations

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

2.3.1. Topic Area 1 - Component/System Performance

Examinations of components and systems within the RB, PCV, and RPV provide critical information related to their survivability, operability, and peak conditions (e.g., pressure and temperature) they experienced during the accident. Topic Area 1, which is led by Jeff Gabor, Jensen Hughes, and 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 guidance?
- Are analysis improvements needed?

Topic 1 area leads track component and system performance examination information in a table (Table 2-1) that they update each year based on new examination information and insights. This year, leads updated this table to reflect new insights from 1F1, 1F2, and 1F3 inspections and evaluations to characterize the status of RPV pedestals and deformation of shield plugs.

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

Location	1F1	1F2	1F3
X-100B PCV penetration[b]	Possible melted shielding material [42] No damage observed on outside [43]	NA	NA
X-51 PCV penetration[c]	NA	No damage observed; pressurized water could not penetrate blockage in standby liquid cooling (SLC) system line [44, 45]	NA
X-53 High Pressure Coolant Injection (HPCI) steam supply penetration (1F2/1F3)[d]	High dose rate measured [46]	No damage observed [47]	No damage observed [48]
X-6 PCV penetration (CRD hatch)	NA	Melted material. [49, 50] Melted material expected to be from O-ring and cable coating [51]	No damage observed from inside [52]
Equipment hatch	NA	NA	Water puddle [53, 54] unknown source
Personnel hatch and nearby penetrations	No damage observed [55]	NA	NA
HPCI pipe penetration[e]	No damage observed, but high dose rates measured; traces of flow and white sediment observed [46,55,56]	NA	NA
Traversing In-core Probe (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) [46,57]	Dose surveys do not indicate leakage from PCV through TIP guides. High dose levels in samples of materials from TIP indexer [58]	NA
Wetwell (WW) vacuum breaker line	Leakage on expansion joint of one line (X-5E) [59]	NA	NA
Drywell (DW)/WW vent bellows	Water leakage attributed to vacuum line above [59]	No leakage observed [60]	
DW sand cushion drain pipe	Leakage [61]	No leakage observed [60]	NA
SC water level	Almost full [21]; increased leakage observed following February 2021 seismic event [62]	Middle [21]	Full [21]; increased leakage observed following February 2021 seismic event [62]
DW Water level	~2 m[21]	~0.2 m[21]	~6 m[21]

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

Location	1F1	1F2	1F3
Torus room	Partially flooded [63, 64]	Partially flooded [65]	Partially flooded [65]
	Rusted handrails/equipment [42]	Non-rusted handrails/ equipment [42,66]	Non-rusted handrails/ equipment [42,67]
	NA	Some room penetrations tested, no leakage observed [68]	NA
Main Steam Isolation Valve (MSIV) room	Limited view obtained [69]	Water leakage cannot be observed. [70] Deterioration of HVAC ducting with sediment observed. Reactor well vent line confirmed open, but intentionally by operator prior to accident [71]	Leakage in Line D near bellows [72]
DW shield plugs	Reactor well shield plug displaced [73]	Possible leakage based on radiation measurement profile [74,75]	Leakage likely due to radiation measurements at head and presence of H_2 burn [21,78]
		Minor observed deformation in top plug not attributed to forces during event [76,77]	
DW head/flange	No obvious PCV flange deformations observed; but elastic stretching of bolts during event possible [79] Paint peeling observed [51]	Paint peeling observed.[51]	NA
RCIC or other low SC piping	NA	Suspected leak location, not confirmed [42]	NA
RPV upper head	NA	NA	NA
RPV lower head	Ex-vessel debris images, dose surveys, and sample examinations indicate failure [21,80,81]	Ex-vessel debris and images confirm failure [78]	Ex-vessel debris images confirm failure [78]
RPV pedestal	Significant concrete damage observed [75]	NA	NA
SGTS vent path	High dose levels in vent path confirms rupture disk (RD) operation [82]	High dose levels in vent path, without RD disk operation, indicates backflow from 1F1 vent piping into 1F2 vent piping [82]	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 [82]

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 AND ABBREVIATIONS" 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 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.

Topic Area 1 leads reviewed new insights gained on relevant topics, including 1F2 shield plug deformation and contamination measurements (see Sections 2.2.3 and 2.2.4 and [75,76,77]), 1F1 pedestal wall integrity insights [75], and impacts of the March 2022 earthquake on 1F structures [83,84,85,86], and hydrogen gas holdup in Residual Heat Removal (RHR) piping (Section 2.2.3 and [87]). As appropriate, leads updated Table 2-1 to reflect these insights.

In addition, Dr. Robb compared differences in recent 1F1 PCV examination results with results from a February 2013 analysis [88] of potential melt spreading scenarios using MELTSPREAD[89] and core-concrete interaction (CCI) using CORQUENCH[90] with input from 1F1 MAAP and MELCOR calculations. Of particular interest were differences in the measured heights of relocated materials and MAAP/MELCOR input assumptions/models that could be responsible for these differences. The comparison indicated the *a priori* code predictions for debris levels within and outside the pedestal doorway were much lower than remote observations. However, to prevent misinterpretation of the survey measurements, U.S. experts asked if it were possible to obtain a better visualization of 1F1 PCV debris height distribution. TEPCO agreed to investigate and provide such an image when it is available.

Recent forensics examination information continue to provide Topic Area 1 insights that can be used to reduce code modeling uncertainties and reduce risk for future D&D activities. Insights regarding ex-vessel PCV images were of special interest because of differences between measured heights of relocated materials and values predicted in prior MELCOR/MAAP-MELTSPREAD/CORQUENCH coupled calculations. In addition, recent shield plug investigations (e.g., 1F2) are of interest to on-going efforts to predict radiation transport and releases (Section 2.3.2).

Topic Area leads did not suggest any clarifications to existing examination requests or recommend any new examination requests (see Appendix B).

2.3.2. Topic Area 2 - Radionuclide Surveys and Sampling

Dose surveys and radionuclide deposition samples collected within the RB, PCV, and SFP are another important data acquisition area to support D&D activities. In addition to providing insights about component and system degradation, debris end-state location, and combustible gas effects, these samples and swipes can provide evidence of fission product release fractions and possibly of fission product species.

Topic Area 2, which reviews survey and sampling information, is led by Lucas Albright and David Luxat, SNL. In the Topic 2 presentation (see Appendix C.3.2), Dr. Albright compared recent MELCOR release predictions with site radiation measurements reported in [91]. This comparison considered the effects of events, such as open doors, water additions, and explosions, on predicted releases. Dr. Albright also reviewed results from a radiochemical analysis of drain water sampled during 1F1 venting [92] that indicate elemental iodine was a chemical form of released iodine. The aim of these SNL investigations is to determine whether code predictions are consistent with observations. Dr. Albright concluded that integral observations from the 1F accidents seem to be generally consistent with existing analysis results. However, he observed that additional examination information may be needed to support further refinement of systems analysis code models and current 1F accident progression input assumptions. During the discussions, NRAJ noted that they are updating their evaluation of this dose rate data and will provide their updated results to the U.S. when available.

Recent and ongoing investigations of the 1F2 shield plug have included enhanced spatial mapping of dose rates.[93,94] This was achieved through boring holes partway into the top shield plug with the goal of measuring dose rates in the underlying shield plugs. The recent measurements confirm prior measurements of significant Cs-137 contamination. Such data can support future analysis and model refinement for radionuclide species transport.

Recent forensics examination information continues to provide Topic Area 2 insights that can be used to reduce code modeling uncertainties and reduce risk for future D&D activities. Insights, such as observed events not considered in code calculations and chemical species assumptions, are of special interest because of their impact on estimated source terms.

Topic Area leads did not suggest any clarifications to existing examination requests or recommend any new examination requests.

2.3.3. Topic Area 3 - Debris Endstate

The expert panel also selected debris end-state as an area of emphasis with respect to examination information. The end-state of debris is an important finding from forensics inspections and critical for benchmarking, and as needed, developing new models within severe accident analysis codes. Debris end-state location information is of particular interest at Daiichi because comparisons can be made between the multiple units that were affected. In addition, it is desired to gain insights about debris coolability, the effects of saltwater, and debris spreading from examinations. Information about debris end-state is also required for successful and safe completion of D&D activities. During the FY2023 meeting, Topic Area 3 discussions included two presentations: a presentation by Marty Plys, Fauske and Associates, LLC, (FAI) on data needs to support development of a passive interim storage container design and a presentation by Mitch Farmer, Argonne National Laboratory (ANL), on recent core debris location evaluations.

Dr. Ply's presentation (see Appendix C.3.3.1) emphasized the near-term importance of forensics investigation data to provide the technical and licensing basis for debris retrieval, packaging, and interim storage. Hanford and Sellafield experience indicates that passive vented interim storage of wet debris has the potential to significantly reduce 1F fuel debris storage costs. Because drying prior to storage is avoided, fewer process steps and debris characterization data are required with the passive interim wet debris storage option. FAI experience indicates that required debris characterization data are limited to the particle size distribution (which may be impacted by the debris retrieval method), the residual saturation level of the retrieved debris (which differs for drainable and un-drainable debris), the expected range of decay power within a container, and the debris thermal conductivity [recognizing the non-homogeneity of retrieved debris due to variations in factors such as composition and void fraction, FAI indicated that their approach can tolerate 'reasonable' (up to an order of magnitude) uncertainty in this value].

Dr. Farmer's presentation (see Appendix C.3.3.2) focused on comparing and contrasting recently available observations from the 1F1 cavity floor and PCV floor with observations from prior OECD/MCCI experiments.[95,96][**] Of particular interest were insights regarding 'anchored' crust formation in these prior tests due to (i) reduced gas sparging which decreased the voided height of foaming debris; and (ii)

** As indicated by the crossed-out text on Slide 2 of his presentation, Dr. Farmer included slides regarding in-situ debris water ingression, but decided to focus on the first three bullets due to time constraints. Slides on this topic are included in his backup slides.

concrete-containing corium slumping which becomes more dense as it melts. In these tests, suspended ‘bridge crust’ would anchor to the test section sidewalls and form a gap as it separates from the underlying melt. His presentation includes images, from prior experiments, showing anchored crusts, concrete slumping, and reduced corium volumes (Slides 9 through 11). CORQUENCH models [90], which he developed to predict surface elevation reduction and crust anchoring, are also discussed (Slides 8 and 15-17). In addition, Dr. Farmer shared insights gained from subsequent tests to evaluate the impact of anchored crusts on debris cooling (noting that tests indicate periodic crust anchoring and breaching are expected to occur). Dr. Farmer noted that images from recent 1F1 PCV investigations appear consistent with images from prior tests investigating thermal-induced concrete degradation and MCCI (e.g., Slides 10, 11, and 12). As noted in Section 2.2.4, it was agreed that U.S. experts would provide publicly available documents discussing these tests to NRAJ. In response to a request from NRAJ, Dr. Farmer also provided access to his presentation (see Appendix C.3.3.2) for use at an upcoming public meeting on severe accident topics.

During this presentation, Dr. Farmer speculated about the origin of the white powder deposits observed on some structure surfaces within the 1F1 PCV (a question raised during discussions after the presentations by NRAJ and TEPCO on results from 1F1 PCV investigations). He indicated that prior experience suggests this powder is due to concrete degradation associated with radiation heat transfer from relocated fuel-containing debris materials.

To provide additional insights, Dr. Farmer summarized results from two scoping CORQUENCH calculations: one to simulate behavior in the pedestal doorway region and one to simulate behavior in the larger drywell annulus. Both calculations assumed basalt-based concrete and 1F1 PCV pressures predicted by MELCOR. These scoping calculations, which were run for 14 days (11.25 days of dry cavity ablation followed by cavity flooding up to 2 m), predicted crust anchor formation for most of the 14 days. Although there are many simplifying assumptions, code predictions indicate anchoring/failure events occur, crust ledge formations are possible, and upper debris surface temperatures remain below steel melting temperatures (precluding rebar ablation). If examinations can obtain additional images supporting crust breach phenomena after crust anchoring, it would be an important safety insight, confirming debris coolability insights gained from the ANL Melt Attack and Coolability Experiments (MACE) and OECD MCCI programs.

Recent forensics examination information continues to provide important Topic Area 3 insights. Of particular interest are recent images from 1F1 PCV investigations and insights that can be gained regarding ex-vessel phenomena, such as debris spreading, MCCI, and ex-vessel debris coolability. Important information is being obtained from 1F1 PCV investigations. Many observations appear consistent with results observed in prior MCCI tests, but additional images and sample examinations are needed. During the discussions, Dr. Farmer agreed to provide additional details regarding his insights from prior MCCI tests and the two scoping CORQUENCH calculations he presented. He also agreed to identify areas where model updates may be needed. This additional information is included as Appendix D to this report.

Sample examinations provide important information required for future D&D activities (offering the potential to reduce costs) and are useful for increasing knowledge about the accident progressions in each unit. Drs. Farmer and Plys reviewed, and updated, as needed, detailed information requests pertaining to future ex-vessel debris examinations [Requests PCV-3(a through e), PC-17 through PC-22]. Debris properties, such as porosity, morphology, and particle size distribution, are critical for several D&D activities (e.g., designing casks; developing processes for debris drying, storage, and transportation). Such data are also important for accident mitigation strategies (e.g., assessing how debris coolability phenomena affects water addition strategies).

In summary, no new information requests were recommended for Topic 3. However, detailed information supporting existing requests related to Appendix B debris sample examinations were clarified (including the need for material properties and detection of concrete oxide components).

2.3.4. Topic Area 4 - Combustible Gas Effects

The area of combustible gas effects was included as a fourth topic of investigation because it was recognized that damage within the affected units at Daiichi could provide important insights related to the sources for and transport of combustible gas and the ignition point and damage caused by each explosion. As discussed in [2,3], reviews of higher resolution videos have led NRAJ and TEPCO to explore the possibility that other sources of combustible gases beyond core component oxidation (e.g., fuel cladding, channel boxes, and absorber material) and ex-vessel MCCI contributed to the 1F3 and 1F4 explosions. As discussed in Section 2.2.3 and Appendix C.2.3.3, TEPCO and NRAJ are conducting heating tests of cabling, paint, and insulators to quantify possible sources of combustible gases generated during the 1F accidents. In Topic Area 4, Wison Luangdilok, H2 Technology LLC, reviewed recent TEPCO test results and provided additional suggestions for future testing.

To estimate potential amounts of combustible gas that could be generated during an accident, Dr. Luangdilok discussed TEPCO Holdings off-gassing test results in terms of grams of hydrogen 'equivalent' generated per kilogram of material component heated to 1000 °C. Results (Slide 7) indicate that the most hydrogen would be generated from CV and PN cabling. However, the estimated additional hydrogen produced is less than 6% of the combustible gas estimated to be generated by oxidation of other core components and will not significantly reduce the amount of combustible gas generation currently attributed to ex-vessel MCCI.

Dr. Luangdilok also reviewed proposed future NRA experiments to gain insights about the 1F3 combustion event. He suggested the proposed test program be expanded, considering a broader range of concentrations (obtaining more data in the hydrogen-rich end of the spectrum), mixed gases, geometry-specific fluid dynamic effects, and flame propagation phenomena. During the discussion, it was observed that TEPCO's 5th report on unconfirmed/unresolved issues[21] provided insights about the combustible gas origin and transport. It was also emphasized that the NRAJ-proposed tests are not trying to investigate 1F3 conditions. Rather, the scope is limited to demonstrating new capabilities for analyzing combustion events and assessing structural degradation from such events (because such damage could affect the ability to safely complete future D&D activities). After the NRAJ gains confidence in their new capability, additional test conditions may be considered.

Recent forensics examination information continue to provide Topic Area 4 insights that can be used to reduce code modeling uncertainties and reduce risk for future D&D activities. During the discussion, experts emphasized uncertainties in predicting combustible gas generation, mixing, and transport. On-going NRAJ and TEPCO testing of components, such as cabling, paint and insulators, is providing additional insights regarding potential sources of combustible gas generation not considered in current systems analysis code models. During the discussion, U.S. expert panel members reiterated a recommendation from the FY2022 report, that U.S. Topic Area 4 investigations should remain focused on identifying and reducing, where possible, uncertainties that impact accident management strategies. If an expanded test program is warranted, proposed conditions should first be explored analytically to assess their importance. It was noted that such assessments could be explored with the MELCOR user parameters controlling flame propagation and burn completeness.

No changes were proposed to examination requests related to Topic Area 4 in Appendix B.

2.3.5. Topic Area 5 - Operations and Maintenance (O&M)

Topic Area 5 focuses on insights that will allow industry to better predict and give directions on how to prevent and mitigate severe accidents as well as insights that can be used to improve plant O&M. In his presentations (see Appendix C.3.5), Randy Bunt, Topic Area 5 lead and representative of Southern Company and the BWROG, focused on two areas: an update on the BWROG Terry^{TM} Turbine testing program and other innovations spawned from 1F activities; and an update on efforts to launch a joint U.S. – Japan research program to develop and deploy new technologies to support activities beneficial for advanced and existing nuclear reactor O&M activities as well as 1F decommissioning activities.

Mr. Bunt reviewed post-Fukushima actions taken to prevent and/or mitigate fuel damage during beyond design basis events, including the following:

- U.S. Industry Diverse and Flexible Coping (FLEX) Program (with additional equipment at plant sites and two national response centers)
- Improved SFP level water level instrumentation and strategies to address cooling challenges
- Hardened containment wetwell vents (BWR Mark I and II containments)
- Alternate venting and water addition strategies
- Revised procedures and guidance and updated training [emphasizing the BWROG-developed computer-based Severe Accident Interactive Learning (SAIL) training and guidance].

He also elaborated on several industry-led efforts motivated by evaluations of 1F forensics information:

- Results from the BWROG-led Terry^{TM} Turbine Expanded Operating Band Project (TTEXOB),[106,107 108,109], an international collaborative effort between the BWROG, the IAE, the DOE [funding participation by Idaho National Laboratory (INL), SNL, and Texas A& M University (TAMU)], and the EPRI, are already being used by industry to enhance the safety of the operating fleet. Test results were used to expand and define operating limitations on Terry^{TM} Turbine operation in BWR RCIC and PWR Turbine Driven Auxiliary FeedWater (TDAFW) systems (and support the technical basis for relaxing the low pressure start up test for Terry^{TM} Turbines). Test data provided input on new RCIC system and HPCI performance models (including for operation at higher bearing oil temperatures). The new RCIC model was benchmarked using Tennessee Valley Authority (TVA) data in which the RCIC system ran after a tornado on April 27, 2011. Evaluations concluded the observed 1F3 RCIC operation, based on available plant operation instrumentation data, is repeatable.

- An early insight from 1F forensics investigations was that continued RCIC operation during some events is not strongly influenced by water ingestion.[2] This insight and TTEXOB testing, led to revised guidance and procedures that the RCIC trip on high water can be removed, which simplifies plant operation during Station BlackOut (SBO) events [allowing RCIC and HPCI system performance to be maintained and reducing Safety Relief Valve (SRV) cycling]. Namely, when the RCIC system is operating outside of its original design basis and being used to cool the reactor, procedures were revised to prevent operators from tripping the RCIC turbine (i.e., operators should adjust or divert flow) and to keep the turbine speed above a minimum threshold level. These revisions had a direct impact on recovery from the August 2020 Loss of Offsite Power (LOOP) event caused by a derecho at the Duane Arnold Energy Center (DAEC) plant.[97,98,99] Using the revised procedures (see Appendix E.2 of Reference [2]), the DAEC operators placed both the RCIC and HPIC in manual control, simplifying the plant's coast down and reducing SRV cycling. In addition, the DAEC event confirmed the importance of symptom-based procedures for Emergency Procedures Guidelines (EPGs) / Severe Accident Guidance (SAG) training.[††]

- The BWROG-developed SAIL training, which is part of the U.S. utilities commitment related to severe accident management to the U.S. NRC, is now being provided to licensed and non-licensed plant operators along with decision makers and implementors of accident management programs. This Computer-Based Training (CBT) emphasizes Fukushima case studies. The CBT is available on the National Academy for Nuclear Training e-Learning or NANTeL (U.S. utility members) and the Smart Open Universe Learning or SOUL (NRC and International BWROG members) platforms. SAIL relieves plant owners/operators from having to develop and provide their own training, resulting in cost savings. During the discussion, BWROG representatives observed there is interest by PWR owners/operators in the SAIL training (despite differences in plant systems and phenomena, the fundamental processes that operators should consider are similar). In addition, BWROG representatives suggested that it might be possible to make SAIL available (for a short period of time) to interested expert panel members for review and comment.

- The BWROG continues to have interest in new technologies being deployed to facilitate 1F D&D. In particular, the BWROG is interested in other applications for the following technologies: muon tomography systems; special purpose robots, drones, and Unmanned Aerial Vehicles (UAVs); portable gamma-ray imaging camera; infrared thermography; real-time monitoring with two-dimensional (2D) or three-dimensional (3D) visualization of radiation levels and temperatures; plastic scintillation fiber monitors; and centralized data system to optimize worker exposure. Such technologies offer the potential to improve plant maintenance and operations, similar to the way that some drone- and LIght Detection And Ranging (LIDAR)-based technologies are currently being used by the operating fleet. Possible benefits include: worker dose reduction, worker fatigue reduction, and improved visualization for work planning. As part of this topic, Mr. Bunt reviewed recent efforts by the DOE Forensics Effort (e.g., development of Appendix E.3 in the FY2022 report describing the benefits of these new D&D technologies and recommendations in the FY2022 report that a DOE effort be initiated to facilitate deployment of these new D&D technologies for routine O&M activities). He observed that a draft MEXT/DOE proposal for a joint Japan/US effort with university and industry participation has been developed and is being socialized with individuals at INL, DOE, JAEA, and NDF.

Finally, Mr. Bunt emphasized that the industry has already benefited greatly from the collective research and technology advances associated with the US DOE-led domestic and international 1F activities. It is expected that similar benefits will be obtained from future activities (including the proposed US/Japan effort to develop and deploy new technologies for plant maintenance and recovery).

In discussions after this presentation, expert panel members continued to express support for the CBT efforts that incorporate lessons learned from Daiichi examinations into severe accident training. Expert panel members also expressed support for the Terry™ Turbopump Test program, which has already led to important reactor safety insights and reduced plant operating costs. Panel member re-emphasized the importance of moving forward with the workscope described in the MEXT/DOE proposal and suggested

†† SAG training includes Severe Accident Guidelines (SAGs) used for BWRs and Severe Accident Management Guidelines (SAMGs) used for PWRs.

the workscope described in the MEXT/DOE proposal be funded under the DOE Light Water Reactor Sustainability (LWRS) and Nuclear Energy University Program (NEUP) efforts.

No changes were proposed to examination requests related to Topic Area 5 in Appendix B.

2.3.6. Summary

Discussions for each of the topic areas emphasized the continued importance of recent forensics examination information in providing insights to confirm (and revise, if needed) current guidance for severe accident response and to reduce uncertainties in modeling ex-vessel accident progression. In addition, the discussions and interactions occurring during this meeting emphasize the mutual benefit of this effort. Participants from the U.S. and Japan requested several items, such as additional information pertaining to prior research related to thermal-induced concrete degradation and MCCI research, additional visualizations of debris height distributions within the PCV, the use of a set of Topic Area 3 slides in subsequent discussions within Japan, and review comments on a translated version of the TEPCO document, "6th Progress Report on the Investigation and Examination of Unconfirmed and Unresolved Issues on the Development Mechanism of the Fukushima Daiichi Nuclear Accident". In all of these cases, actions were taken (or are planned) to address these requests.

These topic area presentations and associated discussions led to the following recommendations:

Recommendation: Because of potential D&D benefits, additional consideration should be given to requests related to debris examination information, including requests to characterize debris morphology (e.g., porosity, shape distribution, size distribution), debris thermal properties, and debris permeability and dryout limits).

Recommendation: Japan should emphasize that new images, sample analysis results, and separate effects testing data information address the three NRA questions related to 1F1 PCV investigations. In particular, it is important to review data that could provide insights regarding the hypothesis that observed phenomena are consistent with information from prior thermal-induced concrete degradation and MCCI tests (especially data related to ex-vessel debris coolability).

Recommendation: Additional efforts should be devoted to facilitate deployment of new D&D technologies from 1F for routine O&M activities. The U.S. should expedite efforts to launch a U.S./Japan effort to deploy new D&D technologies for routine O&M activities.

2.4. Systems Analysis Code Development and Application Activities

Systems analysis code are, in essence, the 'repository' for severe accident analysis knowledge. As data are obtained, existing models are benchmarked and new models are developed (as necessary) to simulate that data. The revised codes provide a bases for various industry and regulatory applications. Knowledge gained from 1F examinations is also used to reduce uncertainties in systems analysis and fission product transport codes. During the FY2023 meeting, representatives from EPRI and SNL provided an update on these and other related activities.

2.4.1. Related EPRI-Sponsored Activities

Matt Nudi provided an overview of related EPRI severe accident research activities. His presentation (see Appendix C.4.1) focused on three topics: on-going MAAP shield plug contamination investigations; MAAP enhancements; and predictive capability maturity quantification (PCMQ) method development activities.

Mr. Nudi first described an on-going EPRI analysis, using the MAAP code coupled with the GOTHIC code,[29] to assess the ability of these codes to reproduce the observed shield plug contamination measurements. Mr. Nudi emphasized that an important input for this analysis is the nominal gap area between the shield plug plates and the estimated gap area following shield plug deformation. He mentioned recent MAAP enhancements motivated by 1F insights. These include improved PWR ex-vessel relocation models, improved corium jet fragmentation heat transfer models, BWR suppression pool X-Quencher models (after T-Quencher models were added to MAAP 5.04) to better simulate observed mixing, in-core fission product scrubbing models to account for reductions when water level is above the core, and a water radiolysis model that considers beta- and gamma- radiation and radiation absorption. Finally, he described EPRI efforts to investigate the relative value of various code development and application activities (e.g., experimental data for model development, V&V, and application assessments). The PCMQ method is designed to characterize and quantify the applicability domain of various activities and identify gaps where additional activities are needed. It is expected the PCMQ method will help EPRI prioritize future activities.

In subsequent discussions, expert panel members emphasized the impact of forensics information on MAAP code models. Several model improvements have been made, and uncertainty analyses with the improved code have been used to optimize SAG training. It was observed that additional improvements were needed. For example, changes were needed so that the code could simulate the impact of accident conditions on instrumentation such as water level sensors. Mr. Nudi agreed that such improvements should be implemented (and there are plans to have them included in the next major MAAP release).

In subsequent interactions,[100] NRAJ provided additional shield plug gap and deformation measurement information.[101] Of special note, were two points. First, measured surface drops do not necessarily mean shield plug deformation. However, the surface drop measured for 1F2 is larger than drops measured for 1F5 or 1F6. NRAJ speculates that differences may be due to deformation that occurred during the accident (but evaluations are continuing).

2.4.2. MELCOR Update and Related NRC-Sponsored Activities

In this presentation, Drs. Luxat and Albright described the importance of using uncertainty assessments to ensure that the most risk-significant Fukushima insights are incorporated into MELCOR. In essence, SNL is assessing the impact of proposed modeling changes before they are implemented. Using results from State-of-the-Art Reactor Consequence Analyses (SOARCA) BWR uncertainty assessments [102], SNL performed an expanded assessment (with 900 MELCOR simulations) considering releases during a 48 hour Peach Bottom SBO event. Results were used to identify 21 important or ‘influential’ modeling parameters grouped into five categories (sequence-related, in-vessel accident progression, ex-vessel accident progression, containment behavior, iodine and cesium chemistry, and aerosol deposition). For example, in the sequence-related parameters, uncertainties pertaining to excessive cycling failures, thermal seizures, partial failures, and failures-to-reclose, were found to be influential on source term predictions. Based on their evaluations, SNL identified several items of importance that are expected to be gained from future 1F forensics investigations: RPV depressurization mechanisms, vessel integrity and breach insights, and ex-vessel melt spreading insights. Dr. Luxat characterized many of these parameters as 'fuses' that could lead to releases and suggested that many of these fuses may happen differently than predicted by current systems analysis code models and that these differences may impact figures of merit, such as Large Early Release Frequency (LERF).

The presentation focused on insights gained from the new eutectics modeling that has been implemented into MELCOR 2.2. This modeling reflects an approximate model of intermetallic interactions that capture the composition-dependent variation of interaction temperature necessary to effectively capture the extent of molten material formation occurring at elevated temperatures during core damage events. While this model is still evolving from the perspective of simulation performance and overall representativeness for a broad range of material systems [e.g., including possible ATF systems], results with the current modeling provide an indication of the key impacts on core damage progression expected through introduction of models that more fully represent the phase diagram and enthalpy of multi-component material systems expected during core damage accidents.

Relative to past modeling practices, initial results obtained using these new models indicate that higher mass fractions of molten debris occur in the discretized core cells of the MELCOR-simulated in-core and lower plenum regions of the reactor vessel. Debris located in core cells is also found to achieve higher temperatures. This may have an impact on the estimated energy release when debris relocates into water pools (e.g., when debris slumps into the lower plenum). It may also have an impact on the nature of debris relocating into the reactor pedestal upon RPV lower head failure. With higher temperatures, more molten debris pours into the containment and is expected to spread more effectively out of the reactor pedestal. The state of the debris relocating out of a failed RPV lower head, however, can be significantly impacted by the mode of lower head failure. Modes that exhibit an earlier failure when the debris is at lower temperatures (i.e., prior to large coherent molten pools forming in the lower plenum) are likely to exhibit more holdup on below-vessel structures and less extensive spreading in the containment. The inclusion of more representative modeling of debris thermodynamic states allows codes, such as MELCOR, to be used to interpret ex-vessel observations at Fukushima Daiichi and infer debris conditions at the time of lower head breach. Hence, continued evolution of this modeling in MELCOR is relevant to improving code capabilities to evaluate safety of current generation plants (including plants considering adopting ATF systems).

Subsequent discussions focused on insights gained from prior U.S. evaluations (e.g., [37], [103], [104], and [105]). In these discussions, experts emphasized the need to consider the relationship between uncertainties in assumptions related to relocated debris conditions (e.g., composition, timing, etc.) and vessel and containment failure modes (e.g., a localized versus a global failure). Finally, it was observed that there may still be gaps in our ability to identify all the parameters for which uncertainties exist. It was observed that, in addition to important insights already gained from 1F forensics investigations, additional insights could be gained to reduce uncertainties in known parameters, possible unknown relationships between these parameters, and in new parameters which have not been identified.

2.4.3. Summary

Presentations in this topic area indicate efforts are underway to update code models and prioritize future modeling revisions based on insights from 1F information. Discussions related to systems analysis code activities led to two recommendations:

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

Recommendation: Prior to updating MELCOR models, the risk importance of proposed changes should be understood.

2.5. Forensics Examination 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. Each year, these information requests have been reviewed and as appropriate, updated. Since FY2015, several new information requests were added, and the status of several U.S. information requests was modified. An important aspect of this process is to document why the information is needed and the expected benefit and use of the obtained information. U.S. participants factored in experience from TMI-2 examinations, prioritizing information that would be beneficial for 1F defueling efforts and for plant safety. In addition, participants considered information provided by representatives from TEPCO Holdings and other Japanese organizations (e.g., JAEA, NDF, and NRAJ).

However, as discussed in Sections 2.1 and 2.2.3, TEPCO's plan for mid- and long-term investigations includes all remaining U.S.information requests and several additional information requests identified by TEPCO.

During this FY2023 meeting, experts identified no new information requests. In addition, U.S. experts decided to change their information request process. U.S. experts agreed that it was appropriate for TEPCO to track and prioritize information requests as D&D progresses. This decisions was based several factors:

- TEPCO's mid- and long-term plan is comprehensive;
- All remaining U.S. information requests, as well as new TEPCO-identified examination needs, have been incorporated into TEPCO's mid- and long-term examination plan; and
- As new information and technologies become available, the status and priority of the remaining information requests changes.

Appendix B.1 has been updated to include information requests identified by TEPCO. Furthermore, the presentation of the information requests in Appendix B.1 has been simplified from what was documented in prior years (columns pertaining to when the information will be obtained and the status of the request have been deleted). It was agreed, however, that U.S. organizations should continue to monitor and evaluate information obtained from the affected reactors at Daiichi. As needed, U.S. experts should also assist by providing additional details to support information requests (the tables found in Appendix B.2) and helpful background information. For FY2023, several tables with detailed information were updated (see Appendix B).

In summary, examination information request discussion led to the following two recommendations:

Recommendation: Further iterations on the status and priority of information requests should be completed by Japan.

Recommendation: U.S. organizations should continue to monitor and evaluate information obtained from the affected reactors at Daiichi. As requested, U.S. experts should assist by providing additional details and relevant background information to support future examinations.

2.6. Summary

The DOE established the forensics effort to work with 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. FY2023 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, based on the insights, recommendations, and action items reported in this section, are summarized in Section 3.

3. KEY FINDINGS AND ASSOCIATED RECOMMENDATIONS

This section summarizes recommendations developed from the FY2023 U.S. DOE Forensics Effort meeting. Because of differences in new information presented and discussed during the FY2023 meeting, specific recommendations listed in Section 2 differ from those developed during FY2022. Nevertheless, these recommendations they can again be grouped into higher-level findings similar to Forensics Effort FY2022 findings.

- ***Finding 1:*** Fukushima-related information from Japan and discussions of this information at DOE forensics meetings continue to benefit the U.S. operating fleet as well as new LWR and non-LWR design efforts.
- ***Finding 2:*** U.S. evaluations of information from Fukushima and input regarding future examinations are of interest to several organizations within Japan.
- ***Finding 3:*** Fukushima-related activities, such as code modeling improvements and analysis, testing, and new technology deployment efforts, have the potential to offer additional benefits to the operating fleet and new LWR and non-LWR designs.

Each of these findings and related recommendations are discussed below.

Finding 1 and Associated Recommendations:

Fukushima-related information from Japan and discussions of this information at DOE forensics meetings continue to benefit the U.S. operating fleet as well as new LWR and non-LWR design efforts.

As emphasized in several presentations, the U.S. nuclear enterprise has and continues to use Fukushima insights to enhance the safety of the operating fleet. In addition to updated assessments of the potential hazards associated with external events, industry increased the equipment available to respond to beyond design basis events and improved operator guidance and training to respond to beyond design basis events. The BWROG-developed CBT ensures lessons learned from forensics examination are incorporated into severe accident training. This training is not only being used for operators and other decisions-makers for BWRs but interactions also indicate this training is of interest to the international community for BWR and other reactor designs.

Examination information continues to improve our understanding of the accident progressions in each unit and the performance of structures, systems, and components during these accidents. During the FY2023 meeting, new information was presented on recent 1F1 investigations and supporting research. Of special interest were images obtained from 1F1 PCV investigations and supporting tests to gain insights regarding the damage shown in these images. Results from these 1F1 PCV investigations offer the potential to reduce uncertainties in modeling ex-vessel phenomena, such as debris spreading, MCCI, and ex-vessel debris coolability. These insights are not only important for the operating fleet but also for new LWRs and non-LWRs that rely on melt spreading and core catchers to mitigate accidents. Of special interest were comments from U.S. experts that some of the images shown in these 1F1 PCV investigations were consistent with images from prior tests investigating thermal-induced concrete degradation and MCCI. To address question required for D&D and gain insights regarding accident progression, Japan will be conducting additional investigations within the 1F1 PCV, obtaining additional images and samples. In addition, Japan is conducting separate effects testing to better understand the observed concrete degradation and assess the remaining integrity of the PCV. Obtained information can also provide important safety insights regarding uncertainties in current models of ex-vessel debris coolability and spreading.

Recommendation: U.S. organizations should continue to monitor and evaluate information obtained from the affected reactors at Daiichi. Important insights continue to come from Daiichi examinations that can be used to validate (and as needed enhance) accident management strategies as well as to reduce uncertainties in systems analysis codes.

Recommendation: Japan should emphasize that new images, sample analysis results, and separate effects testing data information address the three NRA questions related to 1F1 PCV investigations. In particular, it is important to review data that could provide insights regarding the hypothesis that observed phenomena are consistent with information from prior thermal-induced concrete degradation and MCCI tests (especially data related to ex-vessel debris coolability).

Finding 2 and Associated Recommendations:

U.S. evaluations of information from Fukushima and input regarding future examinations are of interest to several organizations within Japan.

Although in-person meetings offer more direct communication between experts in reactor safety and operations, the FY2023 hybrid meeting (with virtual and in-person attendance) continues to stimulate broad domestic and international participation. As documented in Section 2, this communication included active meeting participation and the subsequent exchange of requests by U.S. and Japanese participants for additional information and reviews. For example, representatives from TEPCO and NRA, provided more detailed information about 1F3 shield plug measurement data. During the meeting discussions, U.S. experts agreed to:

- Provide publicly available references summarizing prior research results (vessel failure, thermal-induced concrete degradation, and MCCI).
- Review and update, as needed, the Appendix B.2 details for information requests pertaining to ex-vessel examinations. Detailed information for PC-3(a through e), PC-17 through PC 22 were reviewed and updated (as needed). These revisions were prompted by recent 1F1 PCV examination information and by updated information regarding what properties would be needed to implement wet interim storage options for debris removed from 1F.
- Provide an Appendix to this report with additional details regarding Topic Area 3 discussions about insights from prior MCCI tests and results from scoping CORQUENCH calculations (as well as areas where code models may need updates).
- Provide review comments on the report, "6th Progress Report on the Investigation and Examination of Unconfirmed and Unresolved Issues on the Development Mechanism of the Fukushima Daiichi Nuclear Accident" when a version has been translated into English.

The first three of these actions were completed by US experts (the second two items have been incorporated into this report). It is planned that the U.S. will complete the last item when the translated report is available.

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. In their Mid-to-Long-term Examination Plan, TEPCO included all remaining U.S. consensus information requests as well as additional information requests identified by TEPCO. Furthermore, TEPCO periodically provides reports on the status of these requests (reflecting D&D priorities, new insights from investigations, and new technologies that become available). Hence, U.S. experts agreed that it was appropriate for TEPCO to track and prioritize these information requests as

D&D progresses. U.S. experts will continue to review and comment on the information obtained from examinations and, as needed, provide additional details and relevant background material to support future examinations.

Recommendation: Although knowledge transfer and interactions were improved with in-person attendance, U.S. Forensics Expert Panel Meetings should continue to include options for in-person and virtual participation.

Recommendation: Because of potential D&D benefits, additional consideration should be given to requests related to debris examination information, including requests to characterize debris morphology (e.g., porosity, shape distribution, size distribution), debris thermal properties, and debris permeability and dryout limits).

Recommendation: Further iterations on the status and priority of information requests should be completed by Japan.

Recommendation: U.S. organizations should continue to monitor and evaluate information obtained from the affected reactors at Daiichi. As requested, U.S. experts should assist by providing additional details and relevant background information to support future examinations.

Finding 3 and Associated Recommendations:

Fukushima-related activities, such as code modeling improvements and analysis, testing, and new technology deployment efforts, have the potential to offer additional benefits to the operating fleet and new LWR and non-LWR designs.

Cognizant US organizations for systems analysis codes and containment fission product transport codes (e.g., EPRI for MAAP and GOTHIC and NRC/SNL for MELCOR and MACCS) continue to benchmark code models (and identify possible changes) as new information becomes available. Improvements have been implemented in MAAP and MELCOR models, and uncertainty analyses with the updated codes were used to optimize SAG training. However, post-accident evaluations indicate additional updates are warranted.

In addition to 1F examinations, there are related activities that have already benefited (and have the potential to provide additional benefits) to the global nuclear enterprise. For example, data from 1F2 and 1F3, supplemented by the BWR-led Terry™ Turbopump Test program data, have led to significant improvements in understanding RCIC performance and related operator guidance. This revised guidance was successfully used to improve operator response during a loss of off-site power event at the Duane Arnold plant. Expert panel members also expressed continued support for new CBT efforts that incorporate lessons learned from Daiichi examinations into severe accident training for operators and other decision-makers.

Of particular interest during the FY2023 forensics meeting were presentations describing a new JAEA test facility and advances in new D&D technologies being deployed at Daiichi. U.S. experts offered several suggestions for expanding the testing scoped of the new, well-instrumented JAEA LEISAN test facility, which can be used to rebuild Japanese testing capabilities and gain 1F insights. Expert panel members also continued to express interest in additional lessons that can be obtained in radiation protection and in using new technologies being deployed to facilitate 1F D&D for routine plants operations and maintenance.

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

Recommendation: Prior to updating MELCOR models, the risk importance of proposed changes should be understood.

Recommendation: JAEA should consider expanding the LESIAN test program by evaluating other LWR vessel components of interest and the impact of test results on ATF fuel implementation efforts.

Recommendation: Additional efforts should be devoted to facilitate deployment of new D&D technologies from 1F for routine O&M activities. The U.S. should expedite efforts to launch a U.S./Japan effort to deploy new D&D technologies for routine O&M activities.

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APPENDIX A. FY2023 Meeting Agenda and Attendee List

A.1. November 17-18, 2022 Meeting Agenda

Reactor Safety Technology Expert Panel Forensics Meeting

Meeting Agenda
November 17-18, 2022

Nuclear Energy Institute, 1201 F. Street, NW, Suite 1100, Washington, DC

Thursday, November 17, 2022

8:30 AM	NEI Welcome and Administrative Comments	J. Butler, NEI
8:35 AM	Meeting Overview	J. Rempe, Rempe and Associates, LLC
8:40 AM	Recent Update on 1F Sample Analysis	H. Ikeuchi, JAEA
	Experimental Research Related to Formation of Debris using Large-Scale Equipment	Y. Nagae, JAEA
	Visualization of Radiation Information using an integrated Radiation Imaging System (iRIS) based on Compact Compton Camera *(Virtual Participation)*	Y. Sato, JAEA
9:50 AM	2022 Strategic Plan *(Virtual Participation)*	H. Ito, NDF
10:35 AM	*Break*	*All*
10:50 AM	Welcome and Program Overview – DOE Activities, Plans, and Constraints	D. Peko, US DOE-NE
11:00 AM	Welcome and Overview - Related NRC Activities	H. Esmaili US NRC
11:15 AM	Recent Investigation Findings and Plans for Future Investigations	S. Mizokami, M. Cibula, and K. Owada, TEPCO
	Updates on Key Topics	M. Yasui, K. Iwanaga, and K. Konishi, NRA
12:15 PM	*Working Lunch*	*All*
1:15 PM	Questions and Answers on TEPCO and NRA Presentations	M. Yasui, K. Iwanaga, and K. Konishi, NRA
	Remaining TEPCO Presentations	S. Mizokami, M. Cibula, and K. Owada, TEPCO

Reactor Safety Technology Expert Panel Forensics Meeting

Meeting Agenda
November 17-18, 2022

Nuclear Energy Institute, 1201 F. Street, NW, Suite 1100, Washington, DC

<u>Thursday, November 17, 2022 (Continued)</u>

2:45 PM	Topic 1- Component /System Examination Information and Comments on Future Examination Plans *(Gabor Virtual Participation)*	J. Gabor, Jensen Hughes, and K. Robb, ORNL
3:30 PM	*Break*	All
3:45 PM	Topic 2 – Radiation Surveys, Sampling/Dose Calculation Insights, and Comments on Future Examination Plans	L. Albright and D. Luxat, SNL
4:15 PM	Topic 3 – Core Debris Examinations and Comments on Future Examinations Plans (*Plys Virtual Participation*)	M. Farmer, ANL and M. Plys, FAI
5:00 PM	*Adjourn*	All

Reactor Safety Technology Expert Panel Forensics Meeting

Meeting Agenda
November 17-18, 2022

Nuclear Energy Institute, 1201 F. Street, NW, Suite 1100, Washington, DC

Friday, November 18, 2022

8:30 AM	Topic 4 – Combustible Gas Effects - Thoughts on Cable Degradation Testing and Proposed Future Examinations (*Virtual Presentation*)	W. Luangdilok, H2 Technologies
9:15 AM	Topic 5 - Plant Operations and Maintenance Insights and Comments on Proposed Future Examinations	P. Ellison, R. Bunt, and B. Williamson, BWROG
10:00 AM	*Break*	All
10:15 AM	Related EPRI Activities (Code Updates and Applications, Comments on Proposed)	M. Nudi, EPRI
11:00 AM	Related NRC-sponsored MELCOR & Severe Accident Activities ▪ Water level measurement modeling ▪ Ex-vessel behavior modeling ▪ D&D areas of interest for future modeling updates.	D. Luxat, SNL
11:45 AM	Update on Mid- and Long-term Plan for Examinations	M. Cibula K. Owada, nd S. Mizokami, TEPCO
12:15 PM	*Working Lunch* Update to Consensus Information Requests / Consensus Comments on Proposed Future Examinations Next Steps ▪ Proposed letter report(s) ▪ Action items and schedule,	*All*
2:30 PM	*Adjourn*	

A.2. November 17-18, 2022 Meeting Attendees

Name	Organization
Lucas Albright	Sandia National Laboratories
Lake Barrett	L. Barrett Consulting LLC
Sudhamay Basu	McGill Engineering Associates
Steve Bier	Public Service Enterprise Group - Hope Creek
Randolph Bunt	Southern Nuclear Company, BWR Owners Group
John Butler	Nuclear Energy Institute
Shawn Campbell	U.S. Nuclear Regulatory Commission
Michal Cibula	TEPCO Holdings
Michael L. Corradini	University of Wisconsin-Madison
Phillip G. Ellison	GE-Hitachi, BWR Owners Group
Hossein Esmaili	U.S. Nuclear Regulatory Commission
Mitchell T. Farmer	Argonne National Laboratory
Jeffrey R. Gabor	Jensen Hughes
Randall O. Gauntt	Gauntt Technical Safety Associates, LLC
Taro Hokugo	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Hirotomo Ikeuchi	Japan Atomic Energy Agency
Izumi Ishii	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Hiroyuki Ito	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Kohei Iwanaga	Japan Nuclear Regulation Authority
Yutaka Kadoya	Embassy of Japan
Ryu Kaneko	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Karen Kirkland	Texas A&M University
Tatsuro Kobayashi	TEPCO Holdings
Koji Konishi	Japan Nuclear Regulation Authority
Shinichi Koyama	Japan Atomic Energy Agency
Steven Kraft	Kraft-Contente, LLC
Kohei Kurano	TEPCO Holdings (Electric Power Research Institute Guest Researcher)
Richard Lee	U.S. Nuclear Regulatory Commission (retired)
Wison Luangdilok	H2 Technology, LLC
David Luxat	Sandia National Laboratories
Hiroki Maki	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Donald Marksberry	U.S. Nuclear Regulatory Commission
Robert Martin	BWX Technologies
Shinya Mizokami	TEPCO Holdings
Yuji Nagae	Japan Atomic Energy Agency
Masaki Nakagawa	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Noriyoshi Nakamura	Nuclear Damage Compensation and Decommissioning Facilitation Corporation

Name	Organization
Tony Nakanishi	U.S. Nuclear Regulatory Commission
Junichi Nakano	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Matt Nudi	Electric Power Research Institute
Shuichi Ohashi	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Keisuke Okamura	Embassy of Japan
Yoshimi Ota	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Kenji Owada	TEPCO Holdings
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
Yuki Sato	Japan Atomic Energy Agency
Mike Salay	U.S. Nuclear Regulatory Commission
Yoshitaka Suzuki	Chubu Electric Power Company, Inc.
Hau Ueda	U.S. Nuclear Regulatory Commission (International Atomic Energy Agency assignee)
Tony Usles	U.S. Nuclear Regulatory Commission
Hiroji Wakabayashi	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Tadahiro Washiya	Japan Atomic Energy Agency
Paul Whiteman	Framatome
Bill T. Williamson II	Tennessee Valley Authority, BWR Owners Group
Akio Yabuuchi	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Yasunori Yamanaka	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Masaya Yasui	Japan Nuclear Regulation Authority
Zhe Yuan	U.S. Nuclear Regulatory Commission

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. Until this year, these information requests were reviewed and as appropriate, updated. As emphasize in Section 2, TEPCO's Mid-and-Long-Term Plan for 1F investigations include all remaining U.S. requests. In addition, TEPCO has identified several additional information requests. As discussed in Section 2.5, it was agreed during this FY2023 meeting that US experts should continue to monitor and evaluate information obtained examination information and, as needed, provide additional details and relevant background information to support future examinations.

Appendix B.1 presents 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. In developing the U.S. information requests, participants factored in experience from TMI-2 examinations. 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). In addition, Appendix B.1 also lists additional information requests identified by TEPCO in their mid-to-long term examination plan (see Appendix C.2.3.5).

Several items in Section B.1 are shaded in light purple. This designates that more detailed requests have been developed for these information requests.* The current version of these more detailed requests is 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. In response to discussions during the FY2023 meeting, table with supporting information for information requests PC-3(a through e), PC-17, PC-18, PC 20, PC-21, and PC 22 were reviewed. To reflect new information and associated discussions during the FY2023 meeting, changes were implemented in Tables B-7, B-8, B-10, and B-14. These changes identify additional isotope ratios that could provide insights about the extent of MCCI, more specific locations for obtaining images, the need for debris examinations to also detect control material information need and to characterize the chemical form of fuel-bearing materials.

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

Item[b]	What/How Obtained	Why	Benefit /Use
RB-1	Photos/ videos[c] of condition of RCIC valve and pump before drain down and after disassembly (1F2 and 1F3)	• Determine turbine condition. • Gain insights about status of valve and pump at time of failure [PWRs have almost identical pumps for AFW].	Impacts BWR AM strategies (cause of RCIC room flooding). Use to support RCIC testing project (for confirmation of testing results). Potential PWR impacts (e.g., modeling, AM strategies, etc.). May also be beneficial in engineering of systems and interactions with the plant; may reduce maintenance costs; may reduce FLEX requirements; may increase operator knowledge.
RB-2	Photos/ videos of HPCI System after disassembly (1F1, 1F2, and 1F3)	• Gain insights about degradation due to seismic events (1F1, 1F2, and 1F3) and due to operation (1F3). • Compare endstate of 1F3 (look for flaws) with the endstate of 1F1 and 1F2. If similar flaws are observed in all three units, it would be useful for assessing impact of the seismic event and of longer term operation.	Impacts AM strategies (equipment utilization). May also be beneficial in engineering of systems and interactions with the plant; may reduce maintenance costs; may reduce FLEX requirements; may increase operator knowledge.
RB-3a	Photos/ videos of damaged walls and structures (1F1)	• Determine mode of explosion in 1F1 compared to 1F3.	Understanding what happened; assist D&D efforts. Potential BWR improvements; Impacts BWR AM strategies and code models (venting and interconnection between units); Potential PWR impacts (e.g., modeling, AM strategies, etc.)
RB-3b	Photos/ videos of damaged walls and structures (1F3)	• Determine mode of explosion in 1F3. • Gain insight about highly energetic explosions in 1F3 compared to 1F1.	Understanding what happened; assist D&D efforts. Potential BWR improvements; Impacts BWR AM strategies and code models (venting and interconnection between units); Potential PWR impacts (e.g., modeling, AM strategies, etc.).
RB-3c	Photos/ videos of damaged walls and structures (1F4)	• Determine mode of explosion in 1F4.	Understanding what happened; assist D&D efforts. Potential BWR improvements; Impacts BWR AM strategies and code models (venting and interconnection between units); Potential PWR impacts (e.g., modeling, AM strategies, etc.).
RB-4	Photos/ videos of damaged walls and components and radionuclide surveys (1F2)	• Cause of depressurization. • Cause of H_2 generation.	Understanding what happened; assist D&D efforts. Impacts BWR AM strategies (equipment utilization and venting); Improved BWR code simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.).

Table B-1. Information requests for the reactor building[a]

Item[b]	What/How Obtained	Why	Benefit /Use
RB-5	Radionuclide surveys (1F1, 1F2, and 1F3)	• 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.
RB-6	Radionuclide surveys and sampling of ventilation ducts (1F4)	• Isotope concentration could be used for determining source of H_2 production for CCI.	Understanding what happened. Potential BWR plant improvements (hardened vent use, AM strategies, and multi-unit effects, etc.). Potential PWR impacts (e.g., modeling, AM strategies, multi-unit effects).
RB-7	Isotopic evaluations of obtained concrete samples (1F2)	• Code assessments. • Possible model improvements for building retention assumptions.	Understanding what happened; assist D&D efforts. Improved BWR modeling and emergency planning; cross check of RN surveys. Potential PWR impacts (e.g., modeling, AM strategies, etc.).
RB-8	Photos/ videos and inspection of seismic susceptible or radiation degraded components and structures (e.g., bellows, penetrations, welds, structures, supports, etc. in 1F1, 1F2, 1F3, and 1F4)	• To confirm with data that there were no seismic-induced failures • To determine with data if there are any radiation-degraded components and concrete structures; • To develop lessons learned regarding their performance under high radiation conditions	Understanding what happened; assist D&D efforts. Improved plant robustness; observed differences between 1F1 and 1F3. Potential PWR impacts (e.g., similar penetrations, structures, and components). Additional seismic data for large magnitude earthquakes that is specific to nuclear related components and systems is of interest for operating and new reactors. It may be possible to use results to discern differences between challenges from H_2 explosions and seismic events.
RB-9	a) DW Concrete Shield Plug Radionuclide surveys and gaps between sections (1F1, 1F2, and 1F3 - after debris removed)	• To understand leakage amounts and locations. • Gaps affect fission product transport and deposition.	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 LWR as well as new LWR and non-LWR designs.
	b) Photos/ videos and dose surveys around mechanical seals and hatches and electrical penetration seals (as a means to classify if joints in compression or tension)	• Potential leakage paths for RN and hydrogen release.[d] • 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 LWR as well as new LWR and non-LWR designs.

Table B-1. Information requests for the reactor building[a]

Item[b]	What/How Obtained	Why	Benefit /Use
RB-10	Photos/ videos and dose surveys of 1F1 (vacuum breaker), 1F1, 1F2, and 1F3 PCV leakage points (bellows, penetrations)	• Potential leakage paths for RN and hydrogen release. • To develop lessons learned regarding penetration 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.
RB-11	Photos/ videos and dose information on 1F1, 1F2, 1F3, and 1F4 containment hardpipe venting pathway, SGTS and associated reactor building ventilation system	• To assess performance of SGTS under high temperature and radiation conditions.[e] • 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.
RB-12	Photos/ videos at appropriate locations near identified leakage points in 1F1, 1F2, and 1F3.	• To discern reason for leakage from the reactor building into the turbine building. • To develop lessons learned regarding their performance under high radiation/high temperature conditions	Improved BWR AM strategies (Plant improvements); potential PWR impacts, depending on identified leakage path. Assist D&D efforts.
RB-13	Photos/ videos of 1F1, 1F2, and 1F3 main steam lines at locations outside the PCV	• To determine PCV failure mode. • To develop lessons learned regarding their performance under high radiation/high temperature conditions	BWR AM strategies (plant mods, etc.) and better simulations for training. Assist D&D efforts.
RB-14	Perform chemical analysis of high radiation deposits or particles found inside the reactor building (1F1, 1F2, and 1F3); e.g., the white deposits from the HPCI room using ICP, FE-SEM, XRD, etc.	• Presence of Ca/Al/Si/Mg would indicate MCCI.	Assist D&D efforts for determining debris location.
RB-15	Examinations (water level and additional dose surveys) of 1F1 RCW surge tank and evaluations of RCW water samples	• 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 • Presence of Ca/Al/Si/Mg would indicate MCCI.	Determine the role of the RCW during 1F1 accident. Assist D&D efforts for determining debris location.
TRB-1	Stagnant water analysis (1F1, 1F2, and 1F3)	• Increased understanding of accident progression	
TRB-2	Photos/video, RN analysis of 1F2 RCIC room upper wall surface	• FP leak path clarification	
TRB-3	U2 vent line rupture disc (non) rupture investigation, photos/videos	• Deepening understanding of accident progression • Success/failure of venting	
TRB-4	U4 SGTS filters RN analysis	• Deepening understanding of accident progression • Composition of FPs in vent	
TRB-5	U2 S/C liquid leakage point investigation (RCIC room, etc.)	• PCV leakage point clarification	

Table B-1. Information requests for the reactor building[a]

Item[b]	What/How Obtained	Why	Benefit /Use
TRB-6	U1 - U4 AC piping contamination/rust related investigation, dose survey, photos/videos	• Deepening understanding of accident progression • Effects of venting, FP behavior	
TRB-7	Dose survey and FP analysis of U1/2 exhaust stack base high dose rate area	• Deepening understanding of accident progression • Composition of FP in vent	
TRB-8	U2 torus room inundation marks	• Deepening understanding of accident progression	
TRB-9	Information related to instrumentation soundness -exterior photos to confirm mechanical soundness -electrical inspections results (including records after the disaster)	• Deepening understanding of accident progression	
TRB-10	Information related to SW piping soundness / exterior photos to confirm mechanical soundness	• Deepening understanding of accident progression	
TRB-11	Residual gas analysis	• Deepening understanding of accident progression	

a. See "ACRONYMS AND ABBREVIATIONS" for abbreviations.

b. Items with designators starting with 'T' were developed by TEPCO.

c. 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.

d. 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.

e. 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[a]

Item	What/How Obtained	Why	Benefit /Use
PC-1	Photos/ videos[b] of drywell head, head seals, and sealing surfaces (1F1, 1F2, and 1F3). Procedures used to tension and torque the bolts used to close the drywell head bolts.	• Determine how head lifted. • Determine peak temperatures. • Look for indicators of degradation due to high radiation and high temperature hydrogen, including hydrogen-induced embrittlement.	AM Strategies; What happened with respect to the leak path; better simulations for training. Assist D&D efforts. Available information indicates that no changes in tensioning procedures are needed. Additional information regarding sealing surface and elastomer condition could provide insights of what occurred and inform consideration of potential failure modes.
PC-2	Photos/videos and radionuclide surveys/ sampling of IC (1F1)	• Evaluate for seismic damage. • Evaluate final valve position. • Gain insights about hydrogen transport.	AM Strategies (plant robustness, use of equipment in limited number of plants with ICs and new passive plants); better simulations for training. Assist D&D efforts.
PC-3	a) Photos/ videos of relocated debris and crust, debris and crust extraction, hot cell exams, and possible subsequent testing (1F1 - 1F3)	• Code assessments • Possible model updates for mass, height, composition, morphology (e.g., coolability), topography of debris, spreading, splashing, and salt effects.	BWR AM Strategies (plant robustness, use of equipment, inform cavity flooding strategies) and better simulations for training. Potential PWR impacts (e.g., modeling.).[c] Assist D&D efforts.
	b) PCV liner examinations of debris (photos/videos and metallurgical exams; 1F1-1F3)	• Code assessments. • Possible model improvements for predicting liner failure and MCCI.	AM Strategies (improved plant robustness); better simulations for training. Assist D&D efforts. Information could inform life beyond 80 as well as new LWR and non-LWR design efforts.
	c) Photos/ video, RN surveys, and sampling of debris and water samples near the pedestal wall and floor (1F1-1F3)	• For benchmarking code predictions of vessel failure location and area, mass, morphology (e.g., coolability), and composition of ex-vessel debris, and MCCI.	BWR AM Strategies, better simulations, etc. Potential PWR impacts (e.g., modeling, AM strategies, etc.). Assist D&D efforts.
	d) Concrete erosion profile; photos/videos and sample removal and examination (1F1-1F3)	• For benchmarking code predictions of MCCI.	BWR AM Strategies (plant mods, etc.) and better simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.). Assist D&D efforts.
	e) Photos / videos of RPV lower head and of structures and penetrations beneath the vessel to determine damage and corium hang-up (1F1-1F3)	• Code assessments. • Possible model improvements.	BWR AM Strategies (plant modifications, etc.) and better simulations for training (improved models for predicting containment pressure-temperature response); Potential PWR impacts (e.g., modeling, AM strategies, etc.). Assist D&D efforts.
PC-4	Photos/videos of 1F1, 1F2, and 1F3 recirculation lines and pumps	• To determine PCV failure mode and relocation path. • To develop lessons learned regarding performance under high radiation/high temperature conditions	AM Strategies (plant mods, etc.) and better simulations for training.
PC-5	Photos/videos of 1F1, 1F2, and 1F3 main steam lines and ADS lines to end of SRV tailpipes, including instrument lines	• To determine RPV failure mode.	BWR AM Strategies (plant modifications, etc.) and better simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.).
PC-6	Visual inspections of 1F1, 1F2, and 1F3 SRVs and MSLs including standpipes (interior valve mechanisms)	• To determine if there was any failure of SRVs and associated piping.	BWR AM Strategies (maintenance practices, etc.), SRV functioning in test facility data, and better simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.).

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

Item	What/How Obtained	Why	Benefit /Use
PC-7	Ex-vessel inspections of cables and operability assessments of 1F1, 1F2, and 1F3 in-vessel sensors and sensor support structures[d]	• Data qualification for code assessment. • Identification of vessel depressurization paths. • To develop lessons learned regarding performance under high radiation/high temperature conditions • To evaluate possible combustible gas sources from cable decomposition	Equipment qualification life (1F1 at 40 years; underwater cabling). Improved AM strategies and better simulations for training for operating, new, and advanced reactor designs
PC-8	Examinations and operability assessments of 1F1, 1F2, and 1F3 ex-vessel sensors and sensor support structures[e]	• Data qualification for code assessment. • Identification of vessel depressurization paths. • Understanding why the RPV A and B pressure signals decalibrated. • To develop lessons learned regarding their performance under high radiation/high temperature conditions	BWR and possible PWR equipment qualification life; better qualifications for training. Insights regarding survivability support revised severe accident strategies. Images of penetration seals associated with PCV pressure sensors are of interest because of potential reduction in PM and surveillance.
PC-9	Photos/videos of 1F1, 1F2, and 1F3 PCV (SC and DW) coatings	• Assess impact for coating survivability. • To develop lessons learned regarding their performance under high radiation/high temperature conditions • To gain insights regarding combustible gas sources	BWR and possible PWR maintenance upgrades. Improved AM strategies and better simulations for training for operating, new, and advanced reactor design efforts
PC-10	1F1, 1F2, and 1F3 RN surveys in PCV	• Dose code assessments. • Possible model improvements.	BWR and possible PWR AM strategies/better simulations (plate out). Assist D&D efforts
PC-11	Photos/videos of 1F1, 1F2, and 1F3 primary system recirculation pump seal and any potential discharge to containment	• To assess performance under high temperature/ high pressure conditions.[f] • 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.[f]
PC-12	Photos/videos of 1F1, 1F2, and 1F3 TIP tubes and SRM/IRM tubes outside the RPV	• To determine if failure of TIP tubes and SRM/IRM tubes outside the RPV led to depressurization. • To develop lessons learned regarding performance under high radiation / high temperature conditions	BWR AM Strategies and maintenance practices, SRV performance insights, and better simulations for training. Potential PWR impacts (e.g., modeling, AM strategies, etc.). Assist D&D efforts.
PC-13	Photos/videos of 1F1, 1F2, and 1F3 insulation around piping and the RPV	• To determine potential for adverse effects on long-term cooling due to insulation debris. • To develop lessons learned regarding performance under high radiation / high temperature conditions	Improved BWR and PWR AM strategies (plant improvements).
PC-14	Samples of conduit cabling, and paint from 1F1, 1F2, and 1F3 for RN surveys	• Dose code assessments. • Possible model improvements.	BWR and possible PWR AM strategies/Better simulations (plate out).
PC-15	Samples of water from 1F1, 1F2, and 1F3 for RN surveys	• Dose code assessments. • Possible model improvements.	BWR and possible PWR AM strategies/Better simulations. Assist D&D efforts.
PC-16	Photos/videos of melted, galvanized, or oxidized 1F1, 1F2, and 1F3 structures	• To provide indications of peak temperatures (for possible model improvements).	Improved AM strategies (Plant improvements).

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

Item	What/How Obtained	Why	Benefit /Use
PC-17	Chemical and isotopic analysis of the upper layer of sediment on drywell floor at the X-100B penetration location in 1F1.The upper surface of the sediment is ~ 30 cm above drywell floor. Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes.	• Presence of concrete oxides would indicate MCCI • Possible model improvements • Testing has shown that the ability to cut core debris is strongly impacted by amount of concrete oxides present • Presence of short-lived fission product isotopes could indicate low-level recriticality. • Given the low level of decay heat present in 1F1, any low-level criticality could impact plant heat balance calculations.	Assist D&D efforts for recriticality prevention, debris stabilization, locating fuel-containing materials, and debris removal and storage. Improved accident management strategies.
PC-18	Evaluate nature of material below the sediment at the 1F1 X-100B penetration location to determine if fuel debris is present. Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes.	• Presence of concrete oxides or core material debris would indicate MCCI • Possible model improvements • Testing shows that the ability to cut core debris is strongly impacted by amount of concrete oxides present • Presence of short-lived fission product isotopes could indicate low-level recriticality. • Given the low level of decay heat present in 1F1, any low-level criticality could impact plant heat balance calculations.	Assist D&D efforts for recriticality prevention, debris stabilization, locating fuel-containing materials, and debris removal and storage. Improved accident management strategies.
PC-19	Chemical analysis (XRF) of black material discovered on CRD exchange rail in 1F2 at X-6 penetration location	• Identification of material could provide an indicator of peak structure temperatures and potential for structure failure. • Possible model improvements.	Assist D&D efforts for determining debris location. Modeling improvements for ex-vessel holdup have been implemented in MAAP and informed accident management strategies and risk assessment metrics.
PC-20	Chemical analysis of black material on 'existing structure' in 1F1 images at location 'D3'	• Presence of Si or core material debris would indicate MCCI • Possible model improvements. • Testing shows that the ability to cut core debris is strongly impacted by amount of concrete oxides present • Presence of short-lived fission product isotopes could indicate low-level recriticality. • Given the low level of decay heat present in 1F1, any low-level criticality could impact plant heat balance calculations.	Assist D&D efforts for recriticality prevention, debris stabilization, locating fuel-containing materials, and debris removal and storage. Improved accident management strategies.
PC-21	Images from examinations in 1F3 X-53 penetration	• Possible model improvements • To estimate possible combustible gas sources from cable decomposition	Assist D&D efforts for determining debris location Improved AM strategies and better simulations for training

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

Item	What/How Obtained	Why	Benefit /Use
PC-22	Chemical analysis of debris from locations at different axial and radial positions (bores, if possible). Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes. (1F1-1F3)	• Presence of concrete oxides would indicate MCCI • Gain insights about material relocations • Material properties important to tooling design (e.g., density and hardness) are known to be a function of material composition (e.g., the ability to cut debris is impacted by amount of concrete oxides present). • Potential concentrations of fuel. • Presence of short-lived fission product isotopes could indicate low-level recriticality. • Given the low level of decay heat present in 1F1, any low-level criticality could impact plant heat balance calculations • Possible model improvements.	Assist D&D efforts for recriticality prevention, debris stabilization, locating fuel-containing materials, and debris removal and storage. Potential modeling improvements for debris coolability during MCCI and inform accident management strategies and risk assessment metrics.[c]

a. See “ACRONYMS AND ABBREVIATIONS” for abbreviations.

b. 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.

c. 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.

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

e. 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.

f. 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.

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

Item[b]	What/How Obtained	Why	Benefit /Use
RPV-1	a) 1F1, 1F2, and 1F3 dryer integrity and location evaluations (photos/videos[c] with displacement measurements, peak temperature evaluations). If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.	• Code assessments. • Possible model improvements.	Improved AM strategies; Improved simulations for training. Assist D&D efforts.
	b) Photos/videos, probe inspections of 1F1, 1F2, and 1F3 MSLs; interior examinations of MSLs at external locations. If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.	• Code assessments. • Possible model improvements.	Improved AM strategies; Improved simulations for training. Assist D&D efforts.
	c) Photos/videos of upper internals and upper channel guides. If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.	• Code assessments. • Possible model improvements (for predicting peak temperatures, displacement, melting).	Improved AM strategies; Possible plant modifications; Improved simulations for training. Assist D&D efforts.
RPV-2	a) 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.	• 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.
	b) 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.	• Code assessments. • Possible model improvements.	Improved AM strategies, Improved simulations for training. Assist D&D efforts.

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

Item[b]	What/How Obtained	Why	Benefit /Use
RPV-4	a) 1F1, 1F2, and 1F3 shroud inspection (between shroud and RPV wall); Photos/videos of interest. If significant distortion observed, then metallurgical exams of samples would be of interest for D&D.	• 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.
	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.	• 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.
	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.	• 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.
	d) Photos/videos of 1F1, 1F2, and 1F3 core plate and associated structures.	• 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.
RPV-5	a) Remote mapping of 1F1, 1F2, and 1F3 core through shroud wall from annular gap region (muon tomography and other methods, as needed).	• Code assessments. • Possible model improvements.	Improved AM strategies; Possible plant modifications; Improved simulations for training. Assist D&D efforts.
	b) Mapping of end state of core and structural material (visual, sampling, hot cell exams, etc.).	• Code assessments. • Possible model improvements for predicting debris composition, mass, and morphology (e.g., coolability, topography of debris, spreading, splashing, and salt effects.	Improved BWR and potential PWR AM strategies; plant modifications, and improved simulations for training. Assist D&D efforts.
TRPV-1	Photos / videos, sampling of in-vessel debris	• Obtaining knowledge on the mass, morphology, composition distribution, spread, etc. of fuel debris and deepening understanding of accident progression	

a. See "ACRONYMS AND ABBREVIATIONS" for abbreviations.
b. Items with designators starting with 'T' were developed by TEPCO.
c. With the exception of general area views, photos and videos should be obtained with reference length scales at appropriate locations. In particular, it is required for photos and videos to be most effective for RPV-1(b), RPV- 2(a), RPV-3 and RPV-4(d)

B.2. Additional Details for Information Requests

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

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

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

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

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

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

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

• **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, RB):**
PC-3a: Photos/videos of debris and crust, debris and crust extraction, possible hot cell exams, and possible subsequent testing (1F1, 1F2, and/or 1F3) High-resolution images (photos/videos) within the PCV drywell of debris and crust 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 and Cs/U, Sr/U, and Te/U ratios to provide evidence of and insights about the extent of MCCI.
• **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
Structural characteristics of the material are important for supporting tooling design for removal; chemical analysis important for criticality evaluations. These same data are important for improving reactor safety analysis models and accident management.
• **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
Benchmark and reduce uncertainty in models for predicting MCCI phenomena. MCCI phenomena are important for assessing combustible gas generation during late phase accident progression, as well as the extent of attack on containment structures. It is important to reduce uncertainty in this phenomenon because it affects strategies for venting and water addition. Improved knowledge will be used to enhance accident management strategies.
• **Methods/Tools Needed to Collect Information or Data:**
• Irradiation resistant high-resolution imaging system • Hot cell elemental analysis system, and/or in-situ elemental analysis using Laser Induced Breakdown Spectroscopy (LIBS) and/or X-ray Florescence • Ultimately, D&D cutting and removal tools able to extract materials
• **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later (Sample removal possible within next 2 years).
• **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
Obtaining /using this information may require additional material property and coolability testing (Young's modulus, linear expansion, ultimate strength, hardness, tensile strength, etc.) for cutting tool development and for model development. Evaluation of this information may require composition information for concrete (to distinguish between sand and concrete).

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

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

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

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

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

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

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

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

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

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

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

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

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

• **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, RB):**
PC-17: Chemical analysis of upper layer of sediment on drywell floor at the X-100B penetration location in 1F1. The upper surface of the sediment is ~ 30 cm above drywell floor. Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes. **PC-18:** Evaluate nature of material below the sediment at the 1F1 X-100B penetration location to determine if fuel debris is present.[a] Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes. **PC-19:** Chemical analysis (XRF) of black material discovered on CRD exchange rail in 1F2 at X-6 penetration location **PC-20:** Chemical analysis of black material on 'existing vertical wall structure' in 1F1 picture outside pedestal doorway **PC-22**: Chemical analysis of debris from locations at different axial and radial positions (bores, if possible). Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes. Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes. (1F1-1F3). These five information requests focus on 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, control material, structural, and concrete components and 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)].
• **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
Required for D&D; desired for improving reactor safety analysis models and accident management.
• **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
As emphasized in Table B-10, debris characterization parameters, such as morphology, particle size distribution, porosity, and permeability, are important for removal, drying and storage activities and for benchmarking and reducing uncertainty in models for predicting MCCI phenomena. It is also important to obtain information to characterize the chemical form of fuel-bearing materials because of the potential for sludge formation (UO_2 oxidizes to form hydrates, such as $UO_{3.2}H_2O$, $UO_{4.2}H_2O$, and $UO_{4.4}H_2O$). This information is needed to determine if sludge can be separated from bulk particulate material created during debris retrieval. 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. Additional PC-19 analysis can be used to assess extent of in-vessel cladding oxidation. PC-18 evaluations can be used to determine if core debris is present at X-100B location, providing insights on 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.
• **Methods/Tools Needed to Collect Information or Data:**
• Hot cell elemental analysis system and/or in-situ elemental analysis using LIBS and/or 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 detect 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.
• **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later.
• **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
Evaluation of this information requires composition information for concrete (to distinguish between sand and concrete) and would benefit from chemical analysis of seashore sand located at the site.

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

Technical Supplement for PC-18 Evaluation

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

There are a variety of potential sources for this sediment material that may include decomposed/flaked paint, thermal insulation, cable insulation, sand/sediment from low quality seawater injection, aerosol from core concrete interaction, among others. If the material is sand entrained with the seawater that was injected or concrete aerosol from core-concrete interaction, then it may be possible to determine the origin based on the relative proportions of dominant concrete oxides such as SiO_2, CaO, Al_2O_3, and MgO in the sediment. For sand from seawater injection, analysis of a sample of beach sand obtained at the site would provide definitive data for direct comparison with elemental analysis data obtained from a sample of the sediment. In lieu of this information, the composition of sand from 12 different beaches along the east and west coasts of Japan have been reported in the literature.[110] 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.[111] 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.[111]

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

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

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

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

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

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

• **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, RB):**
RPV-4: a) 1F1, 1F2, and 1F3 shroud inspection (between shroud and RPV wall); Photos/videos of interest. If significant distortion observed, then metallurgical exams of samples would be of interest for D&D. b) 1F1, 1F2, and 1F3 shroud head integrity and location (photos/videos). If significant distortion observed, then metallurgical exams of samples would be of interest for D&D. c) Photos/videos of 1F1, 1F2, and 1F3 shroud inspection (from core region). If significant distortion observed, then metallurgical exams of samples would be of interest for D&D. d) Photos/videos of 1F1, 1F2, and 1F3 core plate and associated structures **RPV-5** a) Remote mapping of 1F1, 1F2, and 1F3 core through shroud wall from annular gap region (muon tomography and other methods, if needed) b) Mapping of end state of core and structural material (visual, sampling, hot cell exams, etc.) A sufficient number of samples of core material should be examined to determine the approximate proportions of Uranium/Zirconium/Stainless Steel/Boron from any upper core remnants, core plate accumulations, drive tube accumulations above bottom of vessel, and any accumulations on the lower vessel head region. Results can be used to determine roughly the tendency for spatial separation of lower melting and metallic rich core debris materials from the more ceramic remnants and by implication, the temporal separation of relocation events. The same information is needed for the drive tubes below the vessel head and the corium samples retrieved from the cavity region. This information is needed to validate code assumptions of phase interactions during core degradation.
• **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
BWR reactor safety analysis models have very significant uncertainties related to in-core damage progression modeling. These inspections can provide information that can help resolve the generally agreed upon largest uncertainties in BWR severe accident modeling. These uncertainties influence the understanding of containment response during a severe accident and are thus relevant to informing accident management.
• **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
Resolve large uncertainties for in-core damage progression at BWR reactor-scale. These inspections are relevant to addressing areas where testing has been unable to reproduce key areas of BWR in-core debris relocation behavior at reactor-scale. The pathways by which debris relocate within the core-region influence the potential for rapid pressurization of containment to occur (e.g., due to rapid steam or hydrogen generation). The acquisition of knowledge to reduce uncertainties in this area can refine severe accident models, enhancing the effectiveness of accident management training.
• **Methods/Tools Needed to Collect Information or Data:**
• Irradiation resistant high-resolution imaging system • Hot cell elemental analysis system, and/or in-situ elemental analysis using Laser Induced Breakdown Spectroscopy (LIBS) and/or X-ray Florescence • Ultimately, D&D cutting and removal tools able to extract materials
• **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later (Sample removal possible within next 2 years).
• **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
Obtaining /using this information may require additional material property and coolability testing (e.g., oxidation state of in-core debris). Refined understanding of mechanical properties of retrieved in-core debris, however, are of significant benefit to the design and development of cutting tools. Refined understanding of in-core damage progression will require effort to refine analytical models for this phase of a severe accident.

APPENDIX C. Selected FY2023 Presentations

This appendix contains presentations from participants wishing to have them published in this report. Presentations are organized according to topics. Appendix C.1 contains U.S. introductory presentations; Appendix C.2, contains presentations with new information from Japan; U.S. topic area presentations are found in Appendix C.3; and recent U.S. systems analysis code development and application activities are found in Appendix C.4. Section 2 highlights key points discussed during these and other presentations during the meeting. As indicated in the agenda (see Appendix A.1), the schedule was arranged to minimize the impact of time differences on virtual participants in Japan.

C.1. Introductory Presentations

C.1.1. U.S. DOE Forensics Effort

C.1.1.1. Meeting Overview

U.S. DEPARTMENT OF ENERGY | Nuclear Energy

US Efforts to Support Examinations at Fukushima Daiichi –Meeting Overview

Joy Rempe
Technical Lead, Rempe and Associates, LLC

November 17 - 18, 2022

FY23 Agenda

U.S. DEPARTMENT OF ENERGY Nuclear Energy

FY23 Meeting Topics*

- JAEA Updates and Related Research –Ikeuchi, Nagae, and Sato
- NDF 2022 Strategic Plan - Nakagawa, Hokugo, and Ito
- Introductory Remarks by DOE and NRC – Peko and Esmaili
- TEPCO Recent Findings and Future Plans – Mizokami, Cibula, and Owada
- NRA Updates on Key Topics – M. Yasui, Iwanaga, and Konishi
- Q&As on TEPCO and NRC presentations – All
- Topic Area Discussions:
  - Area 1 - Components/System Performance – Robb and Gabor
  - Area 2 - Radionuclide Surveys/Sampling – Albright and Luxat
  - Area 3 - Core Debris Location Evaluations – Farmer and Plys
  - Area 4 - Combustible Gas Effects - Luangdilok
  - Area 5 - Operations and Maintenance –Ellison, Bunt, and Williamson
- Related EPRI Activities – Nudi
- Related NRC-sponsored MELCOR Development and SA Activities – Luxat
- Update on Mid- and Long-term Plan for Examinations – Cibula, Owada, and Mizokami
- Information Request Discussion and Next Steps - All (led by Rempe)

* See agenda for detailed presentation schedule. Link to FY23 Agenda, Viewgraphs, FY22 Report & Draft FY23 Report (when available): https://1drv.ms/u/s!ApliColj18LBgpBLm07qcwqkzNXPUg?e=RRrldv

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U.S. DEPARTMENT OF ENERGY
Nuclear Energy

Possible Ideas for Information Request Discussion

- All remaining US information requests have been incorporated into TEPCO Mid- and Long-Term Examination Plans.
  - Some have been assigned to lower priority categories, but Japan is reviewing categorization and updates as new information (and resources) are obtained.
- Possible changes to current US information request discussion:
  - Rather than continued US review and updates of examination request list, should our discussion focus on identifying any requests for which more detailed writeups (see Appendix B.2) would benefit Japan?
  - In light of the changing nature of some information requests, should the report stop identifying which requests have been completed?

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C.1.1.2. U.S. DOE Forensics Effort 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 17, 2022

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.

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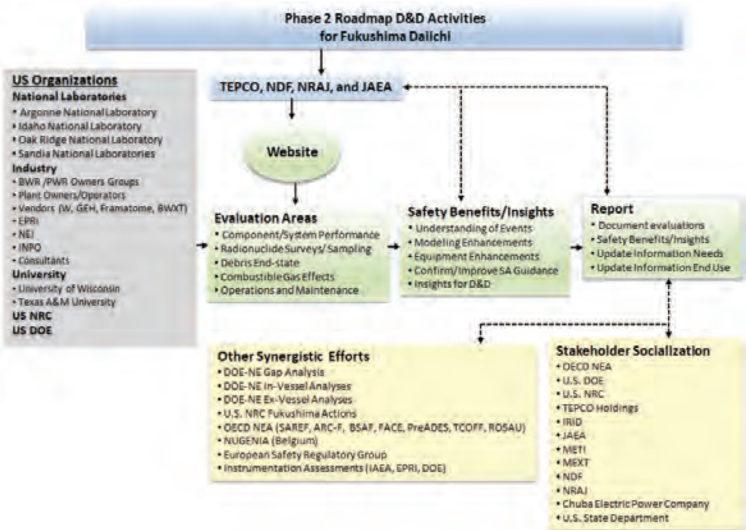
U.S. Efforts Coordinated with Phase 2 Roadmap D&D Activities and Other Programs



FY2022 report publicly available (https://www.osti.gov/biblio/1856658)

3

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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)**
  - **Post-Fukushima actions (FLEX, updated guidance to rely on existing instrumentation with water addition and early venting strategies) as well as severe accident safety testing and analysis programs allowed continued operation of US plants.**
  - **It behooves us to be aware of on-going international efforts to deploy new SSCs in existing and advanced designs to address issues observed at Daiichi (e.g., SSCs to monitor hydrogen production, water level, IC system performance).**
- **Forensics effort continues to evaluate new examination information related to system performance and phenomena observed during events at Daiichi**
  - **Information needed to address knowledge gaps in severe accident understanding and improve operator guidance for severe accident management.**
  - **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 could reduce plant maintenance costs and personnel exposures.**
  - **Per FY22 recommendation, DOE and MEXT exploring options that could further technology development and deployment.**

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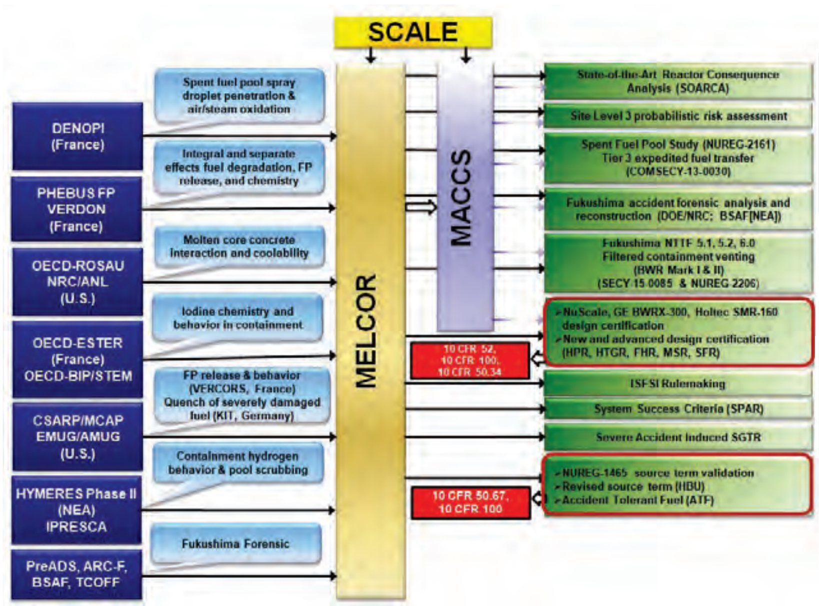
REMINDER

INFORMATION RELEASE

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

Severe Accident Code Development & Regulatory Applications

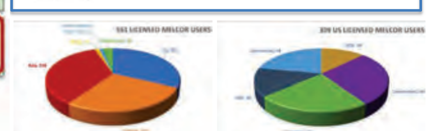


What Is It?
MELCOR is an engineering-level code that simulates the response of the reactor core, primary coolant system, containment, and surrounding buildings to a severe accident.

Who Uses It?
MELCOR is used by domestic universities and national laboratories, and international organizations in around 30 countries. It is distributed as part of NRC's Cooperative Severe Accident Research Program (CSARP).

How Is It Used?
MELCOR is used to support severe accident and source term activities at NRC, including the development of regulatory source terms for LWRs, analysis of success criteria for probabilistic risk assessment models, site risk studies, and forensic analysis of the Fukushima accident.

How Has It Been Assessed?
MELCOR has been validated against numerous international standard problems, benchmarks, separate effects (e.g., VERCORS) and integral experiments (e.g., Phebus FPT), and reactor accidents (e.g., TMI-2, Fukushima).

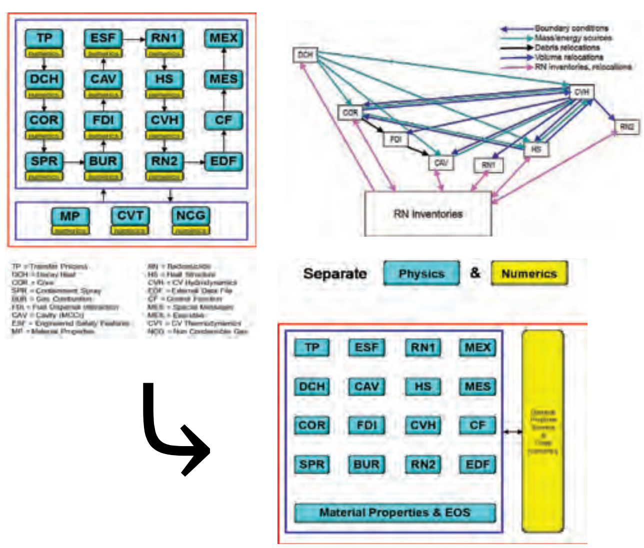


MELCOR User Groups & Technical Meetings

Cooperative Severe Accident Research Program (CSARP) – June/U.S.A
MELCOR Code Assessment Program (MCAP) – June/U.S.A
European MELCOR User Group (EMUG) Meeting – Spring/Europe
Asian MELCOR User Group (AMUG) Meeting – Fall/Asia

MELCOR Modernization (2024)

- Generalized numerical solution engine ✓
- Hydrodynamics ✓
- In-vessel damage progression
- Ex-vessel damage progression
- Fission product release and transport



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International Severe Accident Projects

- **1988-2010** — IRSN: Phébus-Fission Products Program
- **2005-2010** — IRSN: Phébus-International Source Term Program
- **2006-2019** — NEA: Behavior of Iodine Project (BIP)
- **2005-2016** — IRSN: Experimental Program for Iodine Chemistry Under Radiation (EPICUR)
- **2011-2019** — NEA: Source Term Evaluation and Mitigation (STEM) Project
- **2013–2018** — NEA: Benchmark Study of the Accident at Fukushima (BSAF) Project
- **2017–2021** — NEA: Preparatory Study on Analysis of Fuel Debris (PreADES) Project
- **2019–2021** — NEA: Analysis of Information from Reactor Buildings and Containment Vessels of Fukushima (ARC-F) Project
- **2019–2023** — EU: Management and Uncertainties of Severe Accidents (MUSA)
- **2020–2024 (NEA)** — Experiments on Source Term for delayed Releases (ESTER); Reduction of Severe Accident Uncertainties (ROSAU)
- **2022-2024 (NEA)** — NEA: Thermodynamic Characterization Of Fuel debris and Fission (TCOFF-2) 2022-2024
- **2023–2026 (NEA)** — FUKUSHIMA Accident Information Collection & Evaluation (FACE)

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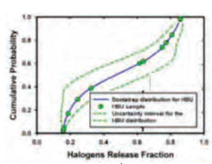
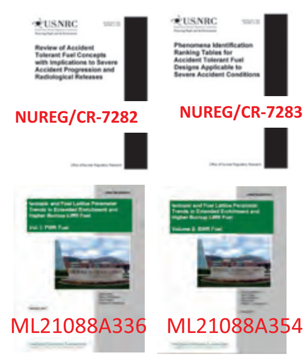
Source term for HBU/ATF

Experimental Basis ← PIRT

- Oxidation/Gas Generation
- Melt Progression
- Fission Product Release
- Fission Product Transport

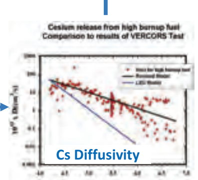


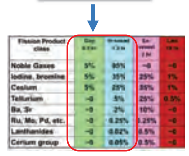
Panel of international severe accident experts Phenomena Identification and Ranking Tables (PIRT) that addressed significant phenomenological issues to improve MELCOR



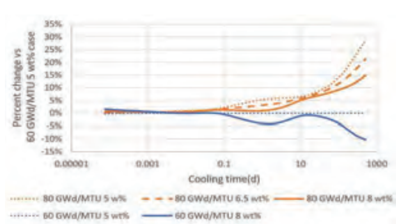
MELCOR → Accident Analysis (Scenario # 1, ………………, Scenario # n) → Synthesize timings & RF → DBA Source Term

HBU – Fall 2021 – Peer Review on-going
Cr-coated - Fall 2022



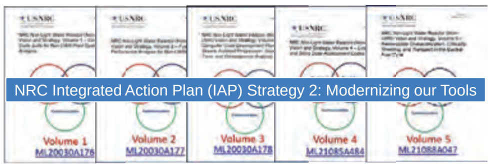


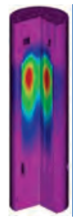
NUREG-1465 (RG 1.183)

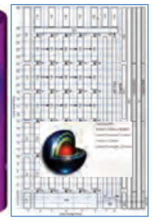


Advanced Nuclear Technology Research

Developed Efficient Strategy and Plan → Developed Reference Plant Models → Developed Expertise & Show Readiness

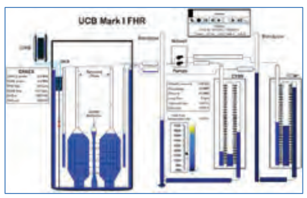


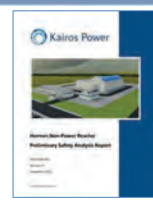




Public Workshops: SCALE/MELCOR Non-LWR Source Term Demonstration Project	
2021	***2022***
Heat Pipe Reactor *Gas Cooled Reactor* *Molten Salt Cooled Reactor*	*Molten Salt Fueled Reactor* *Sodium Fast Reactor*

Modify Reference Plant Model → Support Hermes Review





Knowledge Management Activities

- NUREG/KM-0001, *Three Mile Island Accident of 1979 Knowledge Management Digest* (www.nrc.gov/reading-rm/doc-collections/nuregs/knowledge/km0001)
  - **Supplement 2, *The Cleanup Experience: A Literature Review***, posted January 2021, consolidates experiences and lessons during long-term plant stabilization, data collection, cleanup, and defueling
  - **Supplement 3, *Cleanup Safety Evaluations 1979—1993***, posted July 2022, provides safety evaluations by the licensee and NRC that considered 17 safety issues of 64 cleanup activities, including data collection (+800 pages)
- NRC Public ADAMS now includes digitized records from the legacy microfiche collection (1980-1999) (https://adams.nrc.gov/wba/)

C.2. New Information from Japan

C.2.1. JAEA Updates

C.2.1.1. Recent Update on 1F Sample Analysis

Recent Update on

1F sample analysis

Hirotomo IKEUCHI

Japan Atomic Energy Agency
Collaborative Laboratories for Advanced Decommissioning Science

International Research Institute for Nuclear Decommissioning

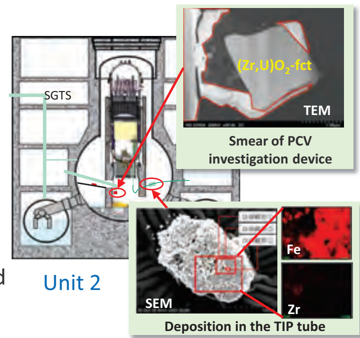
Contents 2

- Introduction
- Samples analyzed in 2021JFY
- Analysis Method
- Result and Discussion
  - Main features found in each sample
  - U-bearing particles: detailed observation
    probable formation mechanism
- Conclusions

Introduction

Background

- Several types of micro-particles containing U and radionuclides (U-bearing particles) have been found in the smear/deposition samples obtained through the PCV and the R/B investigations of Units 1 to 3.
- Characterizing those particles in detail (e.g., microstructure, local composition, crystal structure) would provide us information on the formation conditions under which the particles would have been formed during the accident progression.
  - ➢ Material origins involved in the formation of the particle
  - ➢ Temperature ranges the particle could have experienced
  - ➢ Redox conditions during formation of the particle



Typical particles found in the smear/deposition samples

Objective

- The formation mechanism of the U-bearing particles is discussed.

→ To be reflected in the estimation of fuel debris properties by estimation of the damage state over a wider area (e.g., the possibility of molten pool formation, concrete erosion, water vapor depleted atmosphere, etc.).

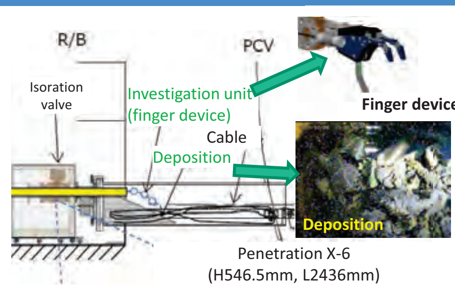
Edit on the figures quoted from: (TEPCO, 2021a; Nakayoshi, 2022)

Samples Analyzed in 2021JFY

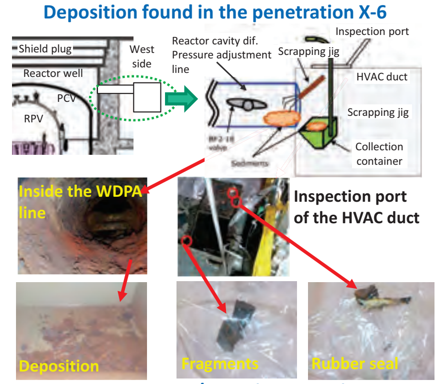
Samples from Unit 2

- Deposition in **the penetration X-6 of PCV**
  - ➢ **Smear** of the investigation unit (finger device) of **the penetration X-6**
    - – Gas phase containing radionuclides could have been passed through from inside PCV to R/B.
- Samples obtained through **the reactor well investigation**
  - ➢ **Deposition** in the west reactor cavity differential pressure adjustment **(WDPA) line**
  - ➢ **Fragments/Rubber seal** from the inspection port of **the HVAC duct**
    - – Inner wall of the WDPA line and surface of the HVAC duct was highly corroded probably due to the corrosive gas containing Cs during the accident.

⇒ Probably contain information at the time of accident progression of Unit 2



Deposition found in the penetration X-6



Deposition in/near the WDPA line

Edit on the figures quoted from: (TEPCO, 2020; 2021b)

Analysis Method

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Analytical items for each sample

Locations	Sample ID	Analytical Labs	Analytical items						
			Appearance	IP	FE-SEM WDX	FE-SEM EDX	TEM	ICP-MS	Gamma-ray
Penetration X-6	2PEN2101	JAEA	●	●	●	–	–	●	●
	2PEN2102	JAEA	●	●	●	–	–	●	●
	2PEN2103	NFD	●	–	–	●	●	–	–
Near reactor cavity: WDPA line	2WEL2101A	JAEA	●	●	●		–	●	●
	2WEL2101B	NFD	●	–	–	●	●	–	–
	2WEL2101C	JAEA	●	●	●	–	–	●	●
Fragments from duct	2WEL2102A	JAEA	●	●	●	–	–	●	●
	2WEL2102B	NFD	●		–	–	–	–	–
Rubber seal from duct	2WEL2103A	JAEA	●	●	●	–	–	●	●
	2WEL2103B	NFD	●	–	–	●	–	–	–

NFD : Nippon Nuclear Fuel Development Co.Ltd.

Analysis Method

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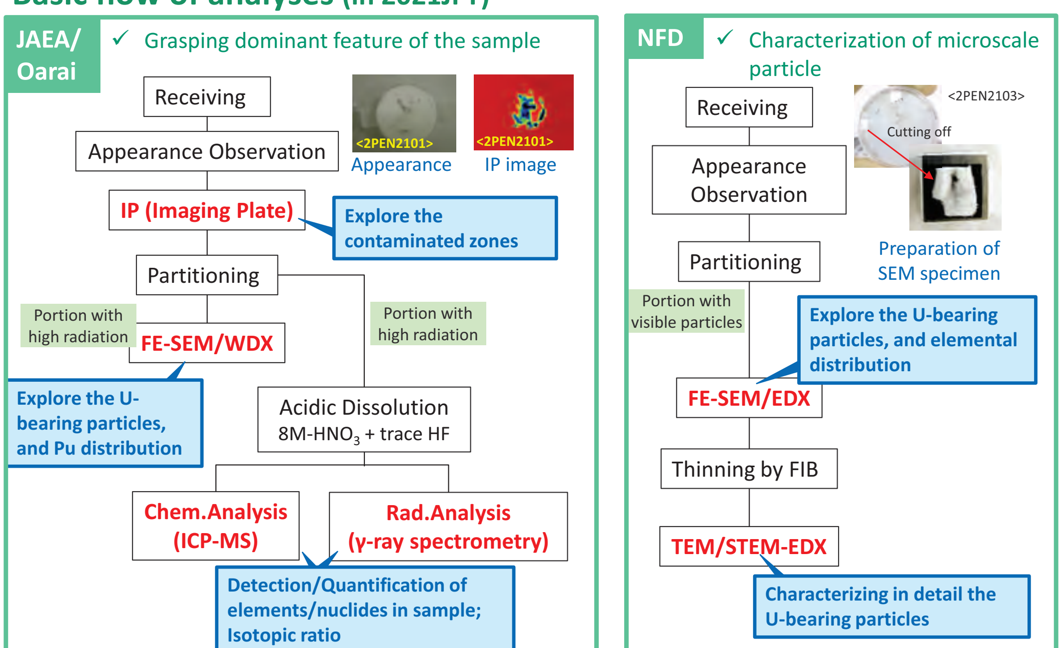
Basic flow of analyses (in 2021JFY)



Result and Discussion

7

Main features found in each sample : Penetration X-6

- **FE-SEM/WDX**
  - U-bearing particles detected:
    - > **U-Zr type** : *solidified melt of fuel components*
    - > **U-Pu type** : *fragments of fuel, or solidified melt.*
  - Particles containing specific elements: **Fe, Cr, Ni, Pb, Sb, Cu, Zn, Ti, …**

    → Probably includes degraded/altered materials from:

    Coating material (Ti, Zn), Grease (Mo),

    Cables and instruments (Cu, Ni, Mg, Sb, Ca, Pb),

    Thermal shield (Al),

    Structural materials (Fe, Cr, Ni, Mo, Zn),

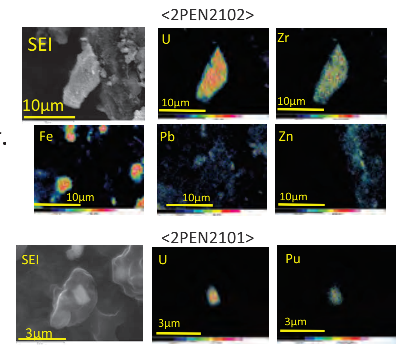
- **ICP-MS** (Isotopic ratio)

  U ($^{235}U/^{238}U$) : *fuel originated*

  Mo : Close to natural ratio

- **Gamma-ray spectrometry**

  Radioactive nuclides detected : **^{60}Co, ^{125}Sb, ^{134}Cs, ^{137}Cs, ^{154}Eu, ^{241}Am**



Elemental mapping (FE-SEM/WDX)

Isotopic ratio of typical elements (ICP-MS)

Samples	Mo		U
	($^{97}Mo/^{95}Mo$)	($^{98}Mo/^{95}Mo$)	($^{235}U/^{238}U$)
2PEN2101	0.73±0.08	1.7±0.2	0.0200±0.0003
2PEN2102	0.73±0.06	1.4±0.1	0.0193±0.0002
Natural Isotopic Ratio	0.60	1.52	0.0073

* Combined uncertainties are indicated as 1σ.

* Estimations are described in *Itallic*

Result and Discussion

8

Main features found in each sample : Deposition in the WDPA line

- **FE-SEM/WDX**
  - U-bearing particles detected:
    - > **U-Zr type** (Fe, Cr in adjacent part):

      → *solidified melt of fuel components and structural materials*
  - Particles mainly composed of : **Fe, Zn**

    → Probably includes *corrosion product of steel components, e.g., carbon steel*

- **ICP-MS** (Isotopic ratio)

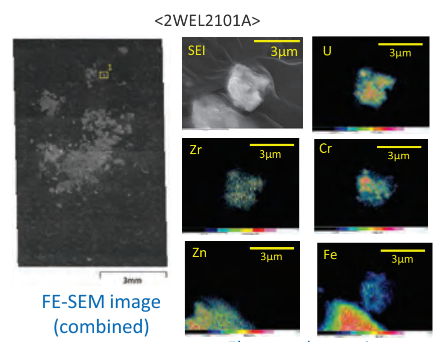
  U ($^{235}U/^{238}U$) : *fuel originated*

  Mo, Ag : different from natural ratio

  B : close to the natural ratio

- **Gamma-ray spectrometry**

  Radioactive nuclides detected : **^{60}Co, ^{125}Sb, ^{134}Cs, ^{137}Cs, ^{154}Eu**



FE-SEM image (combined)

Elemental mapping

Observation by FE-SEM/WDX

Isotopic ratio of typical elements (ICP-MS)

Samples	B	Mo		Ag	U
	($^{10}B/^{11}B$)	($^{97}Mo/^{95}Mo$)	($^{98}Mo/^{95}Mo$)	($^{107}Ag/^{109}Ag$)	($^{235}U/^{238}U$)
2WEL2101A	0.262±0.007	0.99±0.02	1.27±0.03	0.109±0.009	0.0209±0.0003
2WEL2101C	0.23±0.02	0.89±0.03	1.30±0.05	0.054±0.003	0.0206±0.0004
Natural Isotopic Ratio	0.248	0.60	1.52	1.076	0.0073

* Combined uncertainties are indicated as 1σ.

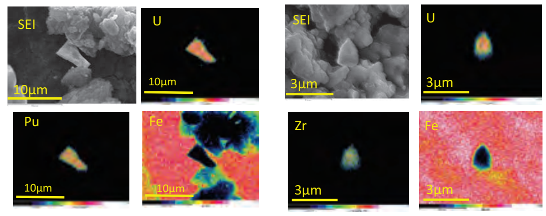
* Estimations are described in *Itallic*

Main features found in each sample : Fragments, and rubber seal from the HAVC duct

◆ FE-SEM/WDX

- U-bearing particles detected:
  - > **U-Zr type** : *solidified melt of fuel components*
  - > **U-Pu type** : *fragments of fuel, or solidified melt.*
- Large area of **Fe**

→ Probably consists of *corrosion product of steel components*



Elemental mapping (FE-SEM/WDX)

◆ ICP-MS (Isotopic ratio)

U : not detected

B, Mo, Ag : close to the natural ratio

Isotopic ratio of typical elements (ICP-MS)

Samples	B	Mo		Ag
	($^{10}B/^{11}B$)	($^{97}Mo/^{95}Mo$)	($^{98}Mo/^{95}Mo$)	($^{107}Ag/^{109}Ag$)
2WEL2102A	0.253±0.009	0.64±0.07	1.6±0.1	1.04±0.02
2WEL2103A	0.25±0.02	0.51±0.04	1.6±0.1	-
Natural Isotopic Ratio	0.248	0.60	1.52	1.076

* Combined uncertainties are indicated as 1σ.

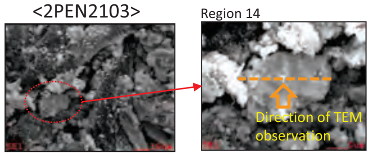
◆ Gamma-ray spectrometry

Radioactive nuclides detected : **^{134}Cs, ^{137}Cs**

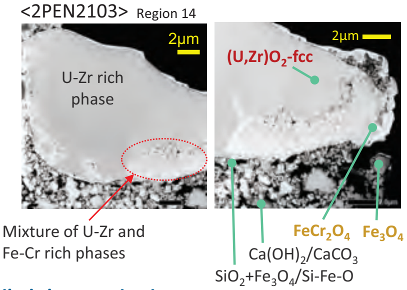
* Estimations are described in *Itallic*

U-bearing particles: detailed observation

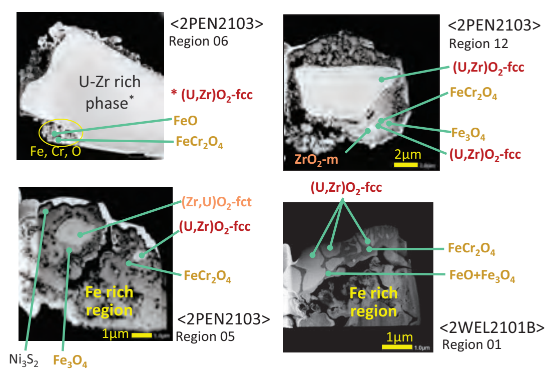
◆ **FE-SEM/EDX** followed by **TEM/STEM-EDX**



Detection of U-bearing particles by FE-SEM/EDX



Detailed characterization (microstructures, elemental composition of each phase, and crystal structures) by TEM analysis



Other U-bearing particles (observed by TEM)

Large particles (up to ~20μm in long) with:
- ✓ Main phase : **(U,Zr)O_2-fcc** (U:Zr ~ 6:4 to 5:5)
- ✓ Fe-Cr rich phases (mixed with or surrounding the main phase) : **FeO, Fe_3O_4, and $FeCr_2O_4$**

Minor particles (submicron to a few μm) with:
- ✓ U-Zr rich phase: **(U,Zr)O_2-fcc** (U:Zr ~ 6:4)
- ✓ Zr rich phase: **(Zr,U)-fct, ZrO_2-m**
- ✓ Fe-Cr rich phases : **Fe_3O_4, $FeCr_2O_4$**,

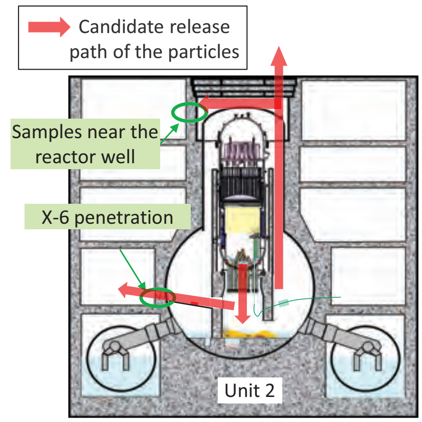
Result and Discussion

11

U-bearing particles: probable formation mechanism

Detailed characterization of the particles provides hints on the conditions in RPV (or PCV) during a certain period of accident progression (during which the D/W pressure had been kept high)

- Mixture of Fe-Cr rich phase (Fe_3O_4 or $FeCr_2O_4$) and U-Zr rich phase (mainly $(U,Zr)O_2$-fcc)
  - → *Solidified from molten mixture of U-Zr-Fe-Cr-O system*
  - → *Indicates mixing process of fuel components (U-Zr-O or UO_2-ZrO_2) and structural materials (Fe-Cr-O)*
- Homogeneous, dense, and single phase of $(U,Zr)O_2$-fcc with U:Zr~6:4 to 5:5.
  - → *Experienced approx. 2000 K or above* *(assuming UO_2-ZrO_2 pseudo-binary phase diagram)*
  - → *Indicates formation by rapid cooling, e.g. quenching by water, dispersion of droplet from molten pool*



Relation between location of the smear/deposition samples and candidate release path of the particles

* Estimations are described in *Itallic* Edit on the figure quoted from: (TEPCO, 2021a)

Conclusions

12

- In 2021JFY, **the smear/deposition samples obtained through investigations in Unit 2** were analyzed by using several techniques (IP, ICP-MS, γ-spectrometry, SEM, TEM) for detail characterization.
  - **The X-6 penetration** : As well as the cable materials (Cu, Sb, Pb, etc.), materials coming from inside PCV (Fe, Zn, U, Zr, etc.) were contained.
  - **Near the reactor well** : Corrosion product of Fe-based metals, on which the U-bearing particles are accumulated.
- Formation mechanism of **the U-bearing particles** found in the samples indicated specific process probably occurred during a certain period of the accident progression of Unit 2:
  - **Formation of U-Zr-Fe-Cr-O melt**
  - **Rapid cooling (dispersion of particles/quenching by water)**

Perspectives

- Analyses of the smear/sediment samples obtained from Unit 1 to 3 is now ongoing (in 2022JFY) to improve the accuracy of estimation on probable formation mechanism.
- The data obtained so far will be provided in the OECD/NEA FACE project for further discussion on the formation mechanisms of U-bearing particles.

References

Nakayoshi, A., Mitsugi, T., Sasaki, S., et al. (2022) : "Analysis of deposits inside the reactor at Fukushima Daiichi Nuclear Power Station in JFY 201-2018 –The subsidy programs "Project of Decommissioning and Contaminated Water Management in the FY2016 Supplementary Budget, (Development of Technologies for Grasping and Analyzing Properties of Fuel Debris)"" Japan Atomic Energy Agency (March, 2022). JAEA-Data/Code 2021-011. [in Japanese]

TEPCO Holdings (2020) : https://www.meti.go.jp/earthquake/nuclear/decommissioning/committee/osensuitaisakuteam/2020/11/3-3-3.pdf [in Japanese]

TEPCO Holdings (2021a) : https://www.tepco.co.jp/decommission/information/accident_unconfirmed/pdf/20210719.pdf [in Japanese]

TEPCO Holdings (2021b) : The committee of Accident Analysis of Fukushima Daiichi Nuclear Power Station 21th meeting (July 2021) Document 5-3. Available at: https://www.nsr.go.jp/data/000358693.pdf [in Japanese]

Thank you for your attention.

Experimental Research Related to Formation of Debris using Large-Scale Equipment

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

Fuel Debris Research and Analysis Division
Core Status Evaluation Group

Yuji NAGAE

Contents

1

1. Subject
   - Difference between actual and experimental feature of debris?

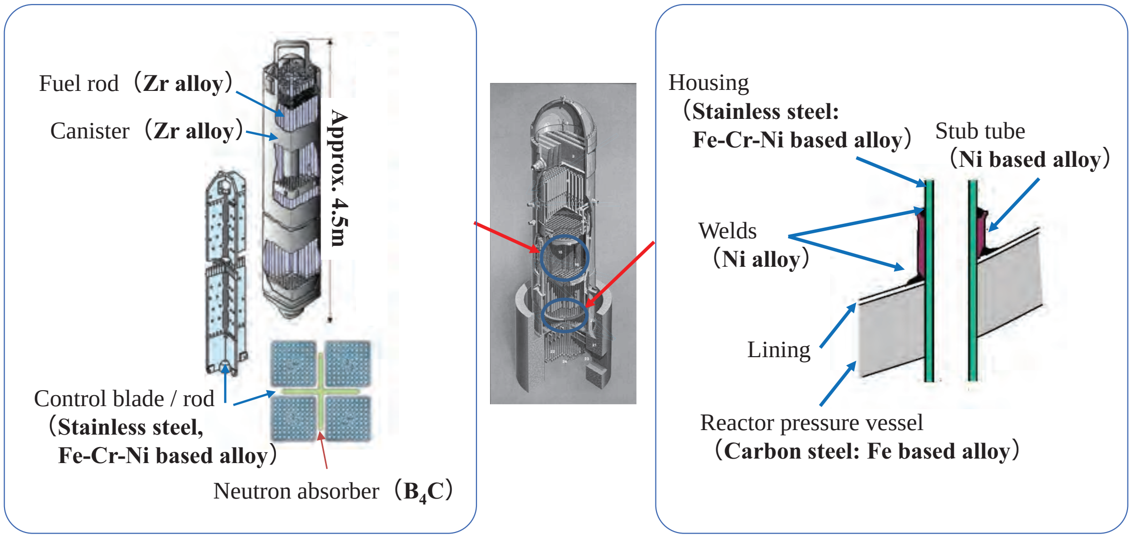
2. Current experimental output of debris formation
   - Large-scale tests for degradation behavior using our facility
     control blade degradation test
     reactor pressure vessel (RPV) failure test

Metallic materials used in BWR power plant

2

Core region (fuel assembly, control blade)

Lower part of RPV



Subject

3

<u>Subject:</u> Difference between actual and experimental feature of debris?

- To understand character or distribution of several types of debris by making a comparison with on-site observation, for instance appearance or sample analysis

<u>Experimental output on degradation behaviors:</u>

(1) Control blade degradation

Three types of debris' formation: Metallic, Oxidic and Partially melt

Ref. A. Pshenichnikov et al., On the degradation progression of a BWR control blade under high-temperature steam-starved conditions, Mechanical Engineering Journal, Vol.7, No.3 (2020), P.1-10.

(2) RPV lower head failure

Possibility of eutectic reaction between metallic debris and structural materials at 1050 - 1250 °C and relocate into CRD tube

Ref. T. Yamashita et al., BWR lower head penetration failure test focusing on eutectic melting, Annals of Energy, Vol.173 (20202), 109129.

Large-scale test facility at Tomioka in Fukushima

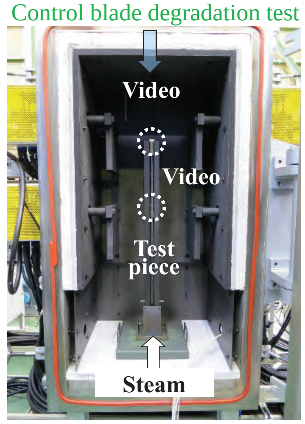
4

LEISAN facility at Tomioka-site
(Large-scale Equipment for Investigation of Severe Accidents in Nuclear reactors)

Unique specifications

- **Rapid heating（0.3～1℃/s）**
  Heat-up rate depending on decay heat
- **Temperature gradient（around 500℃/m）**
  Axial temperature gradient depending on water in lower part
- **Variable steam flow rate**
  Steam flow rate depending on blockage situation
- **Video in-situ observation**
  Real time information on melt, relocation and blockage
- **Measurement of aerosol and gas**
  Real time information on hydrogen, carbo dioxide etc.
- **Usage of almost same sized test piece**
  To glove experimental knowledge actual melt, relocation and blockage

Control blade degradation test



RPV failure test

Test section of LEISAN

Summary of control blade degradation behavior

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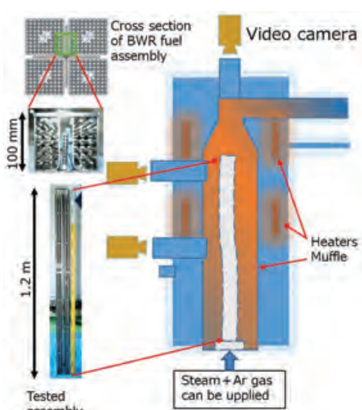
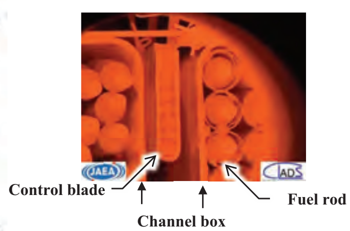
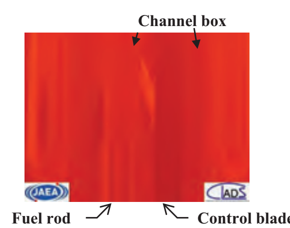
Degradation test of mock-up fuel assembly with control blade for BWR has been performed in JAEA/CLADS at Tomioka to understand the degradation under simulated 1F accident scenario.
Necessary information for the characterization of fuel debris (especially for the solidification of molten metallic materials) was obtained.



LEISAN facility





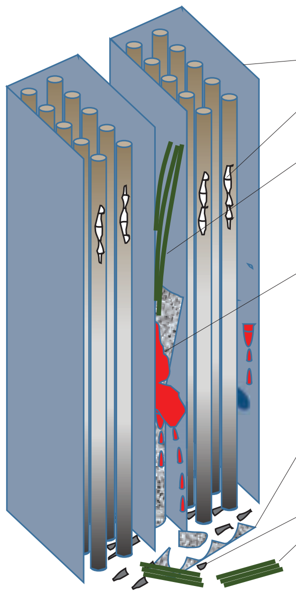
【Melting and relocation behavior of the metallic materials】

- The control blade partly melted. A part of unmelted control blade degraded and collapsed to the bottom.
- The channel box remained its original form.
- At low temperature range (<1200℃), the control blade melt (molten metallic materials) solidified, which "partially" blockaded the flow channel of the cooling gas.

At the initial phase of fuel debris retrieval, there is lack of information. Using simulated materials, this study proposed a realistic approach to reduce uncertainty and risk of fuel debris retrieval.

Three types of debris at the beginning of accident

6



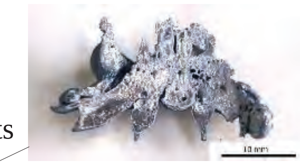
Type II Oxidic debris

Type III Degraded rests of a control blade or assembly parts

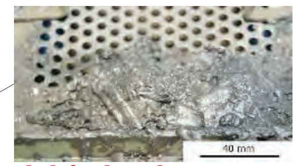
Type I Molten and solidified metallic debris blockage

Type I Relocated and solidified metallic debris

Type III Relocated parts of a control blade and the other assembly parts



Stone-like accumulates



Solidified molten material

Assembly partially molten parts

RPV lower head failure modes

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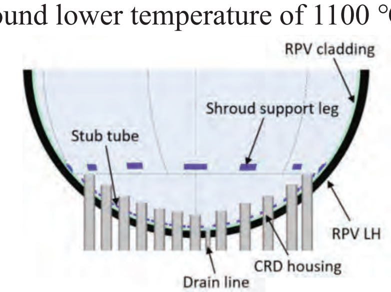
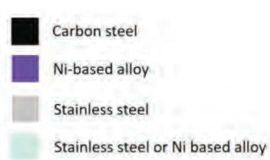
To assume main three fracture modes*:

- Melting above melting point of each structural material
  Formation of molten materials around 1500 ℃ of melting points by decay heat
- Creep fracture by mechanical loading
  Creep deformation and damage by debris' self-weight and thermal stress
- Eutectic reaction between molten and structural materials
  Reaction between molten or solid metallics and structural materials around lower temperature of 1100 ℃

→ In terms of the impact of RPV failure, failure occurs at a temperature lower than the melting point of the structural materials.

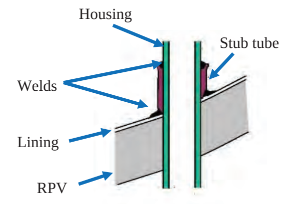




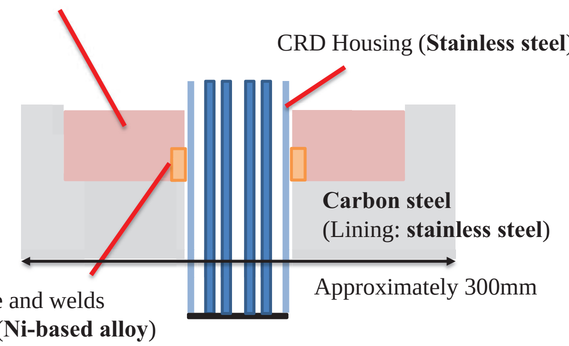
* It depends on scenario. In addition, fracture might happen by multiple fracture modes

RPV failure test – test piece - 8

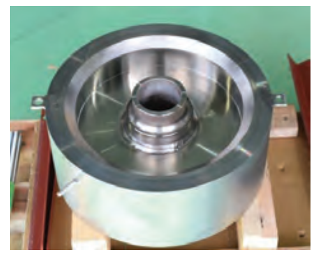
Appearance of test piece before test



Simulated metallic debris (Fe-83wt.%Zr)
the simulated material: pure metals or slightly oxidation
(in actual case, partial or full oxidation of Zr by steam)



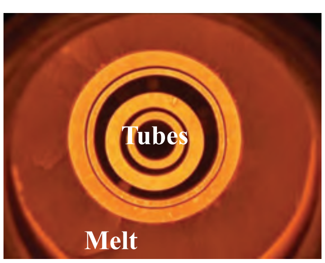
Cross-sectional image of test piece

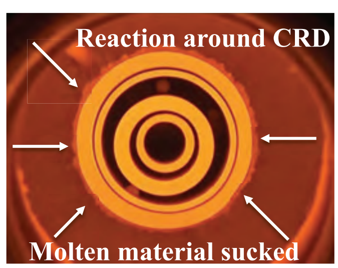


Photo

In-situ observation by video 9

Melting of Fe-Zr compound (1000 °C)	Temp. of CRD (1050 °C)	Temp. of CRD (1250 °C)
Eutectic point Fe-Zr compound: 950 °C	Recognition of reaction around CRD housing and of possibility of sucked molten material up	Failure of stub tube





Approx. 35 min. Approx. 40 min.

Heating rate: Approx. 4 °C / min.

Summary of RPV failure by eutectic reaction

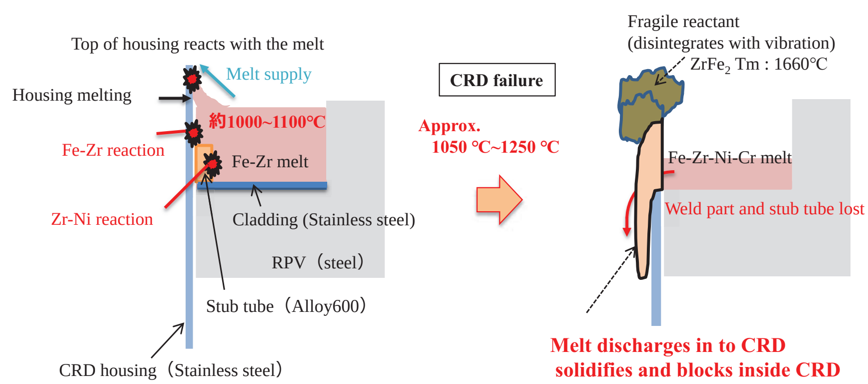
10

Key features:

✓ Melting of the structure proceeds mainly by two eutectic reactions
metal-debris - CRD housing(stainless steel) : Fe-Zr
metal-debris - stub tube(Ni based alloy) : Zr-Ni

✓ CRD failure proceeds at temperatures lower than the melting point of the structure itself
approx. 1050-1250°C

✓ CRD structural part embrittled
Fragile reactant formed, stub tube lost



Summary of current experimental output

11

JAEA/CLADS has conducted two types of large-scaled experiments using the LEISAN facility at Tomioka-site (Large-scale Equipment for Investigation of Severe Accidents in Nuclear reactors). In-situ and post investigation provide updates of understanding for debris formation.

Current experimental output on degradation behaviors:

(1) Control blade degradation

Three types of debris' formation: Metallic, Oxidic and Partially melt

(2) RPV lower head failure

Possibility of eutectic reaction between metallic debris and structural materials at 1050 - 1250 ℃, and relocate into CRD tube

To understand character or distribution of several types of debris experimental tests by combined with and debris' samples- and accident- analysis :

Accident analysis

Thermal hydraulic conditions

- temperature, pressure

Degradation mechanism

conditions

Sample analysis

Characterization of actual debris

- Identify phases using analytical equipment
- Probable formation path by thermodynamics

conditions

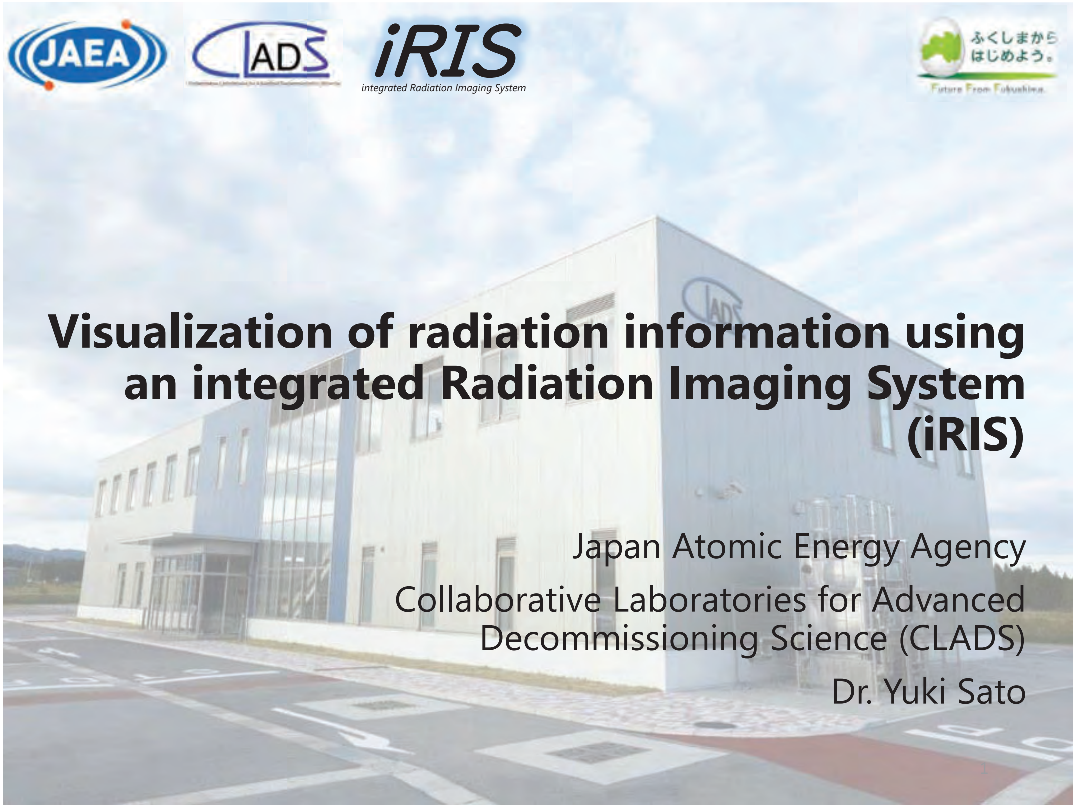
comparison

Separate Effects Tests

Experimental reproduction

- Relocation from in-vessel to ex-vessel
  - other RPV failure mode?
  - relocation path?
- Analysis of solidified products
  - Identify phases using analytical equipment

C.2.1.3. Visualization of Radiation Information



Importance of "visualization" of radioactive substances

Radiation work environment

- Nuclear power plants (including decommissioning work environment)
- Nuclear security (theft of radioactive materials and detection of terrorist acts using such materials)

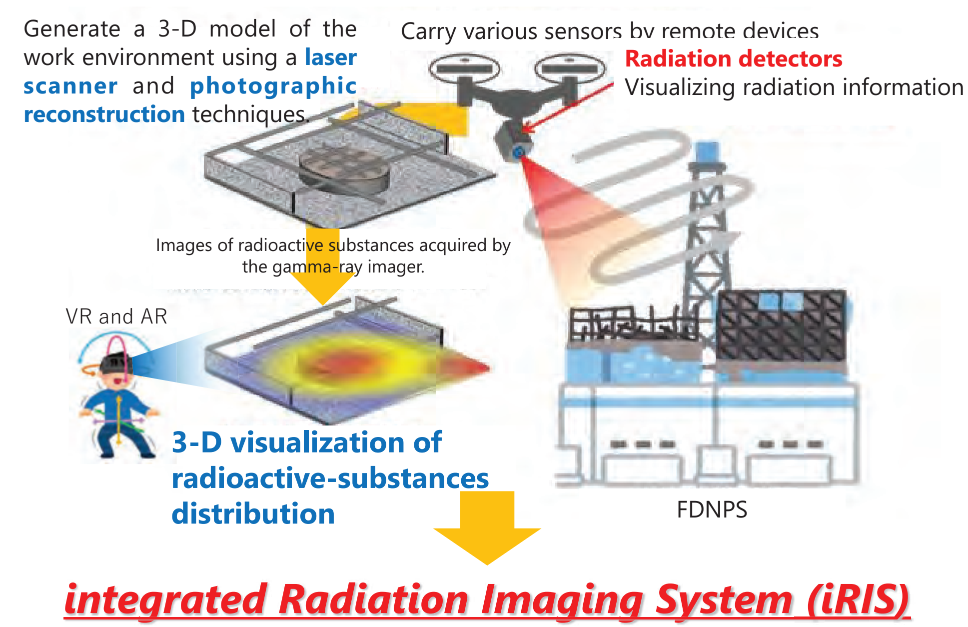
etc…

Why is it important to visualize radioactive substanes?

- Reduction of workers' exposure dose
- Detailed work planning (decontamination, dismantling, and radioactive substances detection)

2

Concept of integrated Radiation Imaging System (iRIS) by JAEA

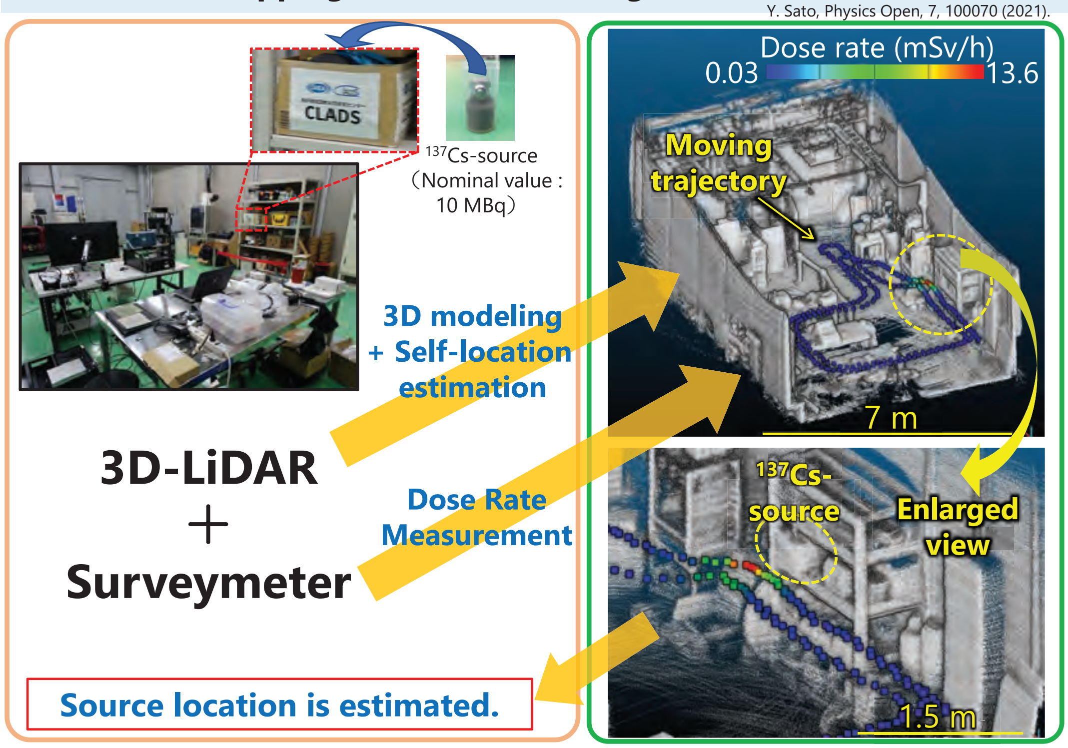


3

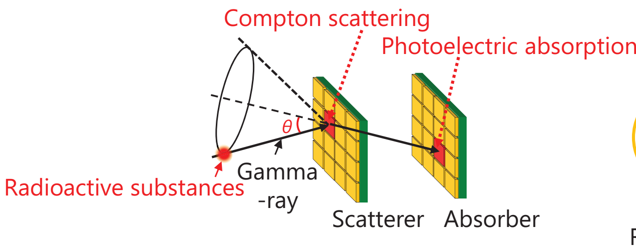
Mapping of dose rates using SLAM devices

4

Y. Sato, Physics Open, 7, 100070 (2021).



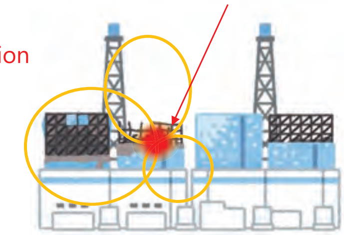
Compton camera



$$\cos\theta = 1 - \frac{m_e c^2}{E_2} - \frac{m_e c^2}{E_1 + E_2}$$

Measure the **interaction position** and **deposited energy** of gamma rays on the scatterer and absorber.

⇒ Estimate the **scattering angle** of gamma rays (Compton cone)

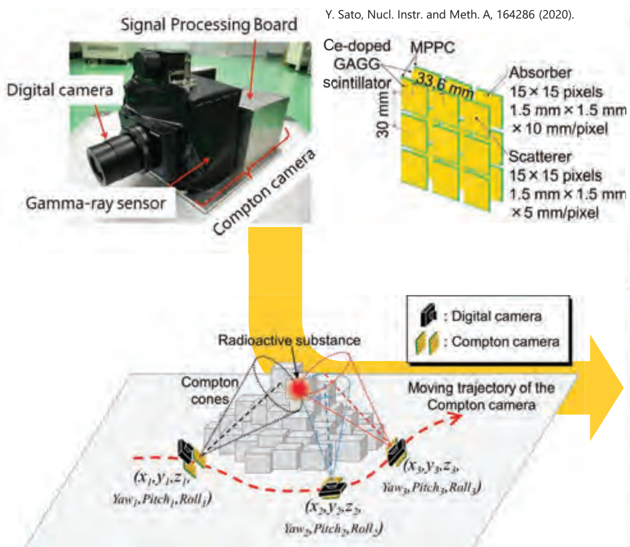


Radioactive substances can be **estimated at the intersection of the Compton cones** in the field of view.

Visualization
of radioactive substances

Combination of Compton Imaging and SLAM

Y. Sato, Nucl. Instr. and Meth. A, 164286 (2020).

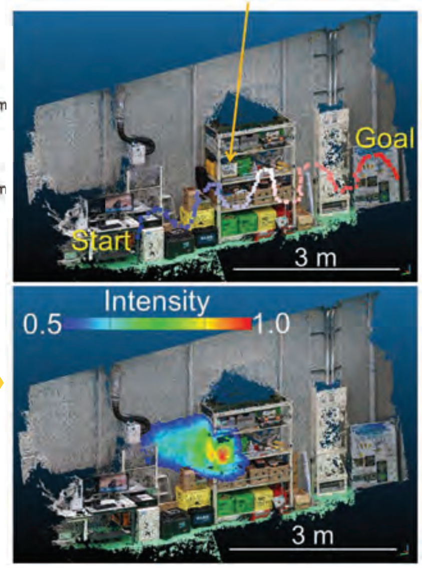


Self-position and posture information of the moving Compton camera is estimated by SfM.

SfM: Structure from Motion

Y. Sato, Physics Open, 7, 100070 (2021).

^{137}Cs-source（Nominal value :10 MBq）



- 116 photographs were taken with measurements taken every second.
- MetaShape Professional was used for SfM.

The Compton camera is based on the gamma catcher manufactured by Chiyoda Technol Corporation, which was jointly developed by Waseda University and Hamamatsu Photonics.

VR experience of radiation environment

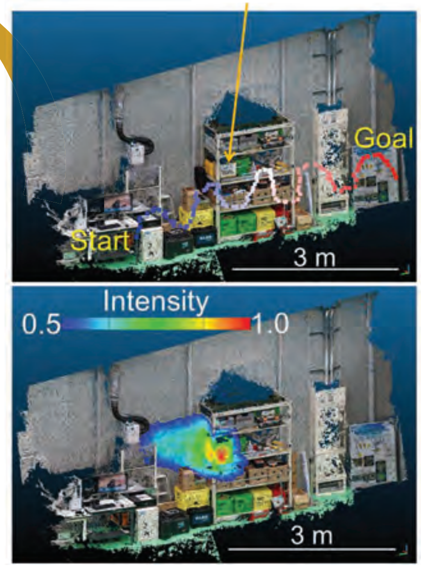
Y. Sato, Physics Open, 7, 100070 (2021).

Controller

VR head mounted display

- HMD: Meta Rift S, Meta Quest 2
- **Unity** was used as VR engine

^{137}Cs-source（Nominal value :10 MBq）



- Experience the work environment from any location using **commercially available VR-HMD**.
- The location of the radiation source can be grasped **inside the VR space**.

- 116 photographs were taken with measurements taken every second.
- MetaShape Professional was used for SfM.

7

VR experience of radiation environment

Y. Sato, Physics Open, 7, 100070 (2021).

Controller

VR head mounted display

- HMD: Meta Rift S, Meta Quest 2
- Unity was used as VR engine

Import

Dose rate data [mSv/h]

- It is acquired by a survey meter.
- Alternatively, they are **calculated from the radioactivity estimated by the gamma imager**.

Calculation

The amount of **worker exposure can be calculated based on the time spent** at the work site.

- Experience the work environment from any location using **commercially available VR-HMD**.
- The location of the radiation source can be grasped **inside the VR space**.

8

AR display of radiation information

The user recognizes the **AR marker with the reading device (app)** and displays radiation information such as air dose rate and hot spot images **in real space**.

●**AR Marker**

- it is necessary to install these markers at the work site in advance.
- It is **not desirable** to go to the radiation environment to **install AR markers prepared in advance**.

●**GPS:**
There are technologies that use GPS as AR markers, but they are **difficult to use inside a building**.

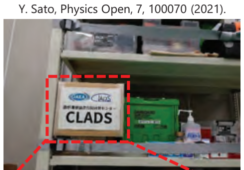
Cardboard box containing ^{137}Cs radiation source

Use photos as AR markers

9

AR display of radiation information

Y. Sato, Physics Open, 7, 100070 (2021).



One of the photos taken for 3-D modeling as an **AR marker**.

Extraction of feature points (Corners and edges)

Display connected to optical camera

Source image acquired with Compton camera

^{137}Cs radiation source

A cardboard box containing a radiation source is held up to an optical camera.

Feature point extraction from photographs and AR display were constructed using Processing and nyar4psg libraries, which are open source for **digital art rendering**.

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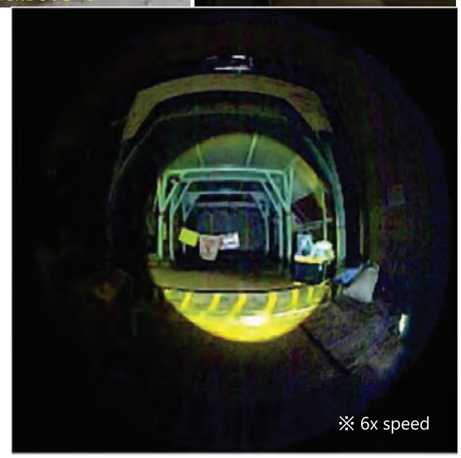
Robotic radiation imaging

In actual radiation work environments, the **use of robots** is desirable from the viewpoint of worker exposure.

Hotspot detection inside the **Unit 1 reactor building**

- ◆Compton camera **mounted on a crawler robot.**
- ◆27 mm-thick lead shields mounted on all sides of gamma-ray sensor.
- ◆Air dose rate in the test area: 5 mSv/h or higher.
- ◆Measurement was **performed remotely from the Main Anti-Earthquake Building** of FDNPS.

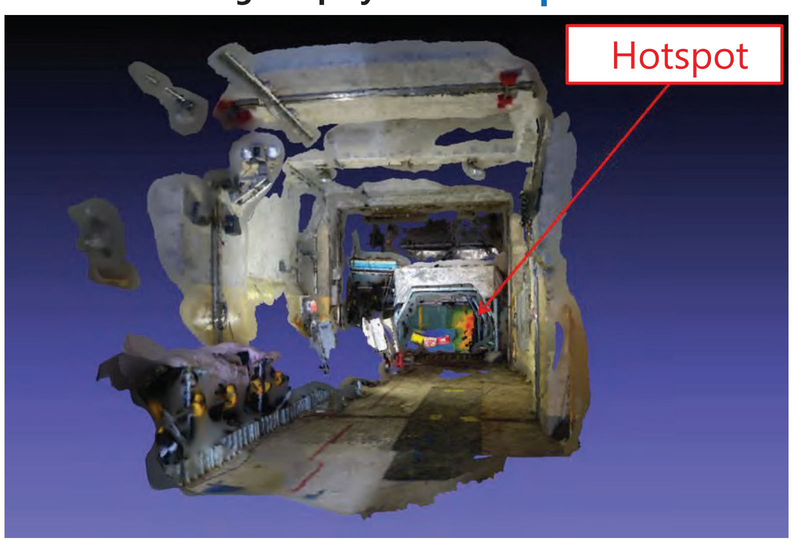


Y. Sato, et. al., Journal of Nuclear Science and Technology, 56, pp. 801-808, (2019)
JAEA Press Releases, August 28, 2018

11

Visualization of radioactive hotspot at the decommissioning site on a 3D map

Radiation image display on the **3D photo-model**



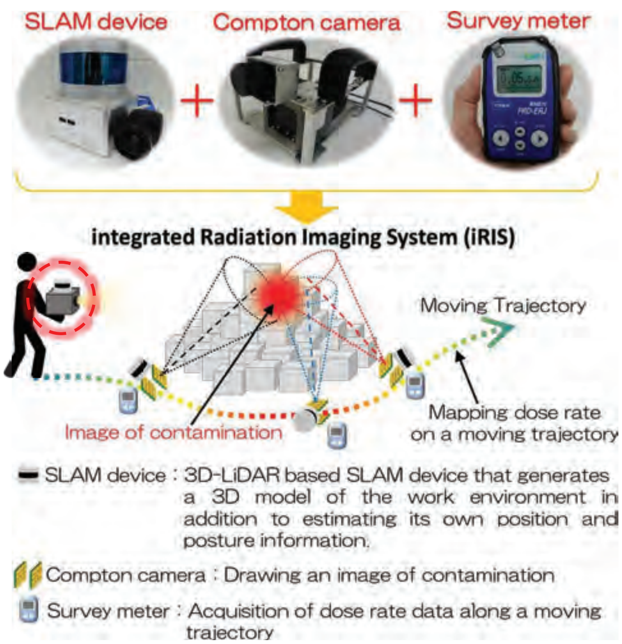
Y. Sato, et. al., Journal of Nuclear Science and Technology, 56, pp. 801-808, (2019)
JAEA Press Releases, August 28, 2018

12

Generate 3-D maps of radioactive contamination while moving

Scan the entire work environment

Generate 3-D maps
visualizing contaminated areas and air dose rates



The Compton camera is based on the gamma catcher manufactured by Chiyoda Technol Corporation, which was jointly developed by Waseda University and Hamamatsu Photonics.

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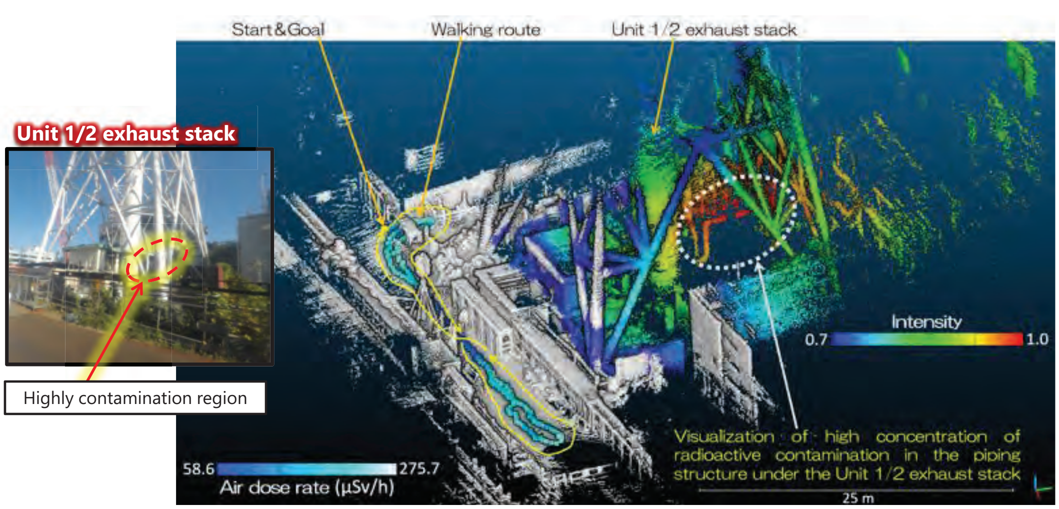
JAEA, press release, 14th, May 2021 & Y. Sato, et. al., Journal of Nuclear Science and Technology, 59 pp. 677-687 (2022)

Visualization of radioactive contamination at the decommissioning site on a 3D map



Creation of 3-D map
visualizing air dose rates and radioactive contamination

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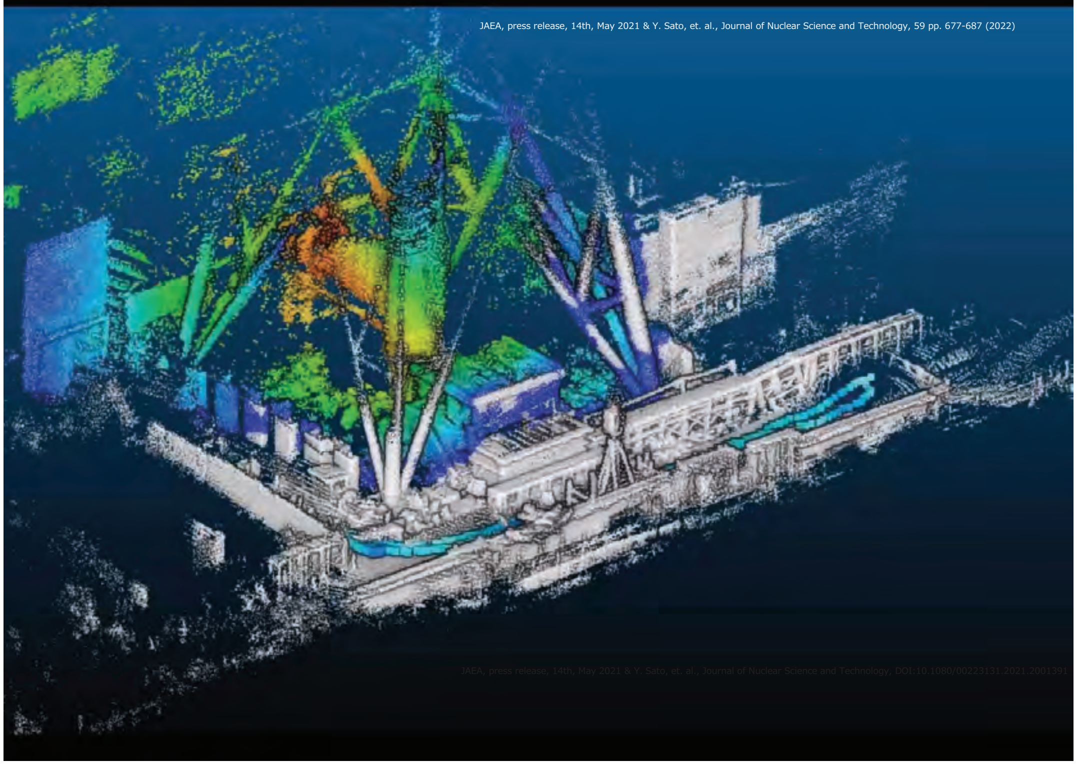
JAEA, press release, 14th, May 2021 & Y. Sato, et. al., Journal of Nuclear Science and Technology, 59 pp. 677-687 (2022)



Visualization and Radioactivity Estimation

JAEA, press release, 14th, May 2021 & Y. Sato, et. al., Journal of Nuclear Science and Technology, 59 pp. 677-687 (2022)

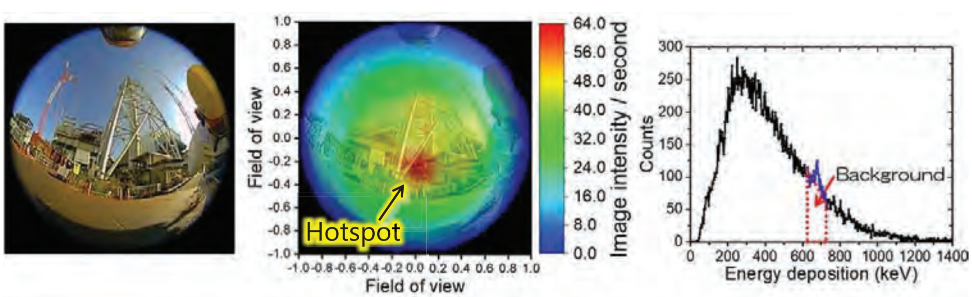


Image intensity

Compensates for attenuation of gamma-ray counts by the ***inverse square of the distance***

$$Q = C \cdot \frac{I \cdot d^2}{e^{-\mu d} \cdot \varepsilon_{loss} \cdot \varepsilon_{eff}}$$

Radioactivity

Proportionality constant *(Derived from the known radioactivity and the image intensity at that time)*

Compensates for attenuation due to ***air (shielding)***

Compensates for the ***count omission***

Compensates for ***BG***

$Q = (3.0 \pm 0.5) \times 10^{12}$ Bq

Consistent

$Q = 9.23 \times 10^{12}$ Bq

Estimated value of radioactivity using pinhole camera by the Japan Nuclear Regulation Authority*.

※Hirayama H, Hayashi K, Iwanaga K, et al. Transactions of the Atomic Energy Society of Japan. 2014; 9: P11025 [Japanese].

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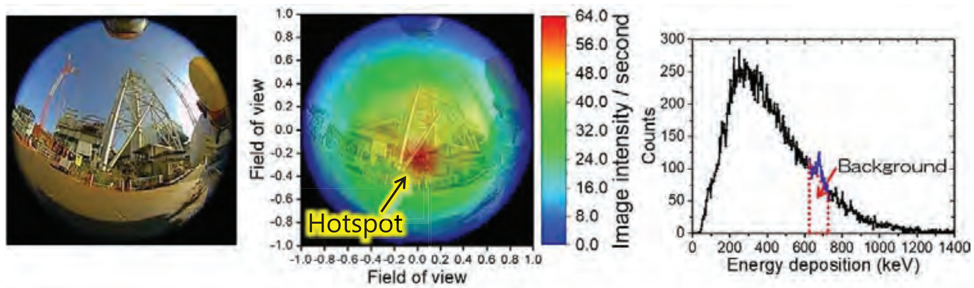
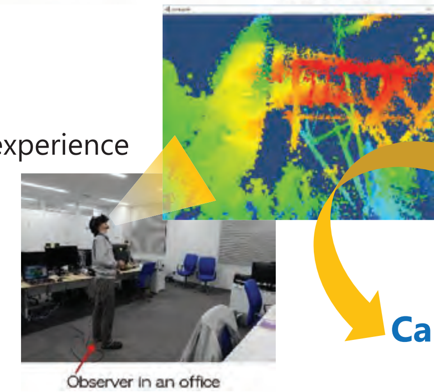
Visualization and Radioactivity Estimation



VR experience



$Q = (3.0 \pm 0.5) \times 10^{12}$ **Bq**

The radioactivity values **could be used to calculate the ambient air dose rate**.

Can calculate worker exposure

Y. Sato, et. al., Journal of Nuclear Science and Technology, 59 pp. 677-687 (2022)

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Conclusion

✓ Visualization of radioactive substances inside FDNPS was introduced.

✓ Progress from 2-D visualization to 3-D visualization.

✓ In addition to radiation measurement, environmental recognition, robotics, and mapping technology must be integrated.

Future Prospects

✓ Integration with VR, AR, *i.e.*, cross reality technologies

✓ In today's presentation, only the mapping of gamma nuclides was presented. Visualization techniques for alpha and beta nuclides should also be integrated.

✓ Contribute to decommissioning through visualization of radioactive substances.

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Acknowledgement

The authors deeply appreciate the highly technical-based support for the demonstration test at the FDNPS conducted by the engineers of **ATOX Co., Ltd.,** and the irradiation test of the Compton camera conducted by the engineers of **Chiyoda Technol Corporation.** The authors would also like to thank engineers of **Tokyo Electric Power Company Holdings, Inc. (TEPCO)**, and engineers of **Visible Information Center, Inc.** for supporting the development of the 3-D reconstruction of the radiation image. The authors also wish to acknowledge engineers of **Hamamatsu Photonics K.K., Prof. J. Kataoka of Waseda University** for the development of the base technologies of the compact Compton camera.

C.2.2. NDF 2022 Technical Strategic Plan for 1F D&D

Technical Strategic Plan 2022 for Decommissioning of the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company Holdings, Inc.

(Explanatory Material)

Reactor Safety Technology Expert Panel
Forensics Meeting

November 17, 2022

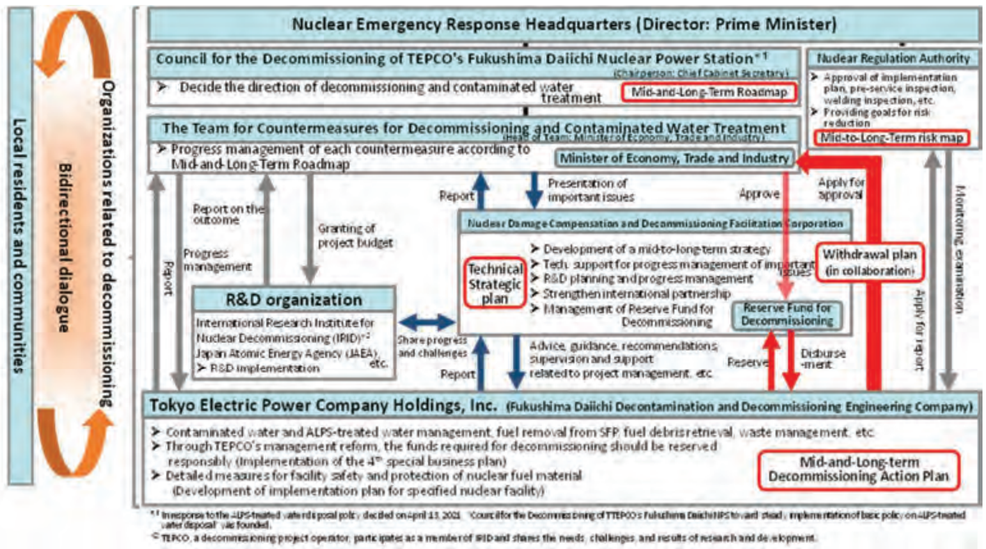
Nuclear Damage Compensation and Decommissioning Facilitation Corporation

NDF

Table of Contents

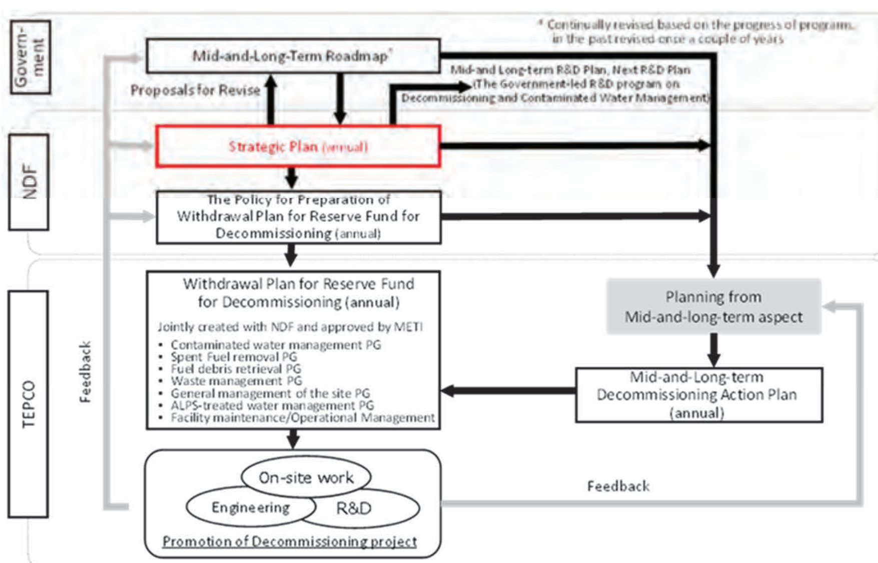
1. Introduction

Division of roles of related organizations responsible for decommissioning of the Fukushima Daiichi NPS



1. Introduction

Positioning of the Technical Strategic Plan

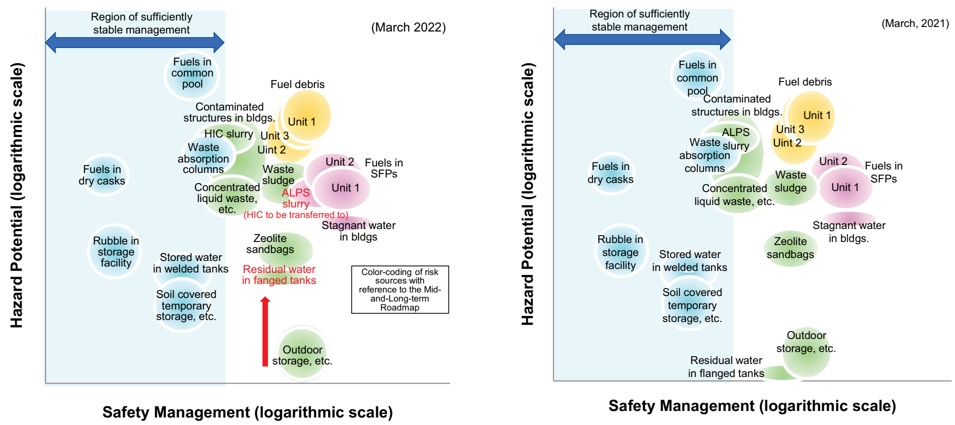


2. Concept for reducing risks and securing safety for decommissioning of the Fukushima Daiichi NPS

Concept on risk reduction

- The interim target of the risk reduction strategy is to bring the risk levels into the “Sufficiently stable management” region (the pale blue area)

The Technical Strategic Plan 2022 (Reference)The Technical Strategic Plan 2021



※1 The red letters present major changes from the Technical Strategic Plan 2021.
※2 The origin of the arrow indicates the location for the residual water in flanged tanks reported in the Technical Strategic Plan 2021, which has been moved upward to reflect the results of the analysis of the radioactivity concentration prior to the treatment. ALPS slurry (HIC to be transferred) was separated from ALPS slurry this time and shown in pink, as is the fuel in the pool.

Fig. Risk levels posed by major risk sources at the Fukushima Daiichi NPS

2. Concept for reducing risks and securing safety for decommissioning of the Fukushima Daiichi NPS

Approach to ensuring safety during decommissioning

- As for decommissioning of the Fukushima Daiichi NPS containing the reactors involved in the accident, its peculiarities regarding safety should be fully recognized to ensure safety and sufficient attention should be paid to “the safety perspective” and “the operator’s perspective”.
  - ✓ Safety perspective : Ensuring safety should be the starting point for consideration. Determining the most appropriate safety measure (ALARP※)
  - ✓ Operator’s perspective : Perspectives and judgements from the standpoint of those who are familiar with the site and perform operations on site

Peculiarities of Fukushima Daiichi NPS

- ✓ A large amount of radioactive material is in an unsealed state, and in unusual and various (atypical) forms
- ✓ Barriers for containing radioactive materials are incomplete
- ✓ Significant uncertainties exist on the state of radioactive materials and containment barriers
- ✓ Difficulty in accessing the site and installing instrumentation devices to obtain on-site information
- ✓ Since the current level of radiation is high and further degradation of containment barriers is a concern, it is necessary to take measures in consideration of the time axis without prolonging the decommissioning activities

※ Abbreviation of As Low As Reasonably Practicable. This is the principle that the radiological impact must be as low as reasonably achievable.

3. Technological strategies toward decommissioning of the Fukushima Daiichi NPS 3.1 Fuel debris retrieval

Major targets and progress for fuel debris retrieval

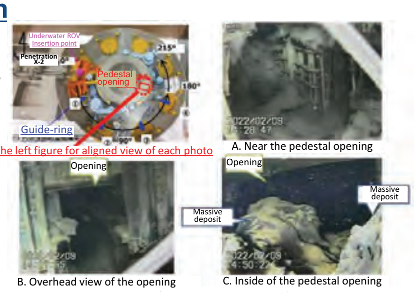
Major targets

- Trial retrieval in Unit 2 was scheduled to begin within 2021, but the process will be reviewed to improve the safety and certainty of the operation in light of the impact of the COVID-19 pandemic, the mock-up testing that has been conducted since February 2022, and the current on-site situation. The work is expected to start in late FY2023.
- For further expansion of fuel debris retrieval, consideration will be given to the methods including those for containing, transferring, and storing of fuel debris, by assessing fuel debris retrieval in Unit 2, internal investigations, research and development, and the on-site environmental improvement, etc.

Progress

Status of internal investigation in Unit 1 PCV

- So far, massive deposits have been observed and it was confirmed that concrete in the pedestal was partially missing in the vicinity of the worker access opening.
- Based on the past assessment by IRID and the observations of the pedestal, for partially missing concrete, TEPCO assumes that it is unlikely to lead to large-scale damage or other problems.
- It is necessary to expand its findings through further internal investigations and to conduct the impact assessment on the plant.



(Source: The Secretariat Meeting of the Decommissioning and Contaminated water treatment team meeting, Feb.24, 2022)

Investigation results at pedestal opening

3. Technological strategies toward decommissioning of the Fukushima Daiichi NPS 3.1 Fuel debris retrieval

Challenges and technical strategies for trial retrieval from Unit 2 (internal investigation and fuel debris sampling)

Challenges and technical strategies

- Trial retrieval (internal investigation and fuel debris sampling) is a series of operations, and fuel debris sampling is one part of 11 steps.
- After opening the hatch of the penetration X-6 and extending the containment barrier outside the PCV, it is important to ensure containment as the inside of the enclosure becomes progressively contaminated.
- For on-site applications with uncertainty, the challenges are to ensure functionality verification under various conditions and equipment can be rescued in case of emergency.

✓ It is necessary to ensure that the required conditions are satisfied by conducting mock-up tests.

✓ Due to the uncertainty of the PCV internal situation, work must be performed safely and carefully, keeping in mind that things may not go as planned.

Work steps

1. Preparations (finished)
2. **Install the isolation chamber ← being implemented**
3. Open the hatch of the penetration X-6
4. Remove deposits inside the penetration X-6
5. Install the robot-arm
6. Enter the robot-arm
7. Internal investigation/fuel debris sampling
8. Collect from fuel debris retrieval equipment to transport container/measure dose
9. Accept into glove box/measure
10. Remove container/store in canister and carry out
11. Off-site transport and off-site analysis

Further expansion of fuel debris retrieval in Unit 3

Challenges and technical strategies

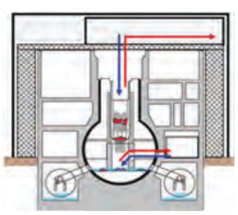
- A variety of retrieval methods have been studied since FY2021, not ruling out any possibilities.
- The dry method*[1] and submersion method*[2] that are being discussed pose the following challenges: feasibility of on-site construction under high radiation dose, significant increase in the amount of construction materials and waste, and measures to be taken when fuel debris is retrieved.

✓ Once feasibility has been confirmed to some extent, it will be necessary to narrow down the options step by step while proceeding with the design.

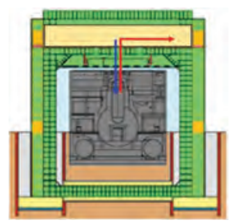
✓ Other retrieval methods should be studied as needed.

*1 A method combining top-access method and side-access method
*2 Unlike the conventional PCV submersion method, a method of submersing the reactor building by enclosing the entire reactor building with a new structure as a boundary (Shell method)

Direction of equipment access
Direction of fuel debris/waste carrying-out

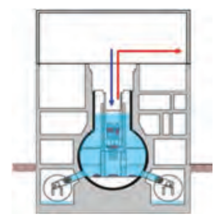


An example of dry methods
(Conceptual drawing of Combination of top-access and side-access)



An example of submersion methods
(Conceptual drawing of Shell method)

<Reference>



Conventional submersion method
(Diagram of PCV submersion method)

Major targets for waste management

Major targets

- The Solid Waste Management Plan is appropriately developed, revised and implemented, with updating the estimated amount of solid waste to be generated in the next ten years periodically. According to this Plan, temporary outdoor storage of the solid waste will be eliminated completely by FY 2028 (except for secondary waste generated by water treatment and targets of reuse/recycling).
- Given the prospects of processing/disposal methods and technology related to their safety presented in FY 2021, appropriate measures should be studied as management approaches for overall solid waste to establish a waste stream* according to the properties of solid waste.

*A series of handling procedures for each type of waste, from generation/storage to processing/disposal.

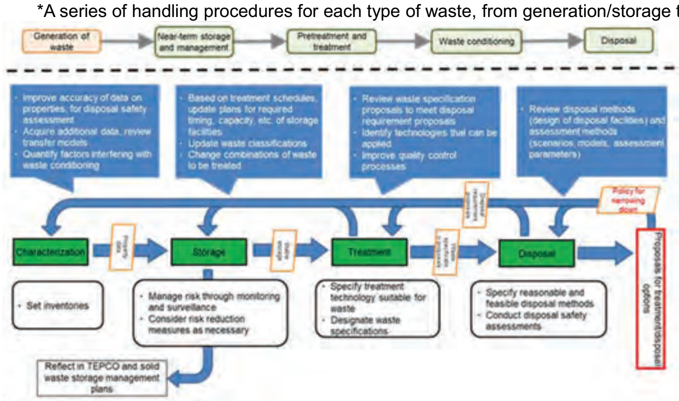


Fig. Procedure to reasonably select safe processing/disposal methods of solid waste

Major challenges and technical strategies for waste management

Challenges and technical strategies

Characterization

- For a variety of solid waste, it is needed to develop a medium-to-long-term analysis strategy that defines its priority, the objective of the analysis, and quantitative targets, etc., and to proceed with analysis/evaluation accordingly.

✓ Accumulate trial results and verify their validity in order to establish a development flow of the medium-to-long-term analysis plan with an analysis project approach using statistical methods,.

Storage

- Storage of solid waste according to the progress of decommissioning work in the future should be advanced in a safe and reasonable manner, including temporary outdoor storage of the solid waste (except for secondary waste generated by water treatment and targets of reuse/recycling) will be eliminated (by FY 2028), that is stated in the Mid-and-Long-term Roadmap.

✓ Examine further possibilities by referring to advanced cases of overseas, while steadily continuing approach for volume reduction.

✓ Promote volume reduction through incineration, and cutting/crushing, and steadily consolidate storage inside buildings.

Processing/disposal

- The Mid-and-Long-term Roadmap stated that the specifications of waste form and their production methods will be determined in Phase 3, the study on appropriate overall measures should be initiated for specific management for solid waste.

✓ Create processing/disposal options for solid waste by examining pending issues related to processing technology and disposal options.

✓ Compare and evaluate options using the property data that are becoming clear, and examine to establish a waste stream that is suitable for the characteristics of solid waste.

Major targets and technical strategies for contaminated water management

Major targets

- To arrange the relationship with a decommissioning process including full-scale fuel debris retrieval beginning in the near future, and to promote examination of the measures of the contaminated water management for medium-and-long term prospects.

Challenges and technical strategies

- The water quality of contaminated water depends on the cutting and fabrication method (forms of α-nuclides)
- It is difficult to assume the water quality in a situation where the fuel debris retrieval method has not been determined. The water treatment systems should have a system configuration to cope with a wide range of water quality.

✓ It is necessary to determine the required specifications for the water treatment systems when retrieving fuel debris, then incorporate into basic design promptly in order to review the overall picture in considering the share of functions with the existing systems and to promote planned replacement of the existing systems.

3. Technological strategies toward decommissioning of the Fukushima Daiichi NPS
3.3 Contaminated and treated water management

Major targets and technical strategies for discharging ALPS-treated water into the ocean

Major targets

- For the ALPS-treated water currently stored in tanks, measures will be taken for discharging the treated water about two years after the Basic policy (released in April 2021)

Challenges and technical strategies

- In addition to "reliably" operate a series of plans including system operation, analysis of ALPS-treated water, maintenance and response measures in the event of trouble, TEPCO must review and expand the plans as needed and ensure its transparency.

✓ It is necessary to reassess radiation impacts on human and environment based on the nuclides to be analyzed, and disseminate the assessment results with high transparency.

3. Technological strategies toward decommissioning of the Fukushima Daiichi NPS
3.4. Fuel removal from spent fuel pools

Major targets and technical strategies for fuel removal from spent fuel pools

Major targets

- The aim is to complete fuel removal from all spent fuel pools of Units 1 to 6 within 2031.
- To start removal of fuel in SFPs in FY 2027 to FY 2028 for Unit 1 and FY 2024 to FY 2026 for Unit 2.

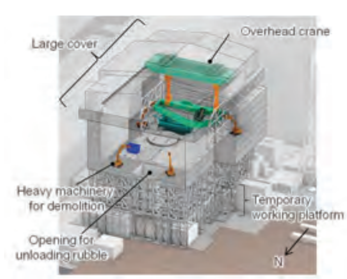


Fig. During rubble removal from Unit 1 (Conceptual drawing)

Challenges and technical strategies

Unit 1 In order to remove the overhead crane in an unstable state, a thorough investigation is needed.
→ It is important to promptly investigate it as soon as investigation becomes possible, and incorporate into safety assessment and rubble removal plans.

Unit 2 The challenge is to ensure that a fuel handling machine with a boom-type crane, which has not been used for nuclear facilities in Japan, is operated remotely.
→ It is important to be sufficiently familiar with the operation and functionality of systems beforehand.

4. Analysis strategy for promoting decommissioning

Significance and technical strategies of analysis

Significance and current state

- At present, due to the large range of uncertainty regarding fuel debris properties, safety measures should be studied conservatively.
- If the range of such uncertainty can be reduced, there is no excessive margins needed, and thus, rational safety measures can be studied, it enables the promptness and rationality of decommissioning to improve.

Challenges and technical strategies

- Since the fine fuel debris generated as retrieval of fuel debris progresses are diverse with high radiation dose, it is a challenge to establish an efficient system for analysis.

✓ It is effective to expand the analysis data under the appropriate division of roles among the facilities in the Ibaraki area, where facilities and equipment are enhanced, and the new analytical facilities.

✓ It is important to efficiently promote human resource development with the cooperation of other institutions.

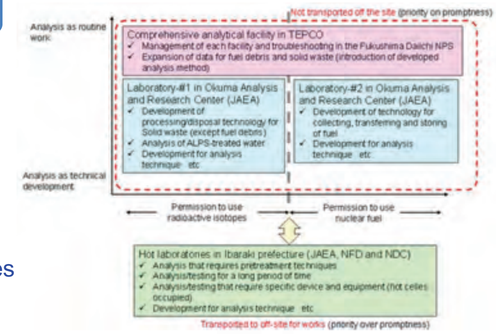


Fig. Characteristics and roles of each facility for analysis

4. Analysis strategy for promoting decommissioning

Improvement of the quality of sample analysis results and use of non-destructive measurement

Challenges and technical strategies

- Sample analysis can perform many analysis tasks, but the time required is long and the amount analyzed is small. So, it is difficult to measure a larger quantity analyzed.

✓ For non-destructive measurement, time measured is shorter than that of sample analysis, and a larger quantity can be measured per measurement.

✓ In order to improve the accuracy of fuel debris characterization, application methods of non-destructive measurement should be studied in the process from retrieval to storage.

Table. Relative comparison of principal specifications between the sample analysis to be performed in the analysis facility and non-destructive measurement to be performed outside the analysis facility

	Analysis of samples performed in analysis facility*	Non-destructive measurement performed out of analysis facility**
Time for analysis/measurement	Long (△)	Short (○)
Analysis/measurement items	Many (◎)	Few (△)
Amount per analysis/measurement	Small (△)	Large (◎)
Waste liquid	Generated (△)	None (○)
Confinement during analysis and measurement	Unsealed	Unsealed or sealed
Dust prevention	Necessary	Necessary
Radiation shielding facility	Necessary	Necessary

◎ : Excellent　○ : Good　△ : Acceptable

*: The analysis will be conducted in a facility dedicated to analysis, such as a hot laboratory suitable for dealing with fuel debris samples.
**: The facility will be used in the process from retrieving to storing fuel debris. The analysis will be conducted in a facility not dedicated to analysis.

5. Efforts to facilitate research and development

Significance and current state

- There are many difficult technical issues requiring research and development to promote the decommissioning from the perspectives of safe, proven, efficient, timely, and field-oriented.
- Eleven years have passed since the accident, the stage is currently shifting to promote development based on the engineering work by TEPCO.

Fig. Scope of studying decommissioning R&D and implementation entities

Strategy

- NDF plans to further strengthen the functions related to R&D planning and proposals and efforts to ensure operation quality.
  - ✓ Starting in 2022, a request for information (RFI) was made to widely solicit information on technical issues to be resolved (planning and proposals).
  - ✓ Review System will be established for all Projects of Decommissioning and Contaminated Water/Treated Water Management to ensure the actual site applicability of the Project and to improve the quality of R&D (ensuring the quality of the Project).
- TEPCO needs to be committed to decommissioning research more proactively including independent technology development by TEPCO, uniting with a new company*.

* *"Toso Mirai Technology Company"* established in October 2022

6. Activities to support our technical strategy

Project management approach

Significance and current state

- In order to facilitate decommissioning, establish and enhance the management system for achieving the goal of the project.
- Project management allows efficient risk reduction of the project by evaluating from the viewpoints of safety, quality, cost, time, technical feasibility and other visions.

Strategy

- Further enhancement of owner's engineering abilities
  - ✓ In facing unprecedented fuel debris retrieval, engineering judgments should be made and "project management capability" and "engineering based on safety and operator's perspectives" that are responsible for the results are needed.
- Developing and securing human resources for smooth implementation of decommissioning projects
  - ✓ Human resource allocation including required capability/competence and workforce count should be planned, and human resource development plan should be provided to achieve it.
  - ✓ Number of experts and time needed by technical field should be assumed, including human resource with higher expertise.

6. Activities to support our technical strategy

Strengthening international cooperation

Significance and current state

- Learn lessons from precedent overseas cases, and utilize the world's highest level of technology and human resources
- Sharing our decommissioning experience in Fukushima Daiich with the international community is Japan's responsibility.
- As an intergovernmental framework, annual dialogue has been held to share information with other countries. The relevant domestic organizations have concluded cooperative agreements with overseas organizations and have disseminated information at international conferences.

Strategy

- It is important to maintain and develop the international community's continuous understanding of and interest in decommissioning and cooperative relationships.
  - ✓ Continuing dissemination of accurate information on the decommissioning progress with ensuring transparency of information.
  - ✓ Eleven years have passed since the accident, it is important to deepen the mutually beneficial relationship while also working to return the know-how and accumulated lessons learned so far to the internal community.

Fig. Exchanging opinions with experts overseas through face-to-face meeting and online (held in June 2022)

6. Activities to support our technical strategy

Local community engagement

Significance and current state

- The fundamental principle for the decommissioning of the Fukushima Daiichi NPS is "Balancing between reconstruction and decommissioning". Revitalizing decommissioning-related industries is an important pillar of TEPCO's contribution to the reconstruction of Fukushima.
- Based on TEPCO's "Commitment to the people of Fukushima to achieve both reconstruction and decommissioning" established at the end of March 2020, efforts for the accumulating decommissioning industries need to be made with a view to creating opportunities for local enterprises to participate in the Fukushima Daiichi decommissioning project and a foundation for the local economy .

① Increased participation of local enterprises
② Support for local enterprises to step up

} Opening prospects of placing orders to local enterprises, 1F inspection tours, holding matching sessions for specific business negotiations, etc.
➢ Efforts need to be made to further create participation opportunities.

③ Creation of new local industries ⟶ Establishing joint ventures with partner companies for accumulating decommissioning industries in the Hamadori region.
➢ Creating new industries is desirable for accelerating the reconstruction of the local community

Strategy

* TEPCO Press release April 27/October 3, 2022
- *Toso Mirai Technology Company* (In collaboration with IHI Corporation)
- 【Tentative name】Hama-dori decommissioning-related product plant (in collaboration with Hitachi Zosen Corporation)

- It is important to consider initiatives that will enable local companies to receive constant and a certain scale of orders.
- Further strengthening of cooperation and collaboration with local governments, including Fukushima Prefecture, and local related organizations, including the Fukushima Innovation Coast Framework Promotion Organization and the Fukushima Soso Recovery Promotion Organization.

C.2.3. TEPCO Holdings Investigations

C.2.3.1. Recent 1F1 PCV Investigations and Findings

Unit 1 PCV
Recent Investigation Findings and Plans for Future Investigations

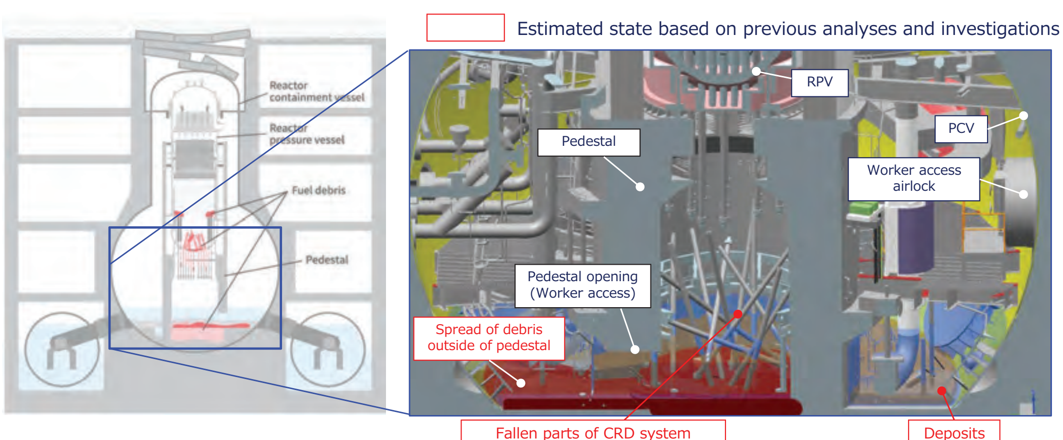
November 17, 2022

Michal Cibula
Tokyo Electric Power Company Holdings, Inc.

Acknowledgment:
Unit 1 PCV internal investigation results were obtained by using the robot developed in Subsidy Program "Project of Decommissioning and Contaminated Water Management" by METI of Japan.

General information

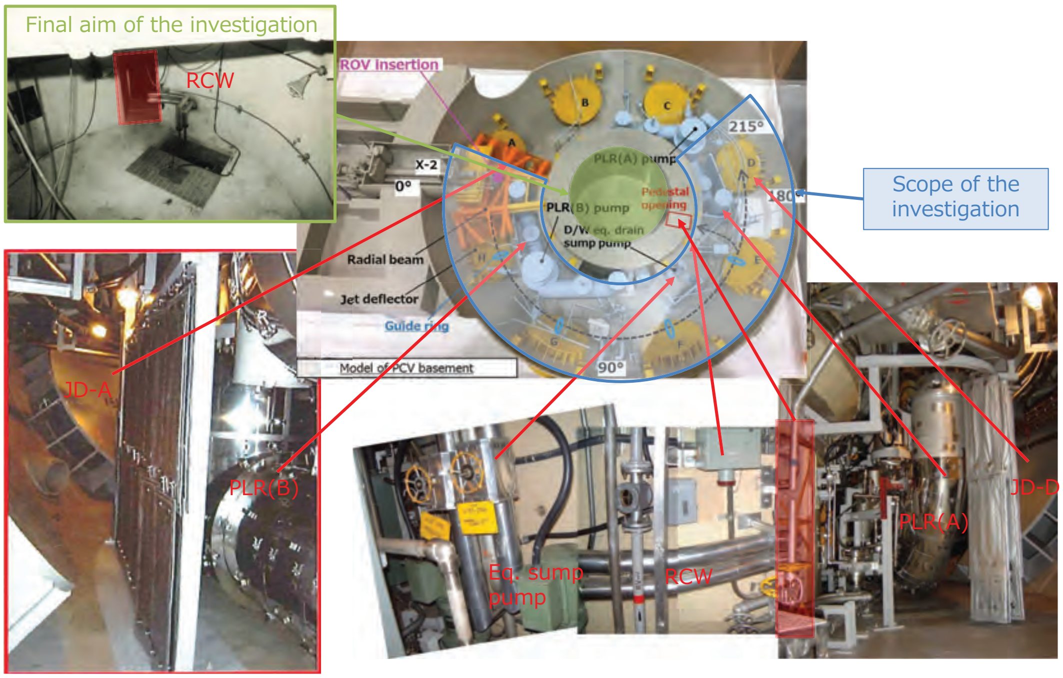
- Plant type: BWR-3
- Containment type: Mark I
- Investigation tool: submersible ROVs
- Investigated area: PCV D/W floor
- Water level: about 2 m
- Pedestal radius: 2.5 m
- Pedestal wall thickness: 1.2 m
- Pedestal axis to shell (floor lvl.): ~6.5 m
- Pedestal floor area: ~20 m^2
- D/W floor area: ~90 m^2



Source : International Research Institute for Nuclear Decommissioning (IRID)

1

State of the PCV basement before accident



Source : International Research Institute for Nuclear Decommissioning (IRID)

2

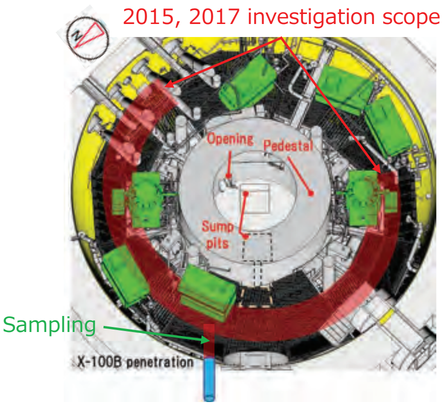
(Ref.) Basement of Unit 5 PCV

3

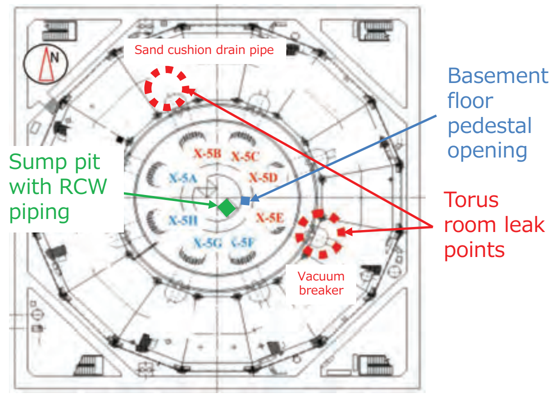
Investigation results up to 2022

IRID TEPCO

- High contamination of reactor building closed cooling water (RCW) system (2011 R/B investigation)
- Water leakage from sand cushion pipe (2013) and vacuum breaker (2014) in torus room
- No large fuel mass detected in the core region (cosmic muon radiography in 2015)
- 1st floor of PCV investigated (2015, 2017) and deposits found below on D/W floor
- Corrosion products and Pb, Sb largely found in sampled deposits (2017)



D/W 1st floor



D/W basement floor

Source : International Research Institute for Nuclear Decommissioning (IRID)

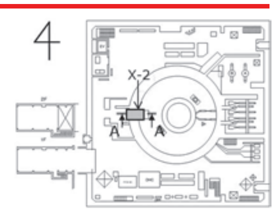
Investigation tool: Submersible ROV

IRID TEPCO

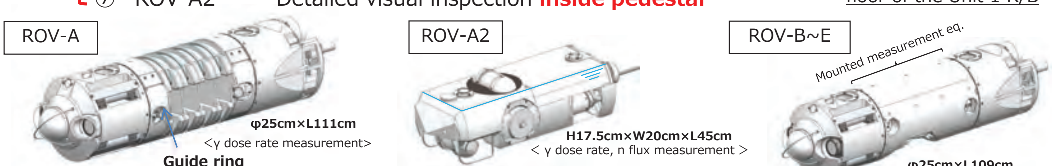
- 6 types of ROVs inserted to PCV D/W through X-2 penetration

1st half
- ① ROV-A — Guide ring installation ✓
- ② ROV-A2 — Detailed visual inspection ✓
- ③ ROV-C — Sediment thickness measurement ✓

2nd half
- ④ ROV-D — Debris detection
- ⑤ ROV-E — Sediment sampling
- ⑥ ROV-B — 3D mapping of sediments
- ⑦ ROV-A2 — Detailed visual inspection **inside pedestal**



Position of X-2 on the 1st floor of the Unit 1 R/B



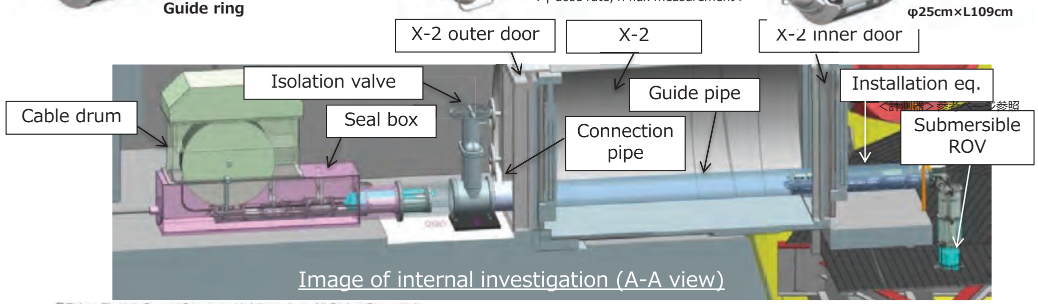
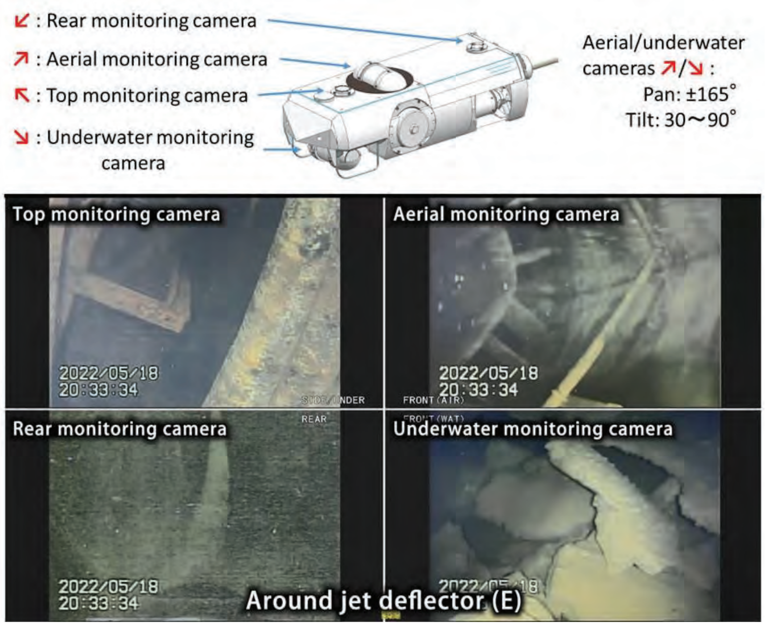


Image of internal investigation (A-A view)

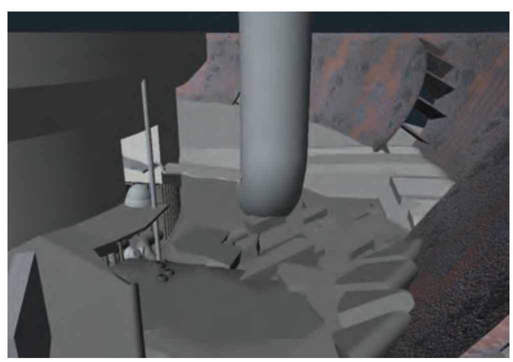
Source : International Research Institute for Nuclear Decommissioning (IRID)

Ref.: A-2 investigation videos



Videos available at:
https://www.tepco.co.jp/en/news/library/archive-e.html?video_uuid=sd7bw090&catid=69631

Visualization of state near the pedestal opening (animation)



State of the pedestal opening

- No significant damage to the upper part of concrete
- Deposits shelves on the right side of the wall increasing in height towards pedestal
- "lavacicles" on bottom surface of the shelves

Free space without water existed below shelves when "lavacicles" formed

Conduit

Pedestal opening before 2011

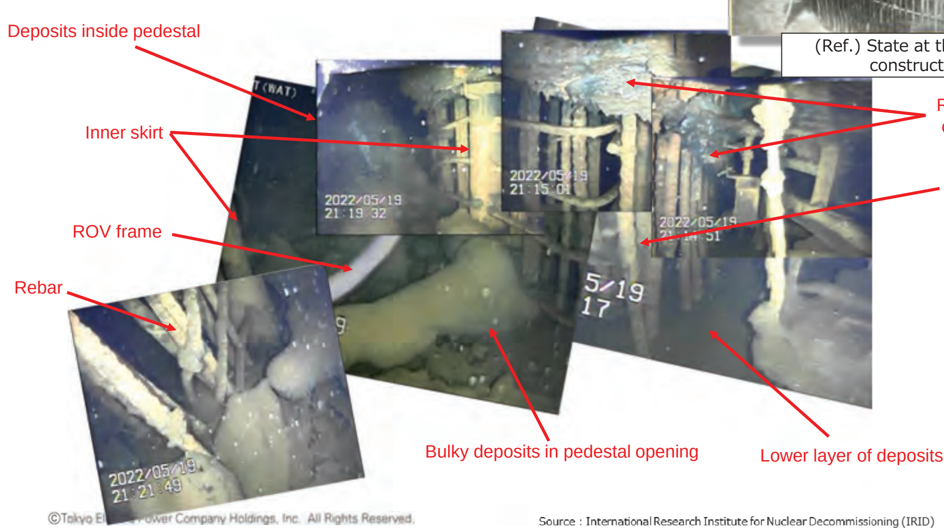
Conduit

Source : International Research Institute for Nuclear Decommissioning (IRID)

State of the pedestal opening

- Concrete degraded up to ~1 m from the floor
- No significant deformation of inner skirt
- Minor deformation of rebar (embossed pattern still visible)
- Height of the deposits inside the pedestal ~1 m

(Ref.) State at the time of construction



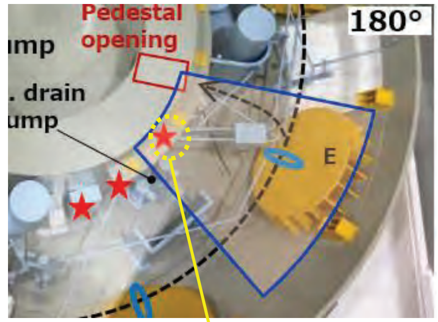
Source : International Research Institute for Nuclear Decommissioning (IRID)

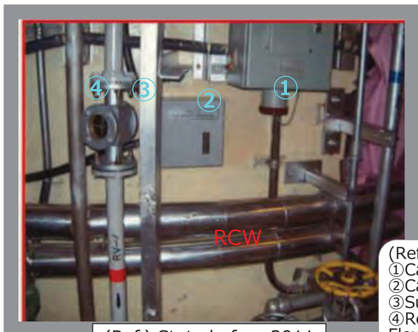
Pedestal concrete degradation below shelves

- Many metallic structures found to be relatively without deformations under the shelf
- RCW piping not found
- Rebar found without significant deformation

Low temperature concrete degradation?

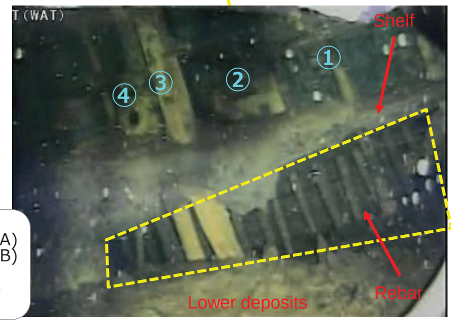
Long term effects (coolant flow, etc.)?





(Ref.) State before 2011

(Ref.)
①Cable relay box (A)
②Cable relay box (B)
③Support
④Reactor Vent line Flow glass



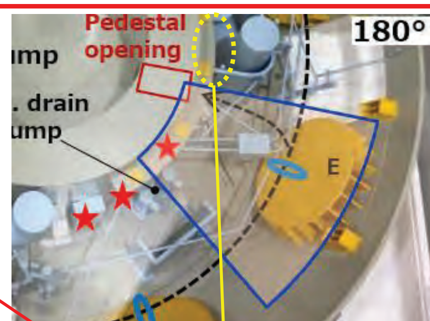
Source : International Research Institute for Nuclear Decommissioning (IRID)

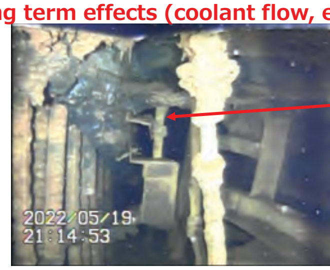
Pedestal concrete degradation below shelves

- Many metallic structures found to be relatively without deformations under the shelf
- Rebar found without significant deformation

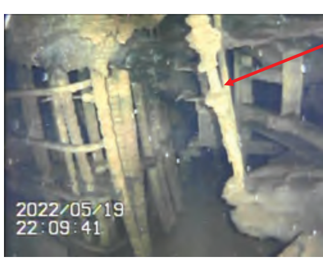
Low temperature concrete degradation?

Long term effects (coolant flow, etc.)?



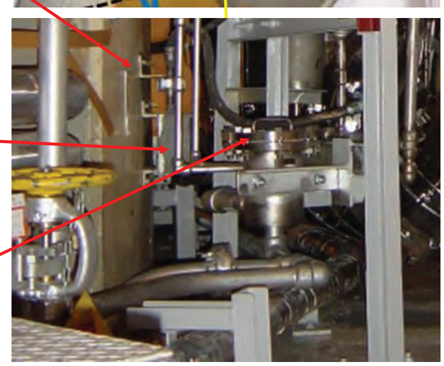


Support



Pipe

Handle



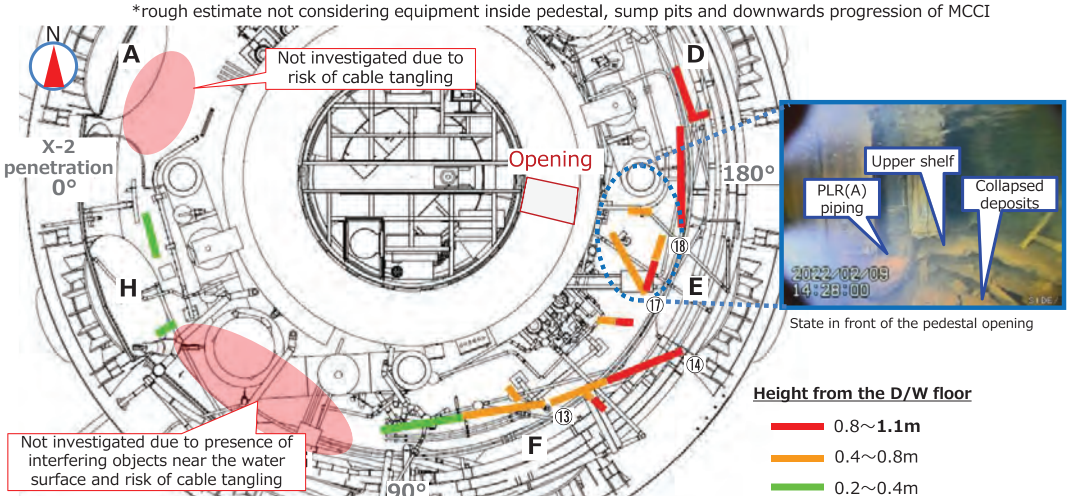
Source : International Research Institute for Nuclear Decommissioning (IRID)

Results of the deposits height distribution investigation

- Generally decreasing height with increase of the distance from the pedestal opening
- **Maximum** evaluated height of **~1.1 m** (**outside of pedestal**)

Contrast with estimate of core and reactor internals piled only in the pedestal: ~ 1.1 m*

*rough estimate not considering equipment inside pedestal, sump pits and downwards progression of MCCI



Measurement: Ultrasonic waves transmission/reflection while moving on the water surface

Evaluation: height of deposits from the D/W floor based on design data and water level of 2.0 m

Details (in Japanese) available at: https://www.tepco.co.jp/decommission/information/committee/roadmap_progress/pdf/2022/d220728_08-j.pdf#page=4

 Source : International Research Institute for Nuclear Decommissioning (IRID) 12

Distribution of deposits

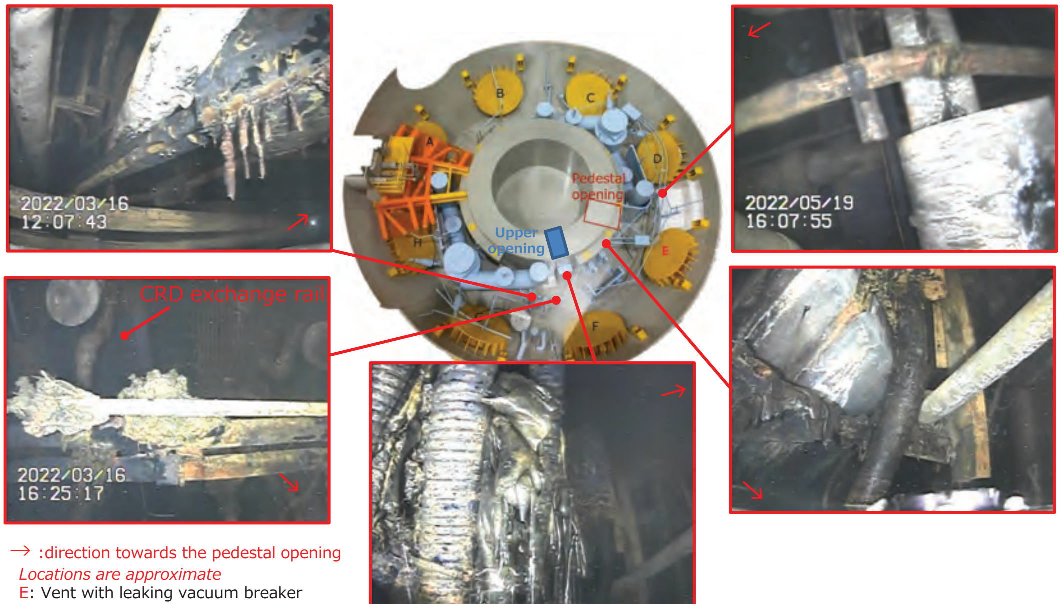
- Deposits height increasing with distance from the pedestal opening in some areas
- Could some of them come from upper floors?

 Source : International Research Institute for Nuclear Decommissioning (IRID) 13

Deposits above water level

IRID TEPCO

- Deposits hanging on pipes and supports
- Traces resembling solidified metal on RCW header and other piping above water level



→ :direction towards the pedestal opening
Locations are approximate
E: Vent with leaking vacuum breaker

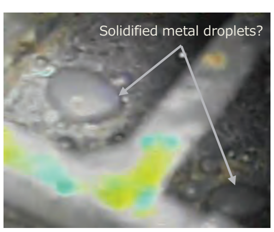
Source : International Research Institute for Nuclear Decommissioning (IRID)

Possible contributions to total amount of deposits from upper floors

IRID TEPCO

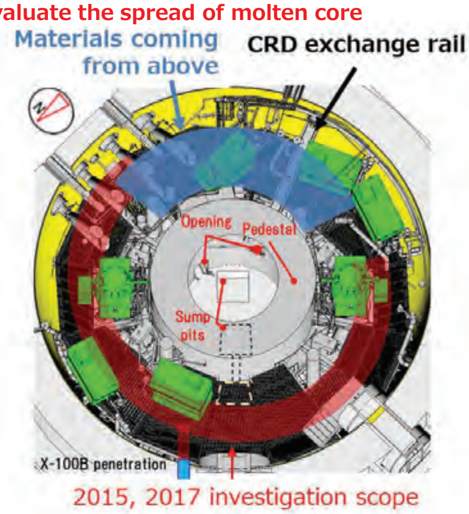
- Lead shielding and Zn galvanized grating from all D/W area (relatively small amount)
- Piping insulation materials? Paints?
- Materials coming from inside the pedestal area through the 1st floor CRD exchange rail opening?

Has to be considered to correctly distinguish and evaluate the spread of molten core

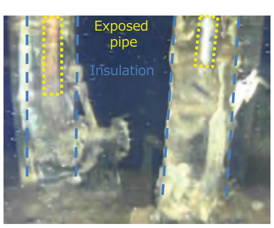


2015 investigation of 1st floor grating area

Gratings of Unit 4 of Fukushima Daini (experienced temperature ~150°C)



D/W 1st floor



RCW piping near equipment drain sump pump (under water)

RCW piping above water

Source : International Research Institute for Nuclear Decommissioning (IRID)

Degradation of thermal insulation

IRID TEPCO

- Degradation of RCW piping insulation observed in multiple locations above and under water
- Urethane insulation missing from some exposed pipes
- "Glittery" deposits near connection lines

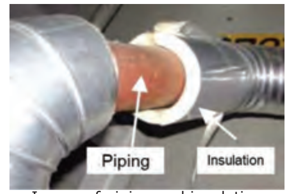
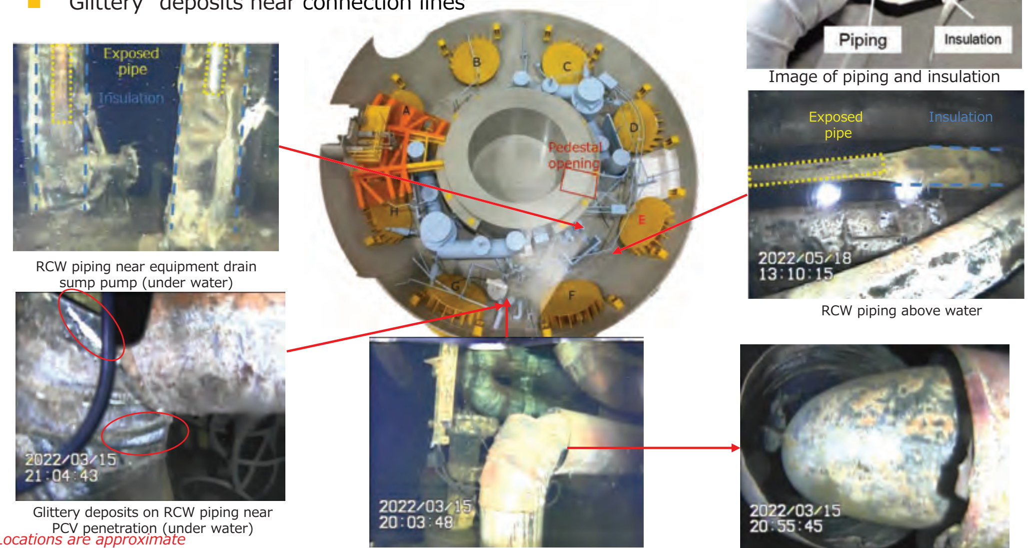


Image of piping and insulation



RCW piping near equipment drain sump pump (under water)

RCW piping above water

Glittery deposits on RCW piping near PCV penetration (under water)

RCW piping near PCV penetration (under water)

Locations are approximate

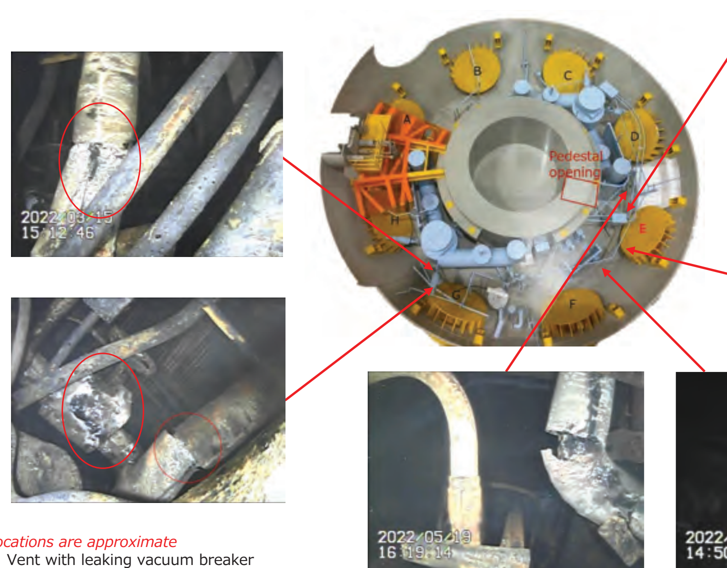
E: Vent with leaking vacuum breaker

Source : International Research Institute for Nuclear Decommissioning (IRID)

State of piping above water level

IRID TEPCO

- "Glittery" metal-like deposits found in higher amounts near pedestal opening and CRD exchange rail (mainly on RCW header piping)
- Peeled/deformed outer cladding in several locations



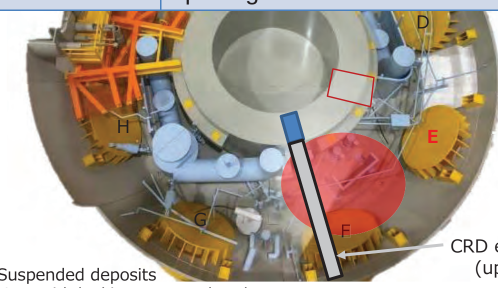
Locations are approximate

E: Vent with leaking vacuum breaker

Source : International Research Institute for Nuclear Decommissioning (IRID)

Suspended deposits

Appearance	Diverse
Size	Diverse
Location	Near eq. drain sump pump, under CRD exchange rail, on top of other deposits or hanging on structures below/under water level
Origin/ composition	High temperature degradation of insulation and shielding materials? Possibly relocation from pedestal area via CRD exchange opening? High temperature near CRD exchange opening?



CRD exchange rail (upper floor)

■: Suspended deposits
E: Vent with leaking vacuum breaker

Source : International Research Institute for Nuclear Decommissioning (IRID)

Suspended deposits

- Deposits found suspended on structures above the water level or below the water level in locations higher than the D/W floor deposits
- Found in high amounts in the vicinity of the CRD exchange rail and pedestal opening on the above floor
- Assumed to be distributed on top of D/W floor deposits in some areas

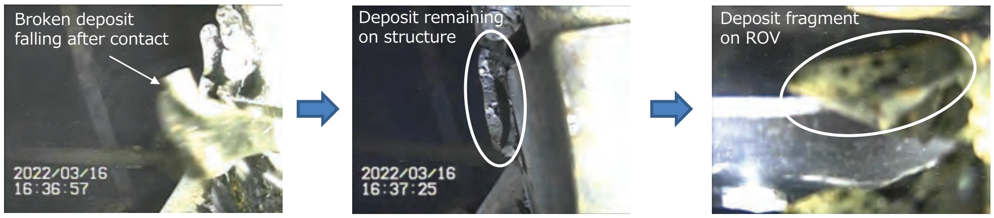
Deposits suspended on piping above water level

Deposits suspended on structures in water (shielding of equipment drain sump pump)

Source : International Research Institute for Nuclear Decommissioning (IRID)

Suspended deposits

- Confirmed fracturing of deposits after unintentional contacts with ROV (both under/above water)
- Porosity could be seen in both cases



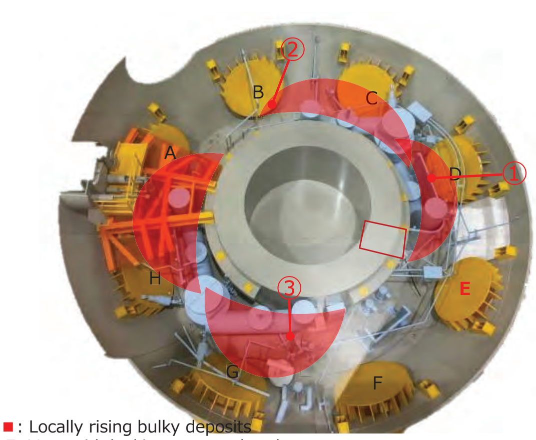
After contact with ROV, fragment of suspended deposit above water level landed on top of the ROV. After that, the deposit could not be directly seen by camera.
After ROV submerged, reflection of ROV's top surface including the fragment could be seen on the water level.

After contact with ROV, fragment of suspended deposit under water level broke away
A gas bubble emerging from the broken deposit and raising towards the water surface was observed.

Source : International Research Institute for Nuclear Decommissioning (IRID)

Locally rising bulky deposits

Appearance	Smooth bulky piles, irregular clusters
Size	Tens of cm thick, several m wide
Location	In the vicinity of PLR piping
Origin/ composition	Melted/relocated from lead wool blanket



■: Locally rising bulky deposits
E: Vent with leaking vacuum breaker

Source : International Research Institute for Nuclear Decommissioning (IRID)

Locally rising bulky deposits

- Found mainly in the vicinity of PLR piping
- Locally increasing in height even with increasing distance from the pedestal opening
- Considered to be coming mainly from the lead wool blankets shielding
- Lead wool blanket connecting pieces often found in the vicinity or on top of these deposits
- Measured temperature inside PCV on 3/20: ~400°C*
  **close to the maximum scale of the instruments*
- Lead melting point: 327.5°C
- Local height maximums near the support structures

Source : International Research Institute for Nuclear Decommissioning (IRID)

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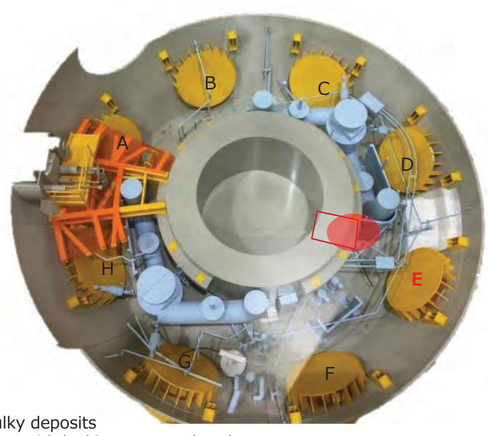
Bulky deposits

Appearance	Smooth, irregular shape
Size	Tens of cm
Location	Inside/in front of pedestal opening
Origin/ composition	Core debris material? Materials from lead wool blankets?



■: Bulky deposits
E: Vent with leaking vacuum breaker

Source : International Research Institute for Nuclear Decommissioning (IRID)

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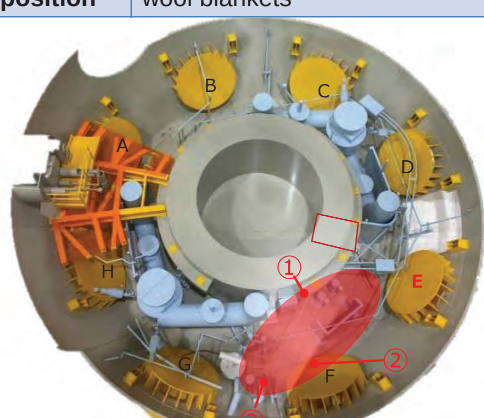
Bulky deposits

- Found loosely distributed inside the pedestal opening (deposits height minimum)
- Location overlaps with the expected distribution of relocated lead materials
- Difficult to distinguish whether these deposits were relocated from inside or outside of the pedestal
- Appearance (size, smooth surface, etc.) resembles deposits in the vicinity of PLR piping
- Future investigation inside the pedestal area should provide more information about their origin

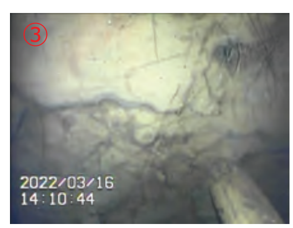
Source : International Research Institute for Nuclear Decommissioning (IRID)

Thread-like deposits

Appearance	Thin thread-like objects on top of or mixed with other deposits
Size	Few cm ~ few m long
Location	Between pedestal wall, jet deflectors F and E
Origin/ composition	Likely originating from upper floors and/or lead wool blankets



■ : Fibrous deposits
E: Vent with leaking vacuum breaker



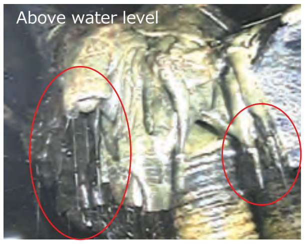
Source : International Research Institute for Nuclear Decommissioning (IRID)

Thread-like deposits

- Found in the same locations as suspended deposits
- Possibly originating from fibrous materials used in lead wool blankets etc.
- Or formed from streams of molten materials relocating from upper levels

Damaged lead wool blanket with exposed fibers

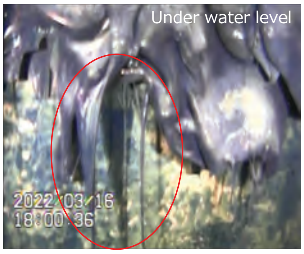
Thread-like deposits above water level



Solidified streams above water level

Unknown mesh object on top of deposits

Thread-like deposits above water level

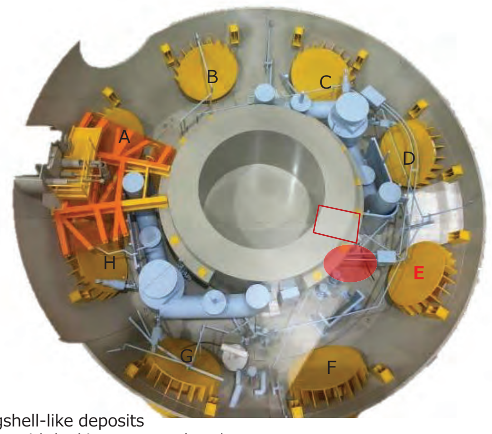


Solidified streams under water level

Source : International Research Institute for Nuclear Decommissioning (IRID)

Eggshell-like deposits

Appearance	Hollow (semi)spherical thin crust
Size	Few cm
Location	In front of pedestal opening
Origin/ composition	Materials relocated from above?



■ : Eggshell-like deposits
E: Vent with leaking vacuum breaker

Source : International Research Institute for Nuclear Decommissioning (IRID)

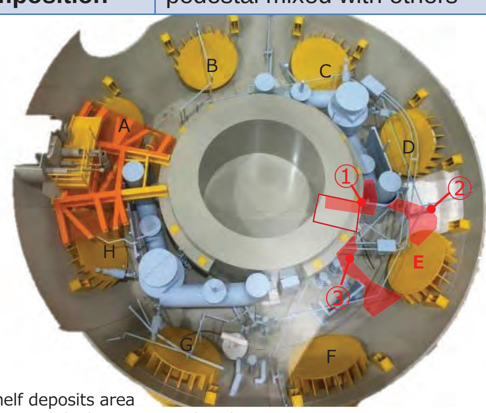
Eggshell-like deposits

- Found mainly near the pedestal opening
- Similar thin crust objects found in other areas too (mainly overlapping with other materials coming from upper floors), but not forming egg-like shapes

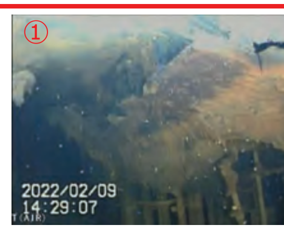
Source : International Research Institute for Nuclear Decommissioning (IRID)

Shelf deposits

Appearance	Flat planks attached to structures
Size	Thickness varying from few cm to several tens of cm, length of few m
Location	Pedestal opening/wall, PCV shell – structures at height of ~1 m and above
Origin/ composition	Unknown, likely materials coming from pedestal mixed with others



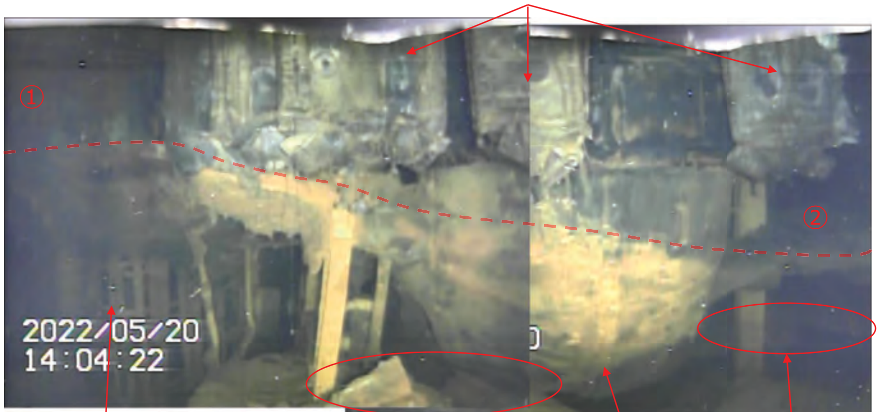
■: Shelf deposits area
E: Vent with leaking vacuum breaker



Source : International Research Institute for Nuclear Decommissioning (IRID)

Shelf deposits

- ①: 1 layer of shelves decreasing in height with distance from the pedestal opening
- thickness of shelf increasing with distance from the pedestal opening
- existence of lower shelf in/near the pedestal opening unconfirmed
- discoloration on PLR piping suggest continuity of shelf between ① and ② (before collapse)



Source : International Research Institute for Nuclear Decommissioning (IRID)

Shelf deposits

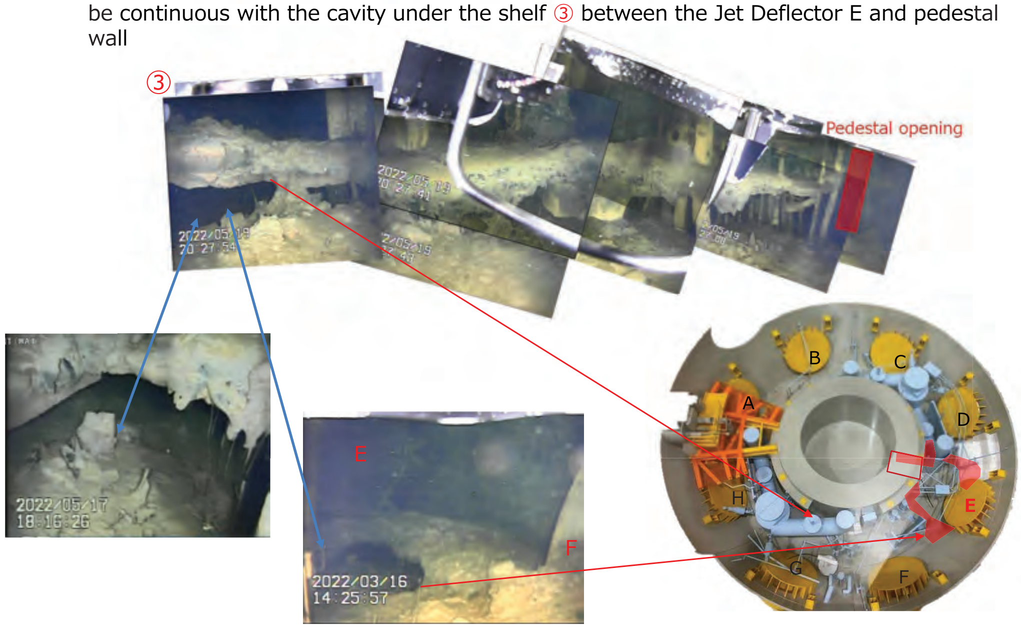
- ②: 2 shelves coupled near liner (no visible damage to structures in between)
  Lower shelf horizontally flat, inner surfaces smooth
- ③: only one shelf (relatively thick) covered with deposits relocated from upper floors
- Collapsed shelves found mainly near ②

Is the difference in appearance of ② and ③ and distribution of collapsed shelves caused by different formation mechanisms or effect of mixing with other materials?

Source : International Research Institute for Nuclear Decommissioning (IRID)

Shelf deposits

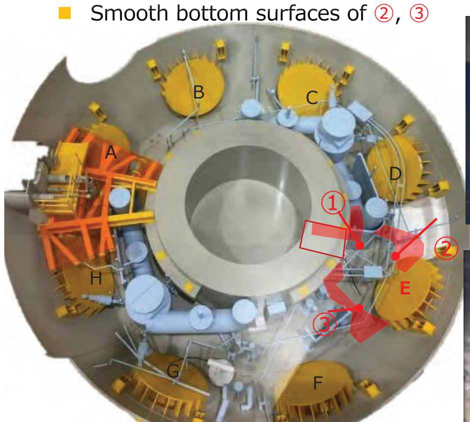
- Cavity between the Jet Deflector F and shielding of equipment drain sump pump seems to be continuous with the cavity under the shelf ③ between the Jet Deflector E and pedestal wall



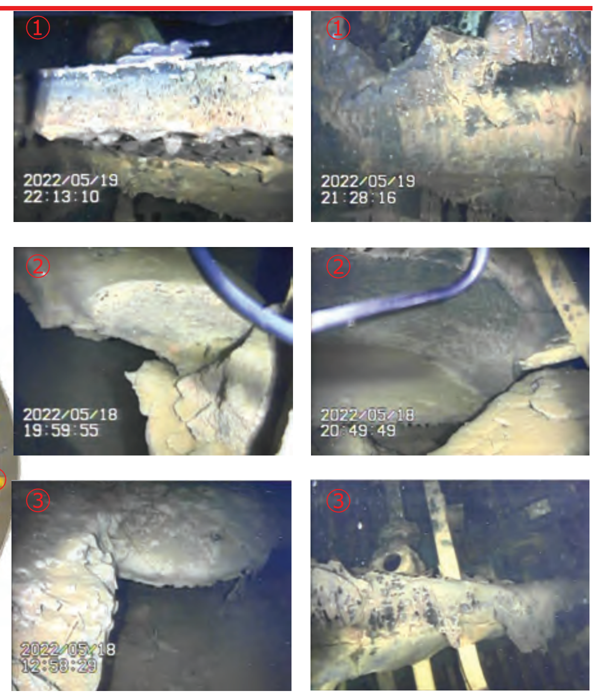
Source : International Research Institute for Nuclear Decommissioning (IRID)

Shelf deposits

- The porosity seen on the edges (crack surfaces) of ① and ② is similar with ② being thicker
- Clear edges could not be confirmed for ③ due to likely piling of other materials
- "lavacicles" on bottom surface of ①
- Smooth bottom surfaces of ②, ③



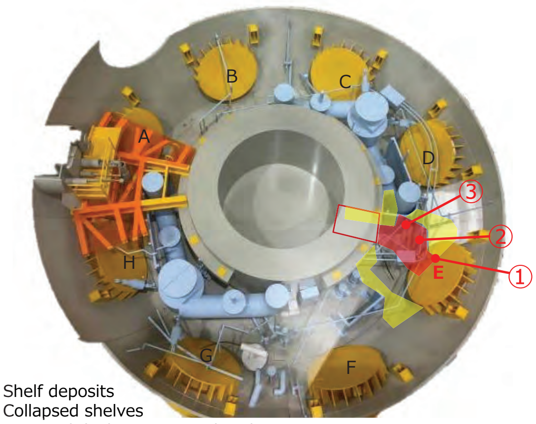
■: Shelf deposits
E: Vent with leaking vacuum breaker



Source : International Research Institute for Nuclear Decommissioning (IRID)

Collapsed shelves

Appearance	Various shapes with outer edges matching the shape of other deposits, shelves or structures
Size	Tens of cm
Location	In the vicinity of shelves attached to structures
Origin/ composition	Broken away from the shelf deposits Composition of shelves unknown

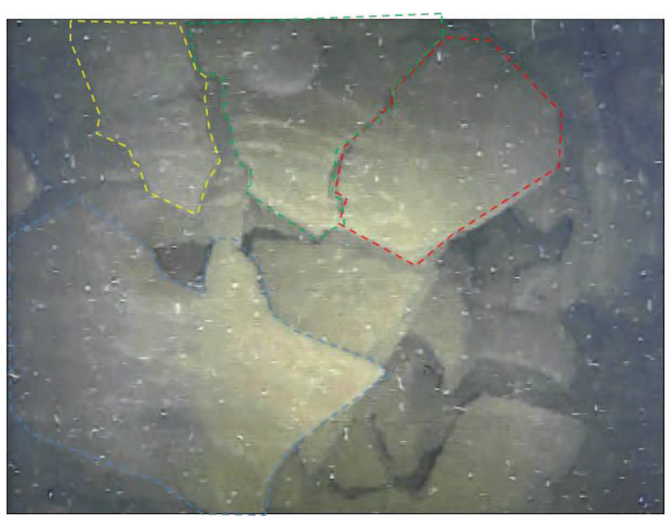


■ : Shelf deposits
■ : Collapsed shelves
E: Vent with leaking vacuum breaker

Source : International Research Institute for Nuclear Decommissioning (IRID)

Collapsed shelves

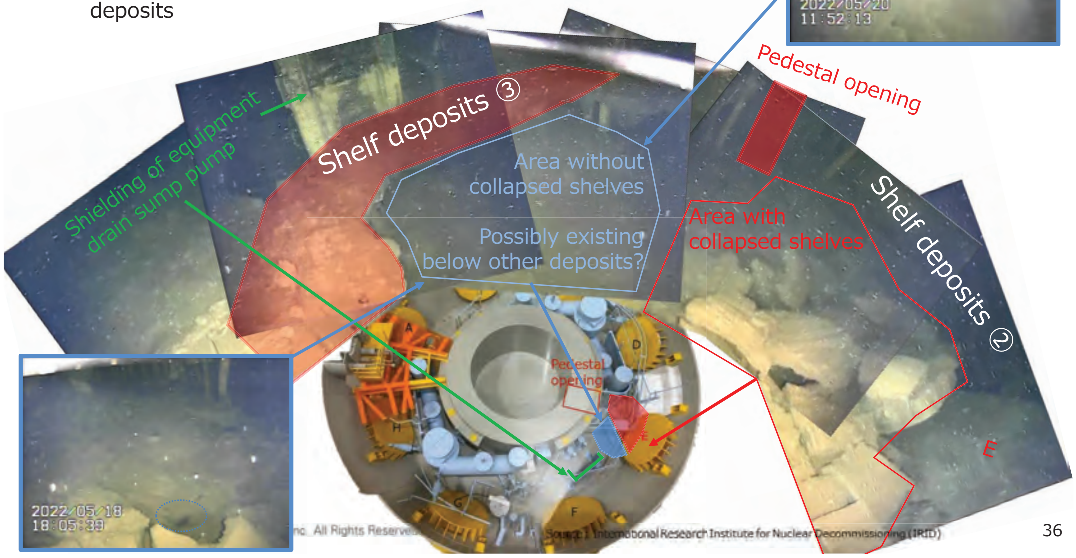
- Shapes of neighboring deposits matching each other in a puzzle-like manner



Source : International Research Institute for Nuclear Decommissioning (IRID)

Collapsed shelves

- Collapsed shelves between Jet Deflector E and PLR piping
- Area between pedestal opening, shielding of equipment drain sump and Jet Deflector E is assumed to contain relatively big amount deposits relocated from above floors
- Collapsed shelves might be existing under different kind of deposits

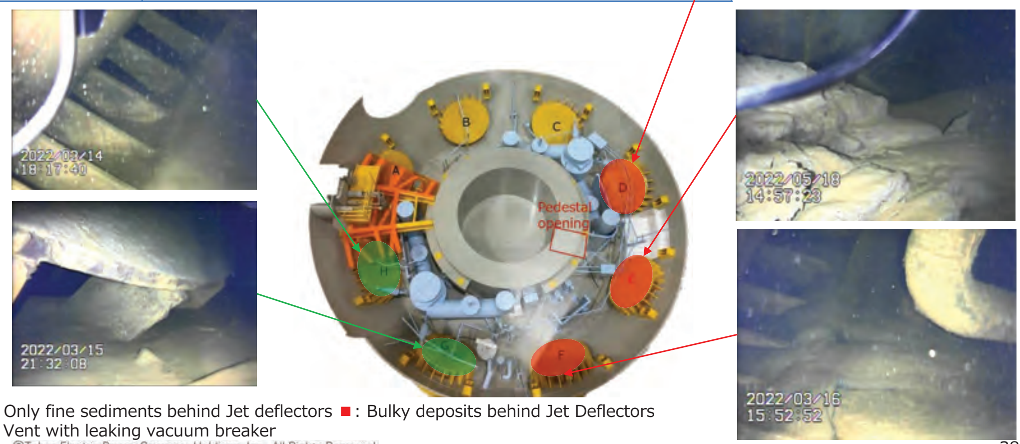


Collapsed shelves

- Porosity seen on the edges (crack surfaces) of collapsed shelves similar to porosity of other kinds of deposits

Deposits behind jet deflectors

Appearance	Continuous solidified streams
Size	Tens of cm
Location	Behind Jet Deflectors D, E, F
Origin/ composition	Same as shelf deposits Possibility of some contribution of materials relocated from upper floors

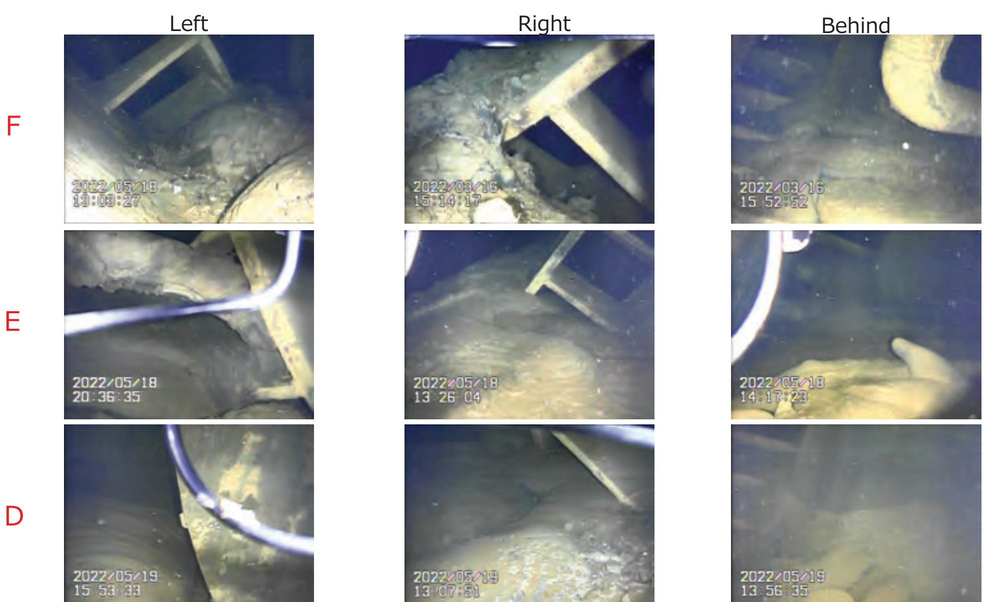


■: Only fine sediments behind Jet deflectors ■: Bulky deposits behind Jet Deflectors
E: Vent with leaking vacuum breaker

Source : International Research Institute for Nuclear Decommissioning (IRID)

Deposits behind jet deflectors

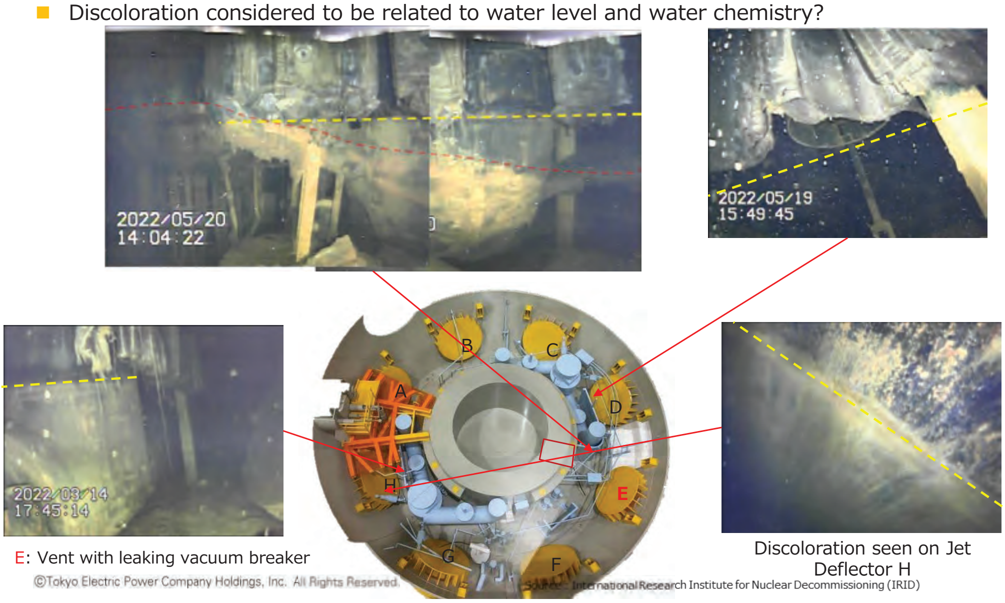
- Bulky deposits flowing towards the S/C predominantly through the bottom openings



Source : International Research Institute for Nuclear Decommissioning (IRID)

Height of failure of lead wool blankets

- Regardless of the height of deposits, the lead wool blankets are torn in the same height
- The height of failure (~1.2 m) is equal to the height of discoloration on shell, jet deflectors and structures in all areas of D/W
- Discoloration considered to be related to water level and water chemistry?



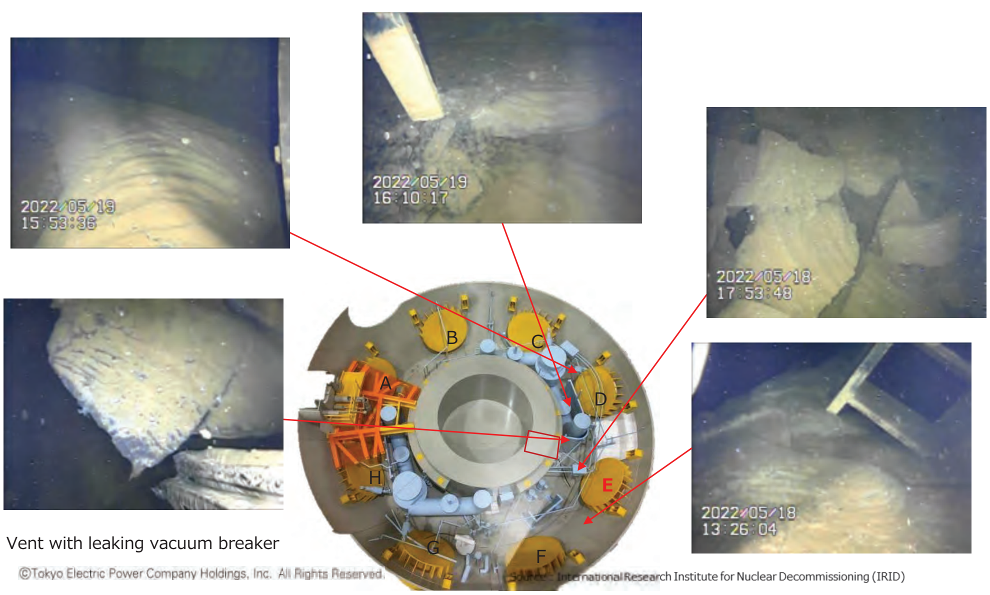
E: Vent with leaking vacuum breaker

Discoloration seen on Jet Deflector H

Source: International Research Institute for Nuclear Decommissioning (IRID)

Ripple-like patterns on surface of deposits

- Can provide information on flow, viscosity, etc.?
- Could indicate relatively fast solidification?



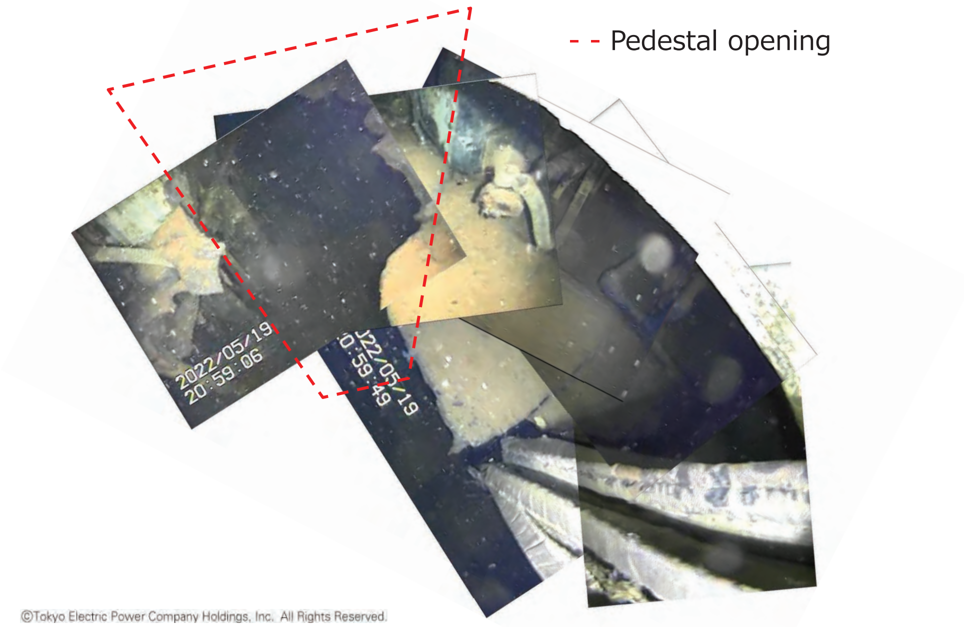
E: Vent with leaking vacuum breaker

Source: International Research Institute for Nuclear Decommissioning (IRID)

Flow of materials from the pedestal opening

IRID TEPCO

- Shape/gradient of deposits in front of pedestal opening suggests big amount of materials being relocated (pushed out) from the pedestal opening



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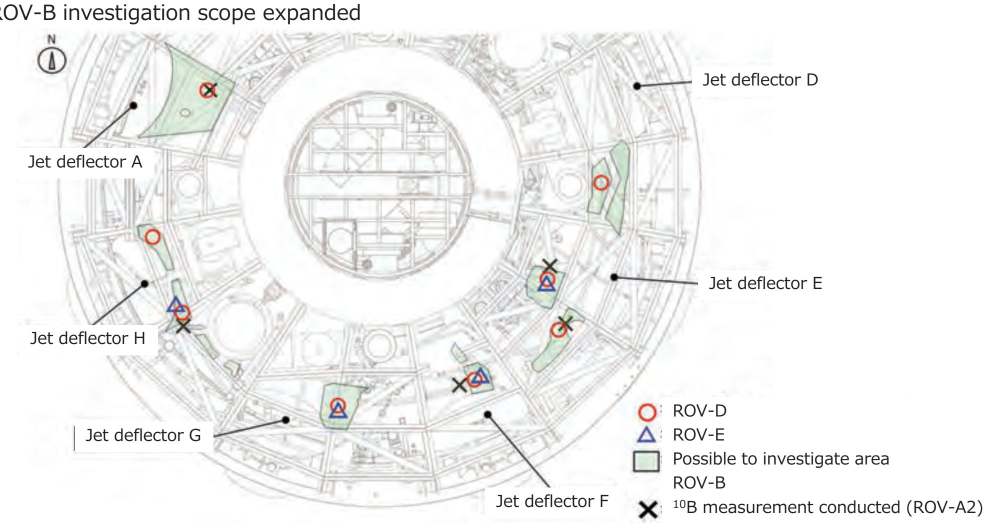
Investigation scopes of ROV-D, E, B

IRID TEPCO

- Areas where ROV can float on water surface and suspend/seat instruments on top of deposits
- ROV-D to firstly collect γ-spectra in points where ROV-A2 previously detected neutrons
- ROV-E to evenly collect samples from the outer circumference of the D/W

(originally planned to be selected based on ROV-D investigation results, but since it is assumed that the surface of deposits is covered by fine non-debris sediments, the plan has been revised)

- ROV-B investigation scope expanded



Source : International Research Institute for Nuclear Decommissioning (IRID)

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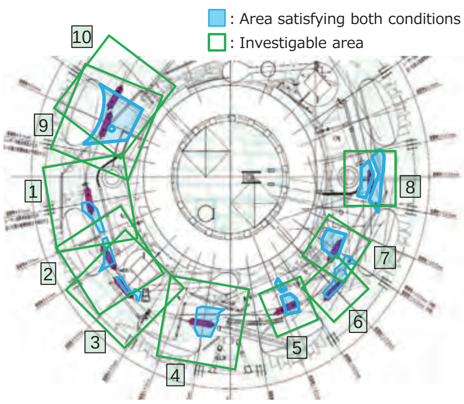
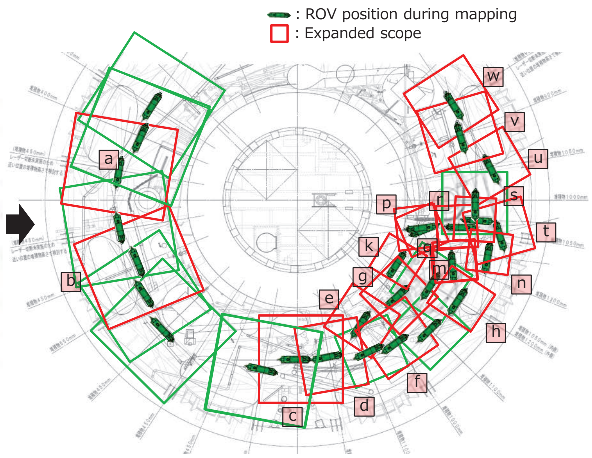
Scope of 3D mapping of deposits (ROV-B)

- Conditions required for 3D mapping
  ①ROV can float on the water surface
  ②Anchor can be seated on top of the deposits
- Areas not satisfying the condition ② can be investigated without seating the anchor, by fixing the ROV on D/W structures ⇒ investigation scope can be expanded (feasibility verified by M/U)

However, the scope might be reduced as a result of cable tangling and effects of water flow



Before M/U

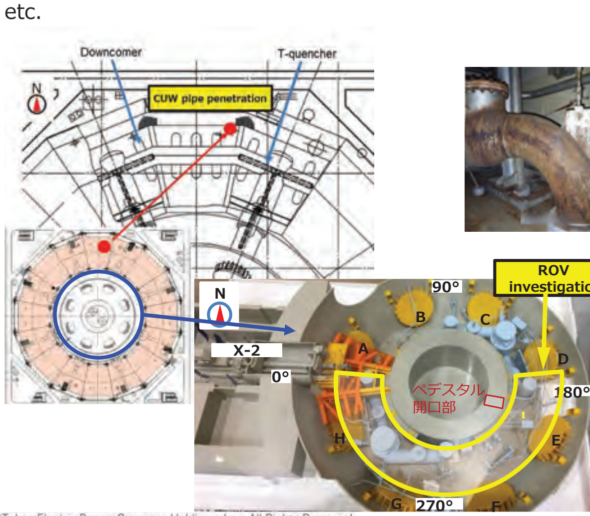


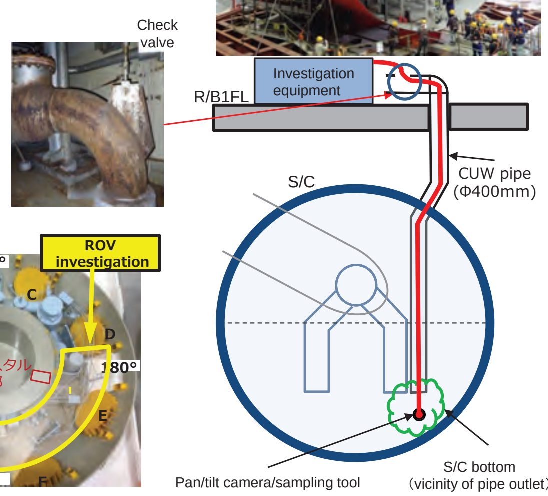
Investigation scope after M/U

Source : International Research Institute for Nuclear Decommissioning (IRID)

S/C investigation plan (January 2023)

- Plan to decrease PCV water level (earthquake resistance improvement)
- S/C water sampling and visual inspection through CUW line
- Possibility to confirm the state of S/C bottom (presence of deposits), inner surface, lower part of downcomer, T-quencher, etc.





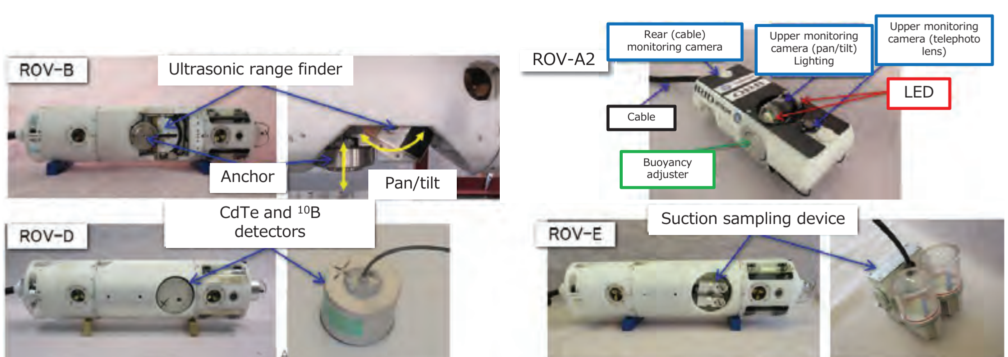
IRID TEPCO

Thank you for attention

IRID TEPCO

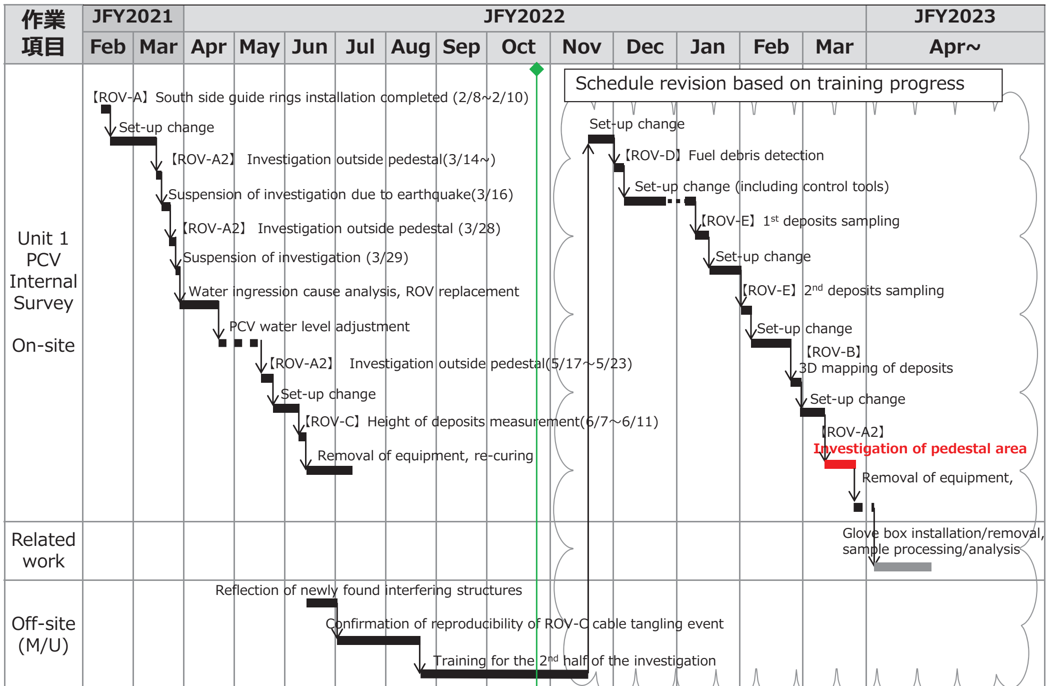
Future investigations plan (investigation items/routes)

Order	Investigation tool	Instruments	Objective
①	**ROV-D** Debris detection	· CdTe semiconductor detector · improved compact ^{10}B detector	Debris detection based on neutron flux, ^{137}Cs, ^{154}Eu measurement (suspending the sensor on the deposits surface)
②	**ROV-E** Sediment sampling	· suction type sampling device	Collection of sediments from the deposits surface
③	**ROV-B** 3D mapping	· ultrasonic range finder (scanner) · thermometer	Confirming of the deposits height distribution in wide range
④	**ROV-A2** Visual inspection	· optical fiber γ dosimeter · improved compact ^{10}B detector	Detailed inspection of pedestal area, inner and outer wall



Source : International Research Institute for Nuclear Decommissioning (IRID)

Overall schedule (as of October 27)



(Note) Time of execution of planned work might change according to on-site work progress

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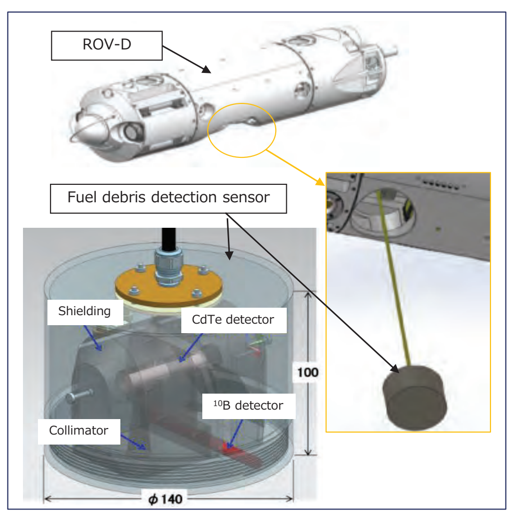
ROV-D (debris detection) γ-spectroscopy

- 4 dominant γ-emitting nuclides in fuel debris (^{154}Eu, ^{137}Cs, ^{60}Co, ^{125}Sb)
- Criteria for fuel debris detection: Combination of ^{154}Eu detection and neutron flux measurement

- ^{154}Eu
  - Low-volatile fission product
  - Accompanying behavior with U
  - Emitted γ-rays relatively easily detectable
  - Selected for detection of fuel materials
- ^{137}Cs
  - High volatility
  - Difficult to use
- ^{60}Co
  - Not FP, but neutron activation product
  - Difficult to use
- ^{125}Sb
  - High volatility
  - Difficult to use

Detectability of fuel materials using γ-emitting nuclides



ROV-D equipment configuration

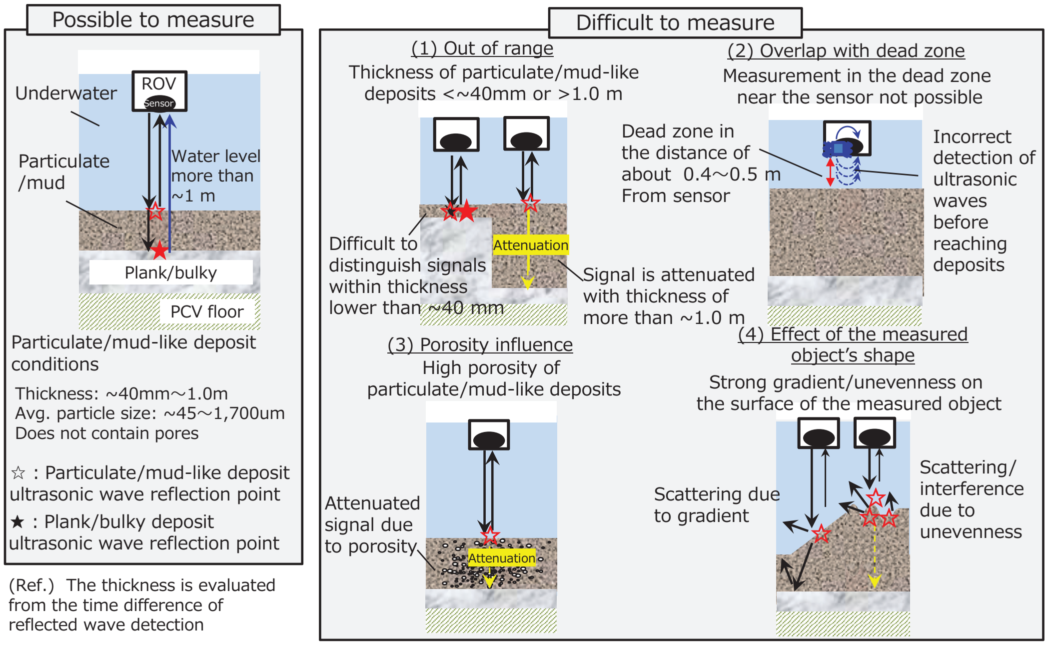
Source : International Research Institute for Nuclear Decommissioning (IRID)

48

(Ref.) Difficulties in measuring particulate/mud-like deposits

IRID TEPCO

- Conditions on the left are a prerequisite for correct measurement of deposit thickness



(Ref.) The thickness is evaluated from the time difference of reflected wave detection

Source : International Research Institute for Nuclear Decommissioning (IRID)

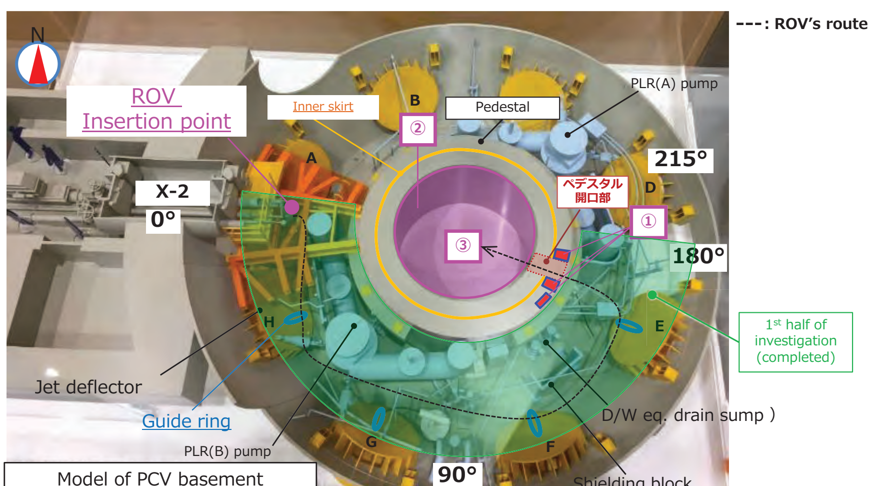
49

ROV-A2 investigation scope (2nd half)

IRID TEPCO

- Detailed inspections of the pedestal walls and area inside the pedestal
  ①Outer wall damage (scope of rebar exposure/concrete damage width/height/spread)
  ②Inner wall damage (scope of rebar exposure/concrete damage width/height/spread)
  ③State inside the pedestal (upper structures, deposits inspection, dose rate measurement, etc.)

No information on the state inside pedestal ⇒ high possibility of ROV becoming unable to return



Source : International Research Institute for Nuclear Decommissioning (IRID)

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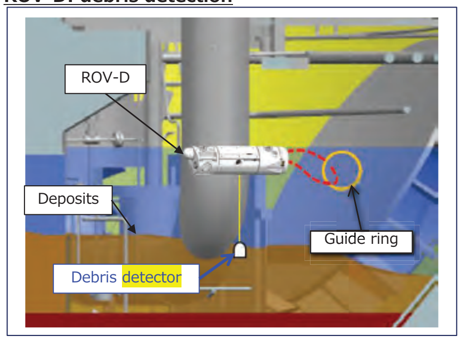
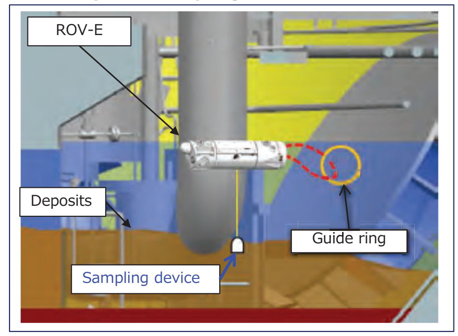
Overall schedule (as of October 27)

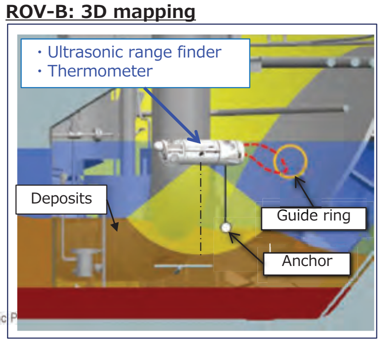
ROV-D: debris detection

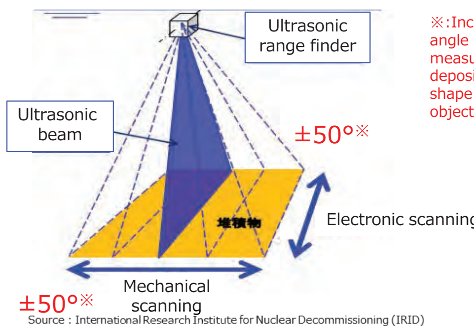


ROV-E: deposits sampling



ROV-B: 3D mapping





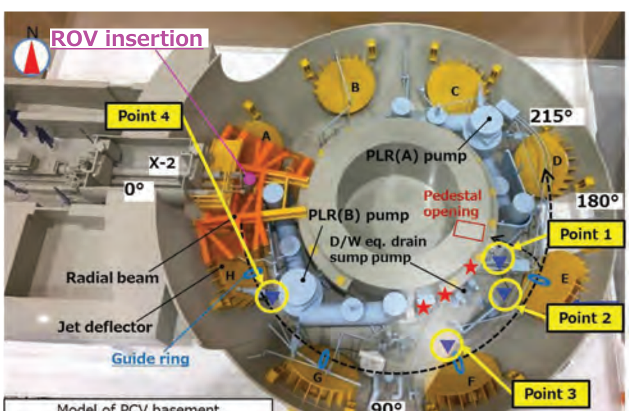
※:Incident angle allowing measurement of deposits surface shape and fallen objects height

Source : International Research Institute for Nuclear Decommissioning (IRID)

Investigation results (A-2)

20, 21 May: Neutron flux measurement results

- Thermal neutron flux confirmed at all 4 measurement points
- Higher flux values detected near the pedestal opening
- Subsequently, height and thickness (ROV-C) and fuel debris presence (ROV-D) will be further investigated

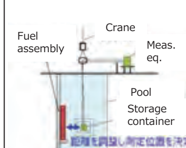


	Point 1	Point 2	Point 3	Point 4
Thermal neutron flux [／cm2／s]	48.0	29.1	50.2	5.8

- 60 min/measurement point
- Flux derived from integrated neutron count

<Ref> SF assembly measurement results by the B-10 detector of A-2 @NFD

■measurement method
・set to center of fuel assembly
・measurement in 3 points
・measurement time: 3 min



Dose rate	Distance	Therm. n. flux※
14.4 Gy/h	約16 cm	8.8×10^{1} /cm²/s
6.5 Gy/h	約33 cm	1.1×10^{1} /cm²/s
1.5 Gy/h	約78 cm	0 /cm²/s

Source : International Research Institute for Nuclear Decommissioning (IRID)

C.2.3.2. Summary Recent Investigation Findings

Summary of recent investigation results in Fukushima Daiichi Nuclear Power Station

TEPCO

November 17, 2022
Kenji OWADA
Tokyo Electric Power Company Holdings, Inc.

Outline

TEPCO

- Efforts in FY2021
  - ✓ Units 1/2 R/B upper floors investigation : Completed
  - ✓ Units 1/2 bottom part of exhaust stack removal (and SGTS piping removal) : Ongoing (delaying original plan)
  - ✓ Unit 1 PCV internal investigation : Ongoing
  - ✓ Unit 2 shield plug perforations investigation : Completed
  - ✓ Unit 2 R/B stagnant water drawdown : Completed
  - ✓ Unit3 MSIV room stagnant water analysis : sampling completed, analysis ongoing
  - ✓ **Unit 1~3 remaining gas examination and investigation : Ongoing**
  - ✓ Unit 1~4 SGTS rooms investigation and filters radionuclide analysis : investigation completed
- Efforts in FY2022
  - ✓ **Unit 2 FHM remote control room investigation : Completed**

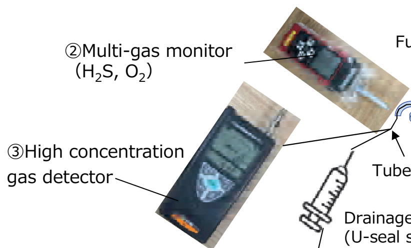
1

Unit 1~3 remaining gas examination and investigation

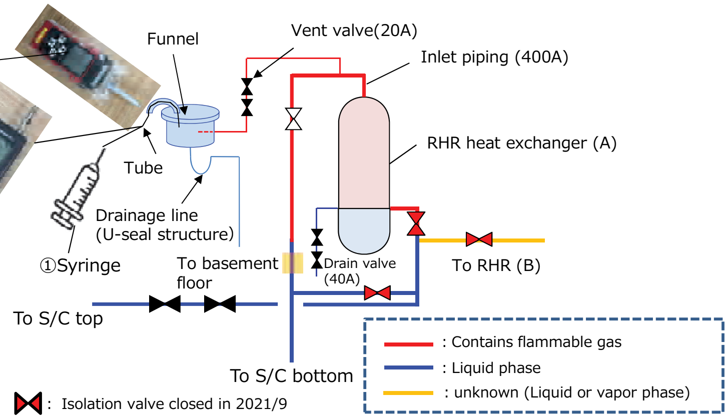
TEPCO

■ Outline

- Installation of the water withdrawal system underway (Unit 3 PCV water level decrease)
- Presence of pressurized remaining gas confirmed in the RHR※ heat exchanger (A) after vent valve opening (preparatory work)   ※Residual Heat Removal System
- Presence of Kr-85, hydrogen, etc. (confirmed by gas sampling and analysis), presumed to be result of gas inflow from the PCV and entrapment during the accident
- Residual gas purge with N_2 completed, work related to the system installation continuing



①Collection and analysis by syringe
・Kr-85 : 2.64 × 10^3 Bq/cm^3
②Measurement by multi-gas monitor
・O_2 concentration : 0%
・H_2S concentration : Approx. 20ppm
③ Measurement by high concentration gas detector
・H_2 concentration increased to approx. 20% and stabilized



Schematic image of RHR piping system

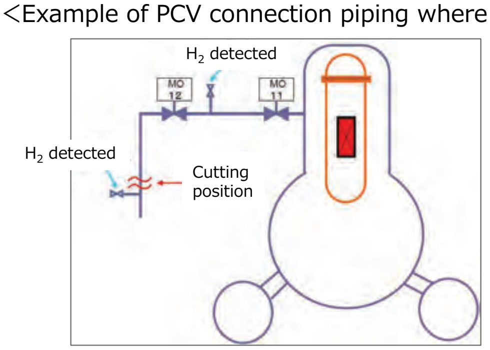
Unit 1~3 remaining gas examination and investigation

TEPCO

■ H_2 gas-related actions up to now

- Filling PCV with N_2 gas after the accident (most of the H_2 generated during the accident assumed to be already diffused into the atmosphere)
- Careful execution of (the previous) D&D work under presumption of H_2 remaining in the facilities, implementation of measures such as N_2 purging for facilities where stagnant H_2 previously confirmed

<Example of PCV connection piping where H_2 stagnation were confirmed>



H_2 stagnation in PCV spray system piping in Unit 1
(PCV gas management equipment installation, in 2011)

Time	Location of confirmed residual H_2※
2011/9	Unit 1 PCV spray system piping
2011/10	Unit 2 flammable gas concentration control system piping
2012～2013	Unit 1/2 S/C
2021/12	Unit 3 RHR piping

※ All have been purged with N_2.

Unit 1~3 remaining gas examination and investigation

TEPCO

■ Narrowing down of the locations where H_2 gas may be remained

	System	Justification
Risk exists	Unit 1 IC (A)	Possibility of isolation valves being open after core damage (based on results of post-accident isolation valves conditions investigation inside and outside the PCV, etc.) Possibility of condensed steam not fully returning to RPV, H_2 being sealed in the heat transfer tubes, etc.
	Unit 3 PCV spray piping	Remaining H_2 confirmed in RHR(A) ⇒ high possibility of H_2 entrapment in RHR(B)
Difficult to determine due to lack of information (conservatively assumed that risk exists)	Units 1~3 System used for fire truck water injection	Fire truck water injection possibly at low pressure/flow rate due to bypass flow to other systems, possibility of H_2 entering the system cannot be ruled out. Entire history of valve operation difficult to ascertain from the records, risk conservatively evaluated as being present
	Unit 1 RCW (DHC)	Entire system highly contaminated, assumed to be due to water or gas inflow from the D/W at the time of the accident as a result of damage to equipment in the D/W. H_2 may have been remained due to water sealing at each load.
	Units 1~3 CRD (HCU)	Located at the bottom of the reactor core, possibility of hydrogen inflow into the HCU (Hydraulic Control Unit) system if hydrogen inflow into the CRD system occurred. Possibility of the HCU being subject to inflow of cooling water after the accident and entrapment of H_2 within the system due to water sealing.

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Unit 1~3 remaining gas examination and investigation

TEPCO

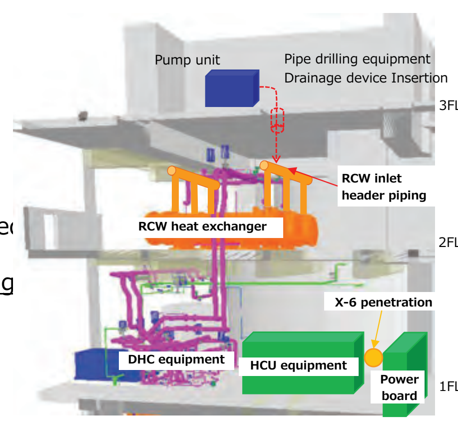
■ Future efforts

- On-site investigations and development of a work plan for the extracted systems with potential of H_2 entrapment. On-site investigations planned in consideration of work safety, and ALARA principles.
- Investigation period : under consideration

■ Related investigation item : TRB-11

■ Related Recent topics :

- **<u>Unit 1 RCW-Hx water sampling</u>**
  - The inner water sampling from highly contaminated RCW of Unit 1, was planned for 2022/1~3 as a dose reduction measure in R/B.
  - Suspended due to partial overlap of the work area with the PCV internal investigation, resumed in 2022/10.
  - <u>Investigation of residual gas in the header piping at the heat exchanger inlet started on 10/24.</u>



Unit 1 R/B 1-3F south side Cross section

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Unit 2 FHM remote control room investigation

TEPCO

■ Objective

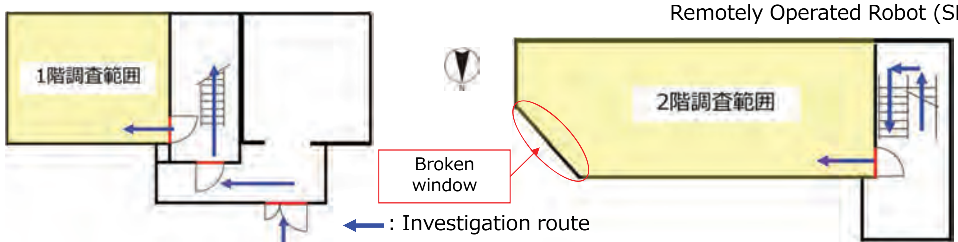
- High levels of contamination in Fuel Handling Machine remote control room (FHM room) and its roof, and a broken window on the second floor previously confirmed
- Not accessed since the accident, located near the shield plug (estimated to be the main release route of radioactive materials)
- Acquisition of information on the radioactive materials released at the time of the accident prior to FHM room dismantling.

■ Outline

- Use of remotely operated robot (SPOT) for acquisition of video images, dose rates, and smear sampling in the room.
- Investigation period : 2022/6~9

■ Related investigation item : RB-4

Remotely Operated Robot (SPOT)



Schematic drawing of the FHM room 1st floor

Schematic drawing of the FHM room 2nd floor

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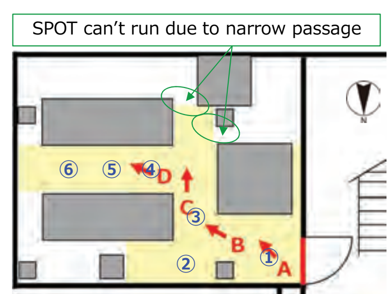
Unit 2 FHM remote control room investigation

TEPCO

■ State of the first floor

- no major damage to walls, ceiling, floor, or equipment.
- Maximum dose rate: 28.0 mSv/h (point ①).



1st floor

■ : SPOT path

①~⑥ : Survey point (measured on 8/1)

A~D : Image capture point (captured on 7/25)

point	γ dose rate [mSv/h]	
	1500mm from the floor	50mm from the floor
①	14.2	28.0
②	14.4	23.2
③	13.1	16.1
④	12.5	15.3
⑤	13.2	15.3
⑥	15.9	21.7

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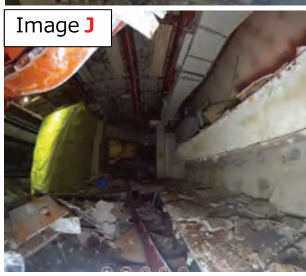
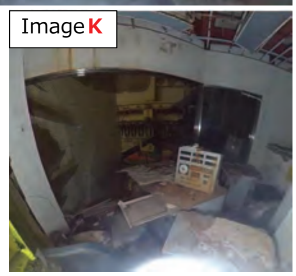
Unit 2 FHM remote control room investigation

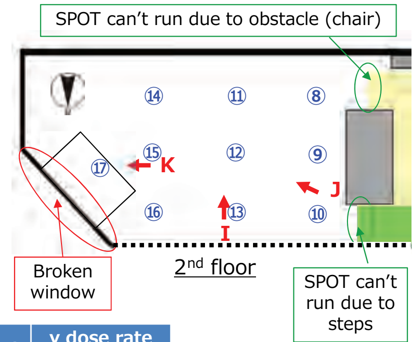
TEPCO

■ State of the second floor

- Gypsum boards fallen from the ceiling
- OA floor plates came off
- Maximum dose rate: 76.1 mSv/h (point ⑮).







Point	γ dose rate [mSv/h]
⑧	50.5※1
⑨	58.8※1
⑩	50.2※1
⑪	57.3※1
⑫	75.2※1
⑬	60.1※1
⑭	66.8※1
⑮	76.1※1
⑯	73.8※1
⑰	53.2※2

■ : SPOT path
■ : OA floor plate came off under SPOT's weight
⑧～⑰ : Survey point (measured on 8/29)
I～K : Image capture point (captured on 8/29)

※1 : Measured at a height of approx. 500 mm from OA floor
※2 : Measured at the height of the operation console (approx. 1300 mm from the OA floor)

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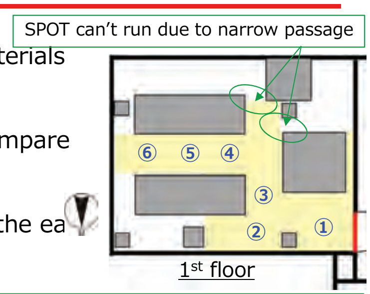
Unit 2 FHM remote control room investigation

TEPCO

■ Dose rate survey results

- Assumed inflow of gases/aerosols containing radioactive materials through the broken window on the 2nd floor
  - Dose rates on the 2nd floor higher than on the 1st floor (compare points ① ~ ⑥ and ⑦)
  - Increasing trend with approaching the broken window on the ea side (points ⑧ to ⑯)



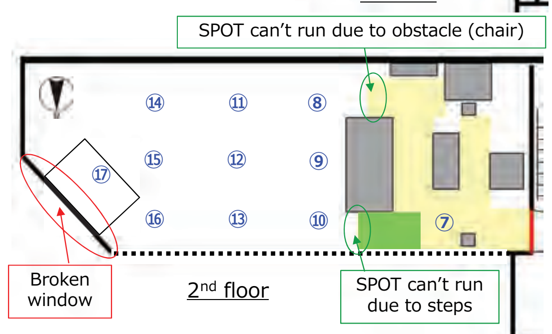
1st floor results

Point	γ dose rate [mSv/h]	
①	14.2※1	28.0※2
②	14.4※1	23.2※2
③	13.1※1	16.1※2
④	12.5※1	15.3※2
⑤	13.2※1	15.3※2
⑥	15.9※1	21.7※2

※1 : Measured at a height of 1500mm from the floor or OA floor
※2 : Measured at a height of 50 mm from the floor or OA floor
※3 : Measured at a height of approx. 500 mm from OA floor
※4 : Measured at the height the operation console (approx. 1300 mm from the OA floor)

2nd floor results

Point	γ dose rate [mSv/h]	
⑦	48.9※1	54.2※2
⑧	50.5※3	
⑨	58.8※3	
⑩	50.2※3	
⑪	57.3※3	
⑫	75.2※3	
⑬	60.1※3	
⑭	66.8※3	
⑮	76.1※3	
⑯	73.8※3	
⑰	53.2※4	



■ : SPOT path
■ : OA floor plate came off under SPOT's weight
① ~ ⑰ : survey point (⑧ ~ ⑰ measured after removal of the north wall)

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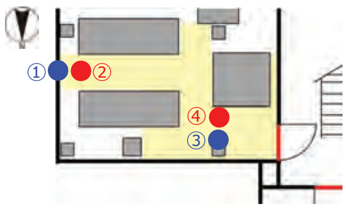
Unit 2 FHM remote control room investigation

TEPCO

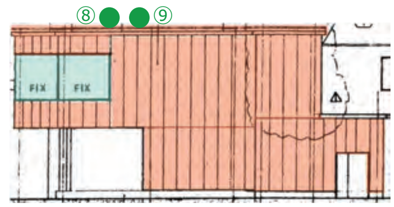
■ Smear sampling results (1^{st} floor and roof)



1st floor smear sampling points of the

● : floor
● : wall
● : roof

■ : SPOT path



Roof smear sampling points

Sampling date	No	α contamination			β contamination			Surface dose rate [mSv/h]			Sampling point
		BG [cpm]	Count rate [cpm]	Surface contamination density [Bq/cm²]	BG [cpm]	Count rate [cpm]	Surface contamination density [Bq/cm²]	BG	γ	β+γ	
7/26	①	0	0	<1.97E-01	1500	2000	6.47E+00	0.10	0.10	0.10	First floor wall
	②	0	0	<1.97E-01	1500	12000	1.36E+02	0.10	0.10	0.10	First floor
	③	0	0	<1.97E-01	1500	20000	2.39E+02	0.10	0.10	0.10	First floor wall
	④	0	0	<1.97E-01	1500	25000	3.04E+02	0.10	0.10	0.15	First floor
8/4	⑧	0	20	4.37E-01	1500	30000	3.69E+02	0.05	0.05	0.30	Roof
	⑨	0	0	<1.97E-01	1500	12000	1.36E+02	0.05	0.05	0.10	Roof

α contamination count rate measuring instrument : ZnS(Ag) scintillation detector
β contamination count rate measuring instrument : GM Survey Meter
Surface dose rate measuring instrument : ionization chamber survey meter

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Unit 2 FHM remote control room investigation

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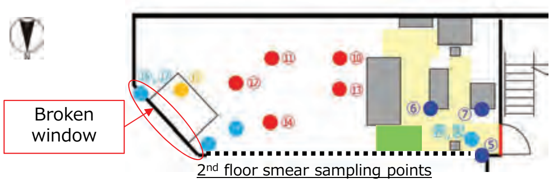
■ Smear sampling result (2^{nd} floor) (1/2)



2nd floor smear sampling points

● : floor
● : operation console surface
● : wall or equipment surface
● : Gypsum panel etc. collection points

■ : SPOT path
■ : OA floor plate came off under SPOT's weight

Sampling date	No	α contamination			β contamination			Surface dose rate [mSv/h]			Sampling point
		BG [cpm]	Count rate [cpm]	Surface contamination density [Bq/cm²]	BG [cpm]	Count rate [cpm]	Surface contamination density [Bq/cm²]	BG	γ	β+γ	
8/2	表	0	0	<1.97E-01	2000	65000	8.15E+02	0.05	0.05	2.40	Gypsum panel (top side when collected)
	裏	0	0	<1.97E-01	2000	70000	8.80E+02	0.05	0.05	0.70	Gypsum panel (bottom side when collected)
	–	–	–	–	–	–	–	0.05	4.00	200.0	Gypsum panel itself (top side when collected)
	–	–	–	–	–	–	–	0.05	4.00	200.0	Gypsum panel itself (bottom side when collected)
8/3	⑤	0	0	<1.97E-01	2000	60000	7.51E+02	0.03	0.03	0.30	Second floor wall
	⑥	0	0	<1.97E-01	2000	15000	1.68E+02	0.03	0.03	0.04	Surface of power board
	⑦	0	0	<1.97E-01	2000	15000	1.68E+02	0.03	0.03	0.04	Surface of power board
8/24	⑩	0	0	<1.97E-01	2000	10000	1.04E+02	0.05	0.05	0.15	Floor

α contamination count rate measuring instrument : ZnS(Ag) scintillation detector
β contamination count rate measuring instrument : GM Survey Meter
Surface dose rate measuring instrument : ionization chamber survey meter

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Unit 2 FHM remote control room investigation

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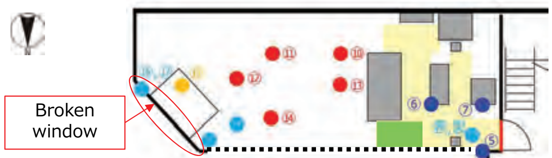
■ Smear sampling result (2nd floor) (2/2)



2nd floor smear sampling points

- ● : floor
- ● : operation console surface
- ● : wall or equipment surface
- ● : Gypsum panel etc. collection points
- ■ : SPOT path
- ■ : OA floor plate came off under SPOT's weight

Sampling date	No	α contamination			β contamination			Surface dose rate [mSv/h]			Sampling point
		BG [cpm]	Count rate [cpm]	Surface contamination density [Bq/cm²]	BG [cpm]	Count rate [cpm]	Surface contamination density [Bq/cm²]	BG	γ	β+γ	
8/24	⑩	0	0	<1.97E-01	2000	10000	1.04E+02	0.05	0.05	0.15	Floor
	⑪	0	0	<1.97E-01	2000	22000	2.59E+02	0.05	0.05	0.25	Floor
	⑫	0	0	<1.97E-01	2000	12000	1.29E+02	0.05	0.05	0.18	Floor
	⑬	0	0	<1.97E-01	2000	19000	2.20E+02	0.05	0.05	0.25	Floor
	⑭	0	0	<1.97E-01	2000	15000	1.68E+02	0.05	0.05	0.10	Floor
	⑮	0	60	1.31E+00	2000	50000	6.21E+02	0.05	0.05	1.00	operation console surface
9/15	⑯	0	0	<1.97E-01	2000	10000	1.04E+02	0.05	0.05	3.00	Window glass fragment (indoor side)
	⑰	0	0	<1.97E-01	2000	60000	7.51E+02	0.05	0.05	3.00	Window glass fragment (operating floor side)
	–	–	–	–	–	–	–	0.20	0.20	3.00	Window glass fragment itself (indoor side)
	–	–	–	–	–	–	–	0.20	0.20	3.00	Window glass fragment itself (operating floor side)

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Thank you for your attention

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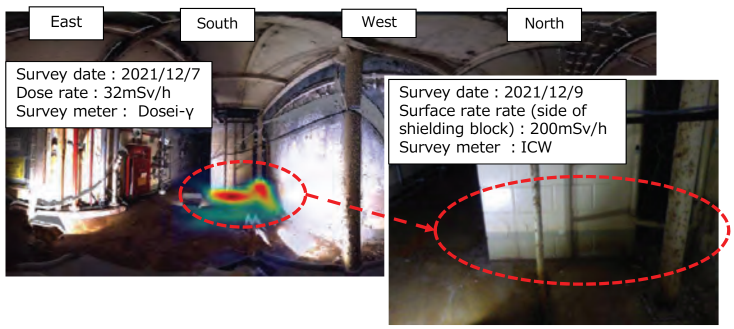
Units 1/2 R/B upper floors investigation

TEPCO

■ Objective
- ➢ Collection of spatial and contamination/dose rate information in R/Bs of Units 1/2 to the extent possible
- ➢ Collection of input information for future R/B investigation plans

■ Implementation status 【completed】
- ➢ Dose rate information near the AC (D/W vent) piping (Unit 1, 3rd floor) investigated

■ Related investigation item : RB-3a,b,8,9b,11,13,15,TRB-6



Contamination of the AC (D/W vent) piping and surrounding shielding blocks

Unit 2 shield plug deformation investigation

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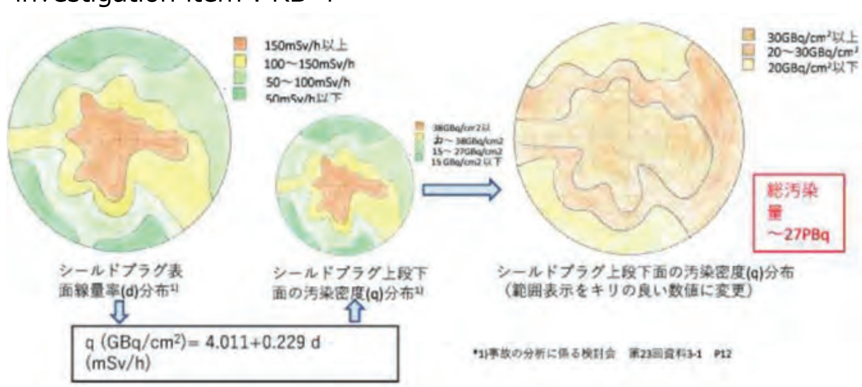
■ Objective
- ➢ Improvement of the accuracy of the evaluation of contamination estimated to be accumulated in the gaps between the upper and middle sections of the shield plug

■ Implementation status 【completed】
- ➢ Dose rate survey utilizing shield plug perforations in order to reduce the influence of the operation floor surface contamination shield plug perforation points,
- ➢ Measurement of dose rates in vertical direction inside the perforations
- ➢ Use of 2 perforations drilled in 2014 and 13 newly drilled perforations
- ➢ Reevaluation by NRA consistent with previous results (tens of PBq of Cs-137 present)

■ Related investigation item : RB-4



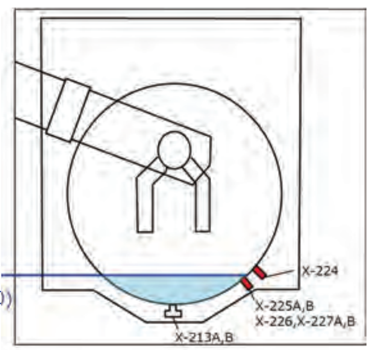
Unit 2 R/B stagnant water drawdown

- **Objective**
  - reduction of the amount of water retained in R/B to half of that in the end of 2020 (FY2022 ~ 2024)
- **Outline**
  - S/C water level equal to that of torus room (opening in the S/C assumed)
  - Opening assumed to be submerged (PCV pressure stable)
  - Exposure of piping near the bottom of the S/C expected with decrease of R/B water level
  - Identification of the S/C opening (its height) based on PCV pressure fluctuation
- **Results 【completion】**
  - Stagnant water level ~T.P. -2,800 mm (immediate target achieved).
  - No PCV pressure fluctuations ⇒ no openings above this level.
- **Related investigation item : RB-10,TRB-8**

S/C penetration	Use	Height of top of the penetration (T.P.)
X224	RCIC pump suction piping	Approx. -2300
X225A, B	RHR pump suction piping	Approx. -2900
X226	HPCI pump suction piping	Approx. -3000
X227A, B	CS pump suction piping	Approx. -3000
X213A, B	Drain (lock-opening)	Approx. -4000

Submerged range

Current stagnant water level (Approx. T.P.-2800)

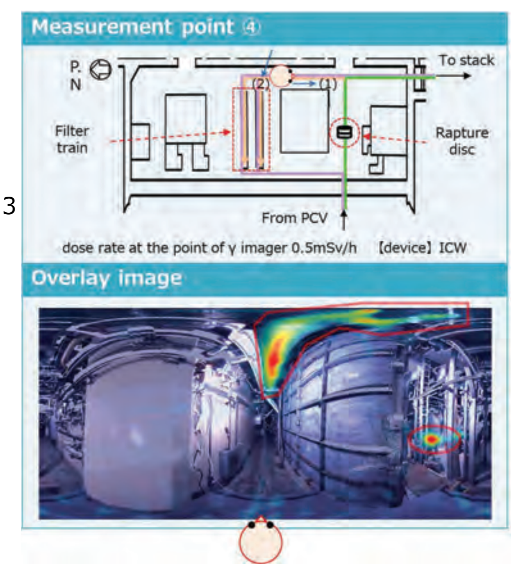


資料 2

Unit 1~4 SGTS room investigation and filters radionuclide analysis

- **Objective**
  - Clarification of radionuclides release behavior during PCV vent
- **Results**
  - Opening of SGTS filter trains of Units 3 and 4 for dose rate surveys, collection of smear samples from filters surfaces
  - Analysis of radionuclides of samples from Unit 3
  - Units 1, 3: contamination assumed to be caused by backflow from Unit 1 and Unit 3, respectively
  - Units 2, 4: contamination assumed to be caused by backflow from adjacent units
- **Related investigation item : RB-11,TRB-4**

Inside of the Unit 3 SGTS filter train（HEPA filter）



Unit 3 SGTS line contamination
(around the branch point with vent piping)

C.2.3.3. Combustible Gas Generation Heating Test Results

Experiment results of combustible gas generation from cables, paints, and heat insulators

2022/11/17 TEPCO

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Combustible Organic Gas Generation Assessment Matrix TEPCO

- Heating tests were conducted on each of the cable, paint, and heat insulator materials to evaluate the type and amount of gas generated.

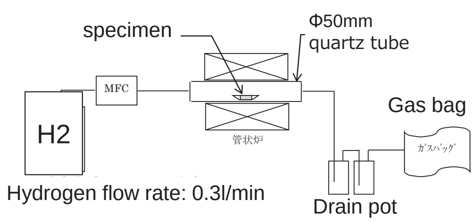
Assessment Matrix

No.	type	Specimen	Usage in plant	Testing Methods	status
1	cable	CV cable electrical insulator : Cross-linked polyethylene Sheath: flame-retardant heat-resistant vinyl	・Used for high-voltage power cable	Thermogravimetry (TG), FT-IR(Fourier transform infrared spectroscopy), SEM-EDX	completed
2	cable	PN cable Electrical insulator: Flame-retardant ethylene-propylene rubber Sheath: Special chloroprene rubber	・Used for control and instrumentation cable ・Installed at the bottom of RPV	Thermogravimetry (TG), FT-IR、SEM-EDX	completed
3	cable	Coaxial cable Electrical insulator: ETFE/Cross-linked polyethylene Sheath: flame-retardant cross-linked polyethylene	・Used for SRNM/LPRMcable ・Installed at the bottom of RPV	Thermogravimetry (TG), FT-IR、SEM-EDX	completed
4	paint	Epoxy paints	・D/W、S/C	Thermogravimetry (TG), FT-IR、SEM-EDX	completed
5	paint	Inorganic zinc paint	・D/W、S/C		FY2022
6	Heat insulator	Urethane heat insulator	・Piping insulation	Thermogravimetry (TG), FT-IR、SEM-EDX	completed
7	Heat insulator	Polyimide heat insulator	・Piping insulation	Thermogravimetry (TG), FT-IR、SEM-EDX	completed

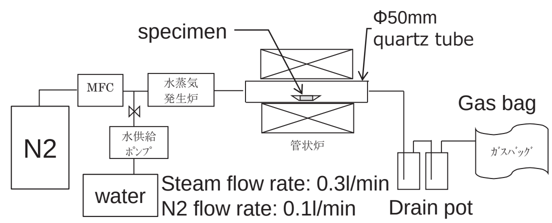
1

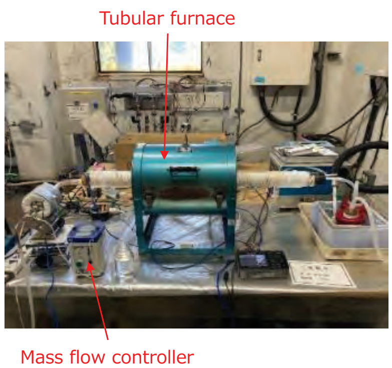
Schematics of combustible organic gas generation experiment

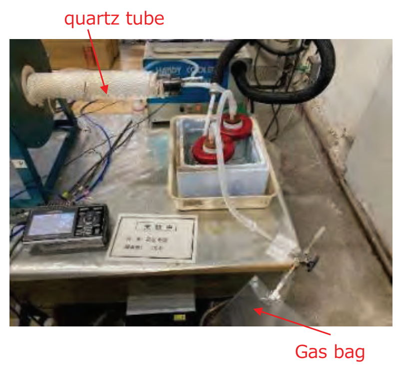
■Hydrogen gas environment



■Steam environment



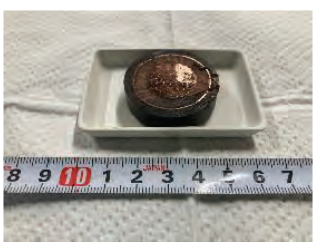




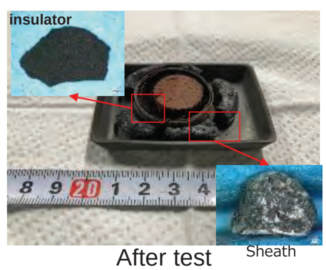
2

CV cable test result （1／2）

■CV cable: comparison before and after the test



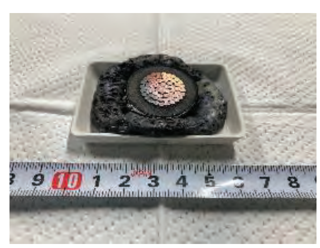
Before test



After test
（1000℃ H2 gas environment）

After test
（1000℃ steam environment）

⇒In the hydrogen gas environment, the organic components, sheaths and insulators, were carbonized, but in the steam environment, almost all of the organic components were disappeared.



After test
（200℃ steam environment）

3

CV cable test result（2／2） TEPCO

■Results of gas analysis generated when CV cable was heated up to 1000°C under hydrogen and steam environment and kept at 200°C for 24 hours under steam environment.
（Converted to gas production per ton of cable）

Sample		CV cable gas generation rate(m3/t)						
Environment		hydrogen			steam			steam
Temperature (℃)		RT～350	350～500	500～1000	RT～350	350～500	500～1000	200
H2		－	－	－	－	－	1.01E+02	－
CO		－	－	2.74E+00	－	－	1.95E+01	－
hydrocarbon	CH4	1.52E-04	1.90E-01	1.10E+01	2.29E-04	6.53E-03	1.81E+01	2.98E-04
	C2H4	1.52E-04	1.42E-01	1.92E+00	－	4.57E-03	1.17E+01	9.92E-05
	C2H6	－	1.33E-01	1.92E+00	－	3.59E-03	2.01E+00	－
	C3H6	－	1.23E-01	6.17E-01	－	1.96E-03	2.41E+00	－
	C3H8	2.27E-04	8.06E-02	1.92E-01	－	1.63E-03	4.58E-01	6.95E-04
	i-C4H10	－	1.33E-03	5.48E-03	－	1.41E-04	7.45E-03	－
	n-C4H10	－	5.12E-02	9.46E-02	－	5.22E-04	2.87E-01	－
	i-C5H12	－	1.80E-02	8.78E-02	－	－	2.64E-01	－
	n-C5H12	－	1.71E-02	8.36E-02	－	－	1.98E-01	－
	otherC1～C5 (CH4 equivalent)	－	2.65E-01	1.06E+00	－	4.03E-03	4.58E+00	－
	total value of all CH4 equiva	1.25E-03	1.90E+00	2.33E+01	5.33E-04	3.92E-02	6.02E+01	2.58E-03
NH3		－	－	－	3.81E-05	－	－	4.96E-05
H2S		8.34E-05	2.09E-03	3.70E-03	7.46E-04	4.46E-05	2.87E-04	3.37E-04
CO2 and other carbon molecule ratio in total generated gas		21.9			17.3			

4

PN cable test result （1／2） TEPCO

■PN cable: comparison before and after the test

Before test

After test
（1000℃ H2 gas environment）

After test
（1000℃ steam environment）

After test
（200℃ steam environment）

5

PN cable test result（2／2）

TEPCO

■Results of gas analysis generated when PN cable was heated up to 1000°C under hydrogen and steam environment and kept at 200°C for 24 hours under steam environment.

（Converted to gas production per ton of cable）

Sample		PN cable gas generation rate(m3/t)						
Environment		hydrogen			steam			steam
Temperature(℃)		RT～400	400～500	500～1000	RT～400	400～500	500～1000	200
H2		−	−	−	−	1.55E−01	3.98E+02	−
CO		−	4.09E−02	4.32E−01	−	6.87E−02	1.62E+02	−
hydrocarbon	CH4	4.71E−02	3.80E−01	4.75E+00	3.86E−03	1.22E−01	1.83E+01	3.72E−03
	C2H4	9.11E−02	2.21E−01	4.15E−01	4.87E−03	1.41E−01	6.26E+00	5.32E−04
	C2H6	1.40E−02	2.09E−01	3.89E−01	7.16E−04	5.15E−02	3.05E+00	−
	C3H6	5.16E−03	1.02E−01	1.99E−01	2.86E−04	1.89E−02	2.70E+00	−
	C3H8	7.44E−03	8.18E−02	9.08E−02	2.86E−04	1.63E−02	9.57E−01	1.33E−03
	i−C4H10	−	3.60E−03	3.46E−03	−	6.18E−04	4.79E−02	−
	n−C4H10	5.77E−03	4.91E−02	4.32E−02	−	7.90E−03	5.39E−01	−
	i−C5H12	−	1.06E−02	1.90E−02	−	6.01E−04	2.18E−01	−
	n−C5H12	4.86E−03	2.29E−02	4.32E−02	−	9.62E−04	4.26E−01	−
	otherC1～C5(CH4 equivalent)	7.75E−02	3.35E−01	3.89E−01	2.29E−03	9.10E−02	6.35E+00	−
	total value of all CH4 equivalents	3.95E−01	2.41E+00	7.78E+00	1.86E−02	7.04E−01	5.74E+01	8.91E−03
NH3		−	−	−	−	−	−	−
H2S		1.20E−01	1.47E−01	1.04E−01	6.15E−03	5.84E−02	3.31E−01	4.65E−04
CO2 and other carbon molecule ratio in total generated gas		39.2			15.6			

6

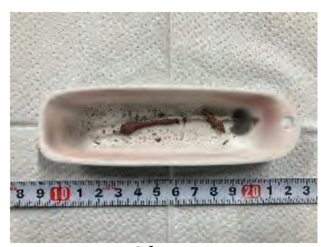
Coaxial cable test result（1／2）

TEPCO

■Coaxial cable: comparison before and after the test

Before test

After test
（1000℃ H2 gas environment）



After test
（1000℃ steam environment）

After test
（200℃ steam environment）

7

Coaxial cable test result （2／2）

TEPCO

■Results of gas analysis generated when coaxial cable was heated up to 1000°C under hydrogen and steam environment and kept at 200°C for 24 hours under steam environment.
（Converted to gas production per ton of cable）

Sample		coaxial cable gas generation rate(m3/t)						
Environment		hydrogen			steam			steam
Temperature(℃)		RT～400	400～540	540～1000	RT～400	400～540	540～1000	200
H2		–	–	–	–	–	3.37E+01	–
CO		–	–	–	–	–	1.17E+01	–
hydrocarbon	CH4	1.27E-03	1.52E-01	2.76E+00	2.14E-03	1.81E-02	7.12E+00	1.29E-03
	C2H4	2.29E-03	1.62E-01	5.31E-01	–	2.05E-02	5.44E+00	–
	C2H6	7.62E-04	1.20E-01	5.84E-01	5.35E-04	1.40E-02	1.26E+00	–
	C3H6	7.62E-04	1.20E-01	1.81E-01	–	7.91E-03	1.51E+00	–
	C3H8	7.62E-04	5.08E-02	6.37E-02	1.87E-02	6.51E-03	2.93E-01	3.22E-03
	i-C4H10	–	1.02E-03	5.31E-04	–	–	–	–
	n-C4H10	–	2.91E-02	2.92E-02	–	1.12E-03	1.72E-01	–
	i-C5H12	–	1.43E-02	1.86E-02	–	–	6.28E-02	–
	n-C5H12	–	1.20E-02	2.02E-02	–	–	3.98E-02	–
	otherC1～C5(CH4 equivalent)	5.84E-03	2.68E-01	3.29E-01	–	1.07E-02	2.93E+00	–
	total value of all CH4 equiva	1.80E-02	1.66E+00	6.37E+00	5.88E-02	1.40E-01	2.72E+01	1.22E-02
NH3		–	–	–	–	–	–	–
H2S		–	–	2.02E-03	3.74E-04	1.49E-04	2.93E-04	1.22E-03
CO2 and other carbon molecule ratio in total generated gas		36.4			33.6			

8

Epoxy paint test result（1／2）

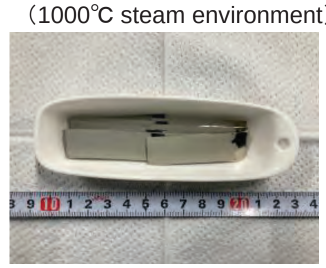
TEPCO

■Epoxy paint : comparison before and after the test

Before test

After test
（1000℃ H2 gas environment）

After test
（1000℃ steam environment）



After test
（200℃ steam environment）

9

Epoxy paint test result （2／2） TEPCO

■Results of gas analysis generated when epoxy paint was heated up to 1000°C under hydrogen and steam environment and kept at 200°C for 24 hours under steam environment.
（Converted to gas production per ton of cable）

Sample		Epoxy paint gas generation rate(m3/t)						
Environment		hydrogen			steam			steam
Temperature（℃）		RT～200	200～600	600～1000	RT～200	200～600	600～1000	200
H2		–	–	–	–	–	1.31E+02	–
CO		–	–	1.50E+00	–	–	2.05E+01	–
hydrocarbon	CH4	1.97E-04	2.36E-01	3.74E+00	1.11E-02	2.39E-02	1.57E+01	8.24E-03
	C2H4	–	4.13E-02	2.69E-01	8.55E-03	4.87E-03	4.44E+00	–
	C2H6	–	5.51E-02	2.17E-01	2.56E-03	3.42E-03	5.40E-01	–
	C3H6	–	3.94E-02	1.72E-02	6.84E-03	3.76E-03	6.22E-01	–
	C3H8	1.97E-04	1.91E-02	8.23E-03	2.56E-03	1.45E-03	1.37E-01	2.06E-03
	i-C4H10	–	–	–	–	4.27E-04	–	–
	n-C4H10	–	5.71E-03	–	1.71E-03	3.42E-04	3.76E-02	–
	i-C5H12	–	–	–	–	–	–	–
	n-C5H12	–	3.54E-03	–	–	5.13E-04	1.91E-02	–
	otherC1～C5（CH4 equivalent）	2.36E-03	1.24E-01	1.65E-02	3.59E-02	9.40E-03	8.21E-01	6.18E-02
	total value of all CH4 equiva	3.15E-03	7.28E-01	4.79E+00	1.11E-01	6.92E-02	2.87E+01	8.03E-02
NH3		1.97E-04	3.94E-02	1.50E-03	–	–	–	–
H2S		–	–	8.98E-02	–	–	6.77E-01	4.12E-04
CO2 and other carbon molecule ratio in total generated gas		64.4			60.4			

10

Urethane heat insulator test result （1／2） TEPCO

■Urethane heat insulator : comparison before and after the test

Before test

After test
（1000℃ H2 gas environment）

After test
（1000℃ steam environment）

After test
（200℃ steam environment） 11

Urethane heat insulator test result （2／2）

TEPCO

■Results of gas analysis generated when urethane heat insulator was heated up to 1000°C under hydrogen and steam environment and kept at 200°C for 24 hours under steam environment.
（Converted to gas production per ton of cable）

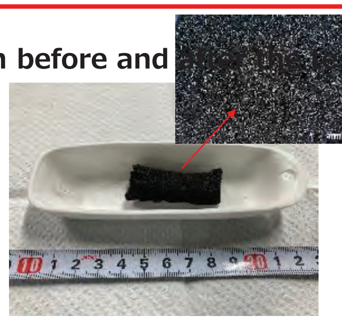
Sample		Urethane heat insulator gas generation rate(m3/t)						
Environment		hydrogen			steam			steam
Temperature（℃）		RT～230	230～370	370～1000	RT～230	230～370	370～1000	200
H2		－	－	－	－	－	2.64E+02	－
CO		－	－	－	－	－	1.16E+02	－
hydrocarbon	CH4	5.83E-04	1.25E-03	7.35E+00	6.29E-03	1.09E-03	2.72E+01	5.49E-03
	C2H4	－	4.17E-04	5.43E-01	3.59E-03	7.03E-04	6.68E+00	－
	C2H6	－	5.00E-04	6.83E-01	8.98E-04	1.56E-04	1.25E+00	－
	C3H6	－	1.92E-02	5.95E-01	2.70E-03	5.47E-04	3.63E+00	－
	C3H8	2.92E-04	5.00E-04	1.51E-01	1.80E-03	3.13E-04	4.30E-01	1.37E-03
	i-C4H10	－	－	－	－	－	－	－
	n-C4H10	－	－	－	－	7.81E-05	6.00E-02	－
	i-C5H12	－	1.17E-03	1.23E-02	－	－	－	－
	n-C5H12	－	3.08E-03	2.28E-02	－	－	1.36E-02	－
	otherC1～C5（CH4 equivalent）	1.52E-01	2.17E-01	3.15E-01	5.03E-01	2.27E-01	2.15E+00	8.37E-01
	total value of all CH4 equiva	1.55E-01	2.92E-01	1.24E+01	5.30E-01	2.34E-01	5.66E+01	8.51E-01
NH3		－	－	6.48E-01	－	－	3.96E-03	－
H2S		8.75E-05	7.50E-05	－	7.28E-03	9.38E-04	4.30E-02	4.67E-03
CO2 and other carbon molecule ratio in total generated gas		73.1			55.5			

Polyimide heat insulator test result （1／2）

TEPCO

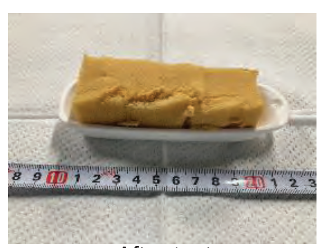
■PN cable: comparison before and [illegible]t

Before test



After test
（1000℃ H2 gas environment）

After test
（1000℃ steam environment）



After test
（200℃ steam environment）

Polyimide heat insulator test result （2／2）

■Results of gas analysis generated when polyimide heat insulator was heated up to 1000°C under hydrogen and steam environment and kept a 200°C for 24 hours under steam environment.
（Converted to gas production per ton of cable）

Sample		Polyimide heat insulator gas generation rate(m3/t)						
Environment		hydrogen			steam			hydrogen
Temperature（℃）		RT～520	520～700	700～1000	RT～520	520～700	700～1000	200
H2		－	－	－	－	－	6.32E+02	－
CO		－	7.38E+00	3.08E+00	－	1.71E+00	3.94E+02	－
hydrocarbon	CH4	1.40E-02	2.22E-01	9.85E+00	6.57E-03	5.14E-02	2.36E+01	1.52E-04
	C2H4	1.08E-03	1.35E-02	5.23E-02	－	6.57E-03	1.36E+00	－
	C2H6	－	9.23E-03	4.00E-02	－	2.86E-04	3.53E-02	－
	C3H6	－	3.08E-03	－	－	2.00E-03	7.87E-02	－
	C3H8	1.08E-03	2.65E-02	7.38E-02	－	2.86E-04	1.60E-01	－
	i-C4H10	－	－	－	－	－	－	－
	n-C4H10	－	－	－	－	－	－	－
	i-C5H12	－	－	－	－	－	－	－
	n-C5H12	－	－	－	－	－	－	－
	otherC1～C5（CH4 equivalent）	4.95E-02	1.78E-02	2.46E-02	－	－	7.60E-02	－
	total value of all CH4 equivalents	7.22E-02	3.69E-01	1.05E+01	1.31E-02	7.14E-02	2.69E+01	4.55E-04
NH3		5.38E-04	3.08E-02	1.60E+00	1.64E-03	－	1.36E-03	－
H2S		－	－	－	3.29E-04	5.71E-05	1.63E-03	－
CO2 and other carbon molecule ratio in total generated gas		75.9			25.7			

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Summary

- Gas analysis of three types of cable, epoxy paint, and two types of thermal insulators at 1000°C temperature rise and 200°C 24-hour holding under hydrogen gas and water vapor environment
- Confirmed that almost no flammable gases were generated under a 200°C, 24-hour environment.
- Confirmed that more combustible gas is generated in a water vapor environment than in a hydrogen environment.

＜Plan for FY2022＞

- Experiments for inorganic zinc paint, organic zinc paint, and KGB cable (silicone cable)

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Thank you for your attention

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C.2.3.4. Sixth Unconfirmed/Unresolved Issues Report

The 6th Progress Report
on the Investigation and Examination of
Unconfirmed and Unresolved Issues
on the Development Mechanism
of the Fukushima Daiichi Nuclear Accident

2022/11/10
Tokyo Electric Power Company Holdings, Inc.

Overview of the 6th Progress Report

52 unidentified and unsolved issues of detailed accident progression were extracted.

↓

By the 5th Progress Report in 2017, 34 issues were reported, including 10 issues that are important for understanding the accident progression.

Unreported Issues	Reported issues (34)		
Issues that help understanding of accident progression 18 issues	Issues that help understanding of accident progression 23 issues	Debris estimation diagram based on onsite information 1issue	Important issues that help understanding of accident progression 10 issues
6th Report 4 new studies	6th Report 5 new studies added※	6th Report updated※	Reported by 4th report

In the 6th progress report, we reported the additional 9 studies and updated the current estimates for fuel debris distribution.

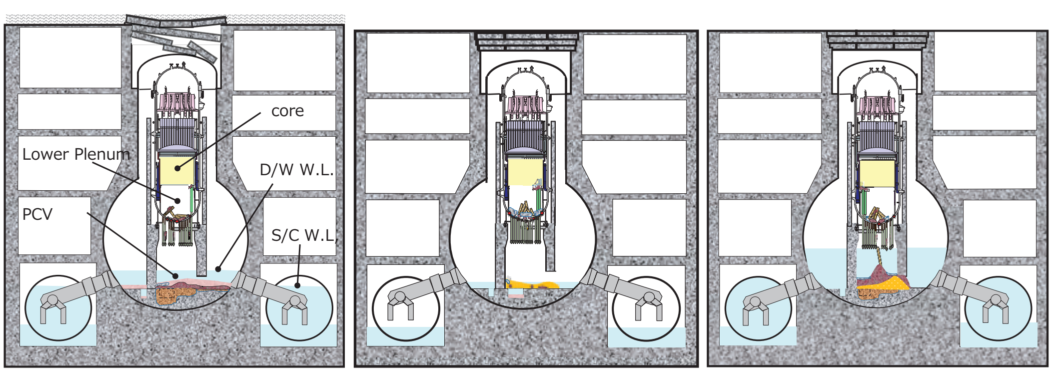
The list of additional attachment reports
in 6th progress report

1. **Debris estimation diagram**
2. Identification of the cause of the high radiation dose rate observed in the southeast area of the first floor of the Unit 1 reactor building
3. Estimation of the reason why high dose rate was not observed in the auxiliary cooling water system of the Unit 2 reactor.
4. Decrease in containment pressure in Unit 2 on the morning of March 15
5. Behavior of S/C pressure gauge in Unit 2 after 21:00 on March 14
6. **Re-evaluation on evaluation method of the core damage ratio of the Mark-I containment vessel**
7. Study on the water level in the pressure suppression chamber of Unit 3
8. Accident progress after reactor depressurization of Unit 3
9. Examination of plant conditions during RCIC operation of Unit 3
10. Investigation of accident conditions by sample analysis

TEPCO 2

Unit 1 to 3 Debris estimation diagram

	core	Lower plenum	PCV	D/W water level	S/C water level
Unit 1	Almost none	Almost none	Most part	2m	Almost full
Unit 2	Little V	Lot V	Little Λ	0.3m	Low
Unit 3	Little	Little	A certain level	5m	Full



Unit 1 Unit 2 Unit 3

TEPCO 3

2．1号機原子炉建屋1階南東エリアで観測された高線量率の原因の特定

- 1号機では、事故直後に１階南東エリアにて1000mSv/hを超える高線量率が観測されている。
- 当該エリアでは、2011年6月に床貫通部から湯気が流出していることが確認された。
- エリア近傍にはPCVベントで使用したAC配管が敷設されていることもあり、本検討では南東エリアで高線量率が観測された原因と考えられるこれらの影響を中心に、その他の可能性も含めて原因を特定した。

南東エリアの床貫通部で確認された湯気
(2011年6月撮影)

＜検討のアプローチ＞
- 高線量率の原因となりうる汚染源を洗い出し、南東エリアへの影響を検討
- 抽出した汚染源について、以下の観点から南東エリアへの影響を評価

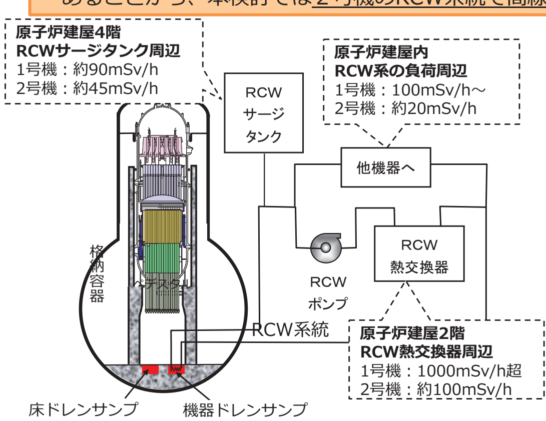
(1)汚染の原因
(2)汚染源からの放射線による影響
(3)汚染源からの放射性物質の移行の有無

⇒ 南東エリアで観測された高線量率は、PCVベントに使用したAC配管からの放射線の影響が支配的な要因であると特定

柏崎刈羽原子力発電所安全対策への反映状況：
ベントラインからの放射線による事故対応操作への影響を低減する対策

3．2号機原子炉補機冷却水系に高線量率が観測されなかった原因の推定

- １号機では、原子炉補機冷却水系（RCW）※の負荷である機器の周辺で高線量率を観測しており、その原因は、原子炉圧力容器から落下した燃料が、格納容器床にある機器ドレンサンプ内のRCW配管を損傷し、RCW系統全体に汚染が拡がったものと推定（第４回進捗報告で報告済み）。
- 一方、２号機でも原子炉圧力容器から燃料の一部が格納容器に落下したと推定しているものの、RCW系統に顕著な汚染の痕跡はみられない。
- この差異を明らかにすることは、燃料デブリの分布を推定に加え、事故進展の推定にも寄与するものであることから、本検討では２号機のRCW系統で高線量率が観測されなかった原因を推定。



1、２号機 RCW系統と汚染のイメージ

＜検討のアプローチ＞
- 2号機の格納容器内部調査の結果をもとに、格納容器内におけるRCW配管の状態を考察
- 事故時のプラントデータから、系統内に汚染が拡がる状況にあったか考察

格納容器内部調査の結果から、2号機のRCW系統で高線量率が観測されなかった要因は、1号機と異なりRCW配管が損傷を受けなかったことが原因と推定。

柏崎刈羽原子力発電所安全対策への反映状況：格納容器内の配管損傷に伴う汚染の拡大を防止する対策

※原子炉建屋内等の機器を冷却する系統。原子炉圧力容器や格納容器に対する開放部のない閉ループの設計。

4．3月15日午前中における2号機格納容器圧力の低下について

- 2号機のD/W圧力は、3月14日23:30頃～15日7:20まで0.7MPa[abs]以上で推移し、一旦途切れた計測が再開された15日11:20には、0.155MPa[abs]まで低下していた。
- PCV圧力の低下は、放射性物質の放出と関連するものであることから、この圧力の低下挙動を解明することは重要であり、本検討では、RPV圧力やPCV圧力等のプラントパラメータの指示値および観測事実と整合するシナリオを検討した。

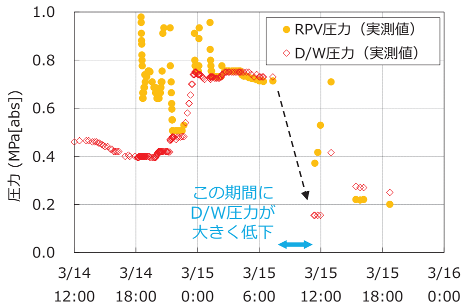


図 RPV・PCV圧力の推移

＜検討のアプローチ＞
以下２つのシナリオの成立性を検討。

① PCVからの大規模な気相漏えいのみによる減圧
- 減圧を再現するPCV気相漏えい面積を評価
- 観測事実等を踏まえたシナリオの成立性の検討

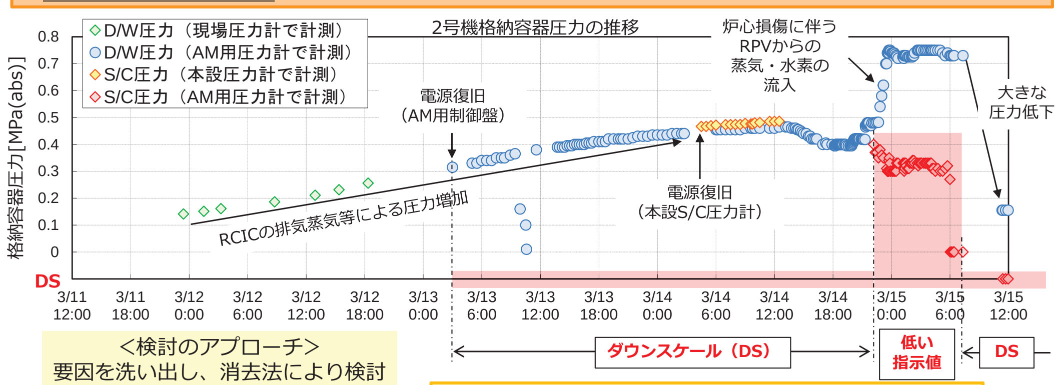
② PCVからの気相漏えいに加え、PCV内の水蒸気の凝縮により減圧
- PCV内の凝縮が促進されるシナリオの想定
- 想定したシナリオにおける減圧挙動の評価
- 観測事実等を踏まえたシナリオの成立性の検討

- PCVからの大規模な気相漏えいのみによる減圧シナリオは、観測事実と整合しない点がある。（事故後２号機のPCVの気密性が比較的高いことや、トップヘッドフランジ以外からの漏えいも考える必要があるもののオペフロ以外の建屋内の汚染が比較的小さいことなど）
- 小規模な漏えいに加えて、水蒸気の凝縮も減圧に寄与したと考えると、観測事実と整合する点が多い。ただし、凝縮の効果はPCV内の状態に大きく依存するため、そのような事故進展となっていたかも含め、引き続き検討を進めていく。

TEPCO 6

5. 2号機3月14日21時以降のS/C圧力計の挙動について

- 事故当時使用していた2号機の格納容器圧力計のうち、AM用S/C圧力計は3月13日3時にバッテリーを接続し電源を復旧したが、ダウンスケール（以後、DSという）や、D/W圧力より約400kPaも低い指示値など、他の圧力計とは大きく異なる値を示した。
- このようなD/W圧力とS/C圧力の大きな乖離は格納容器の構造上発生するものではなく、DSを指示していることからもAM用S/C圧力計が実際の圧力を指示していなかった可能性が極めて高い。
- 格納容器圧力は事故対応において非常に重要なパラメータであることから、AM用S/C圧力計が異常な指示値を示した要因について検討した。



＜検討のアプローチ＞
要因を洗い出し、消去法により検討

要因の分類
①機械的要因
②測定原理に関する要因
③電気的要因

AM用S/C圧力計は水没による電気故障のため、実態とはかけ離れた異常な指示値を示していたと推定

柏崎刈羽原子力発電所安全対策への反映状況：
溢水による計器水没への対策

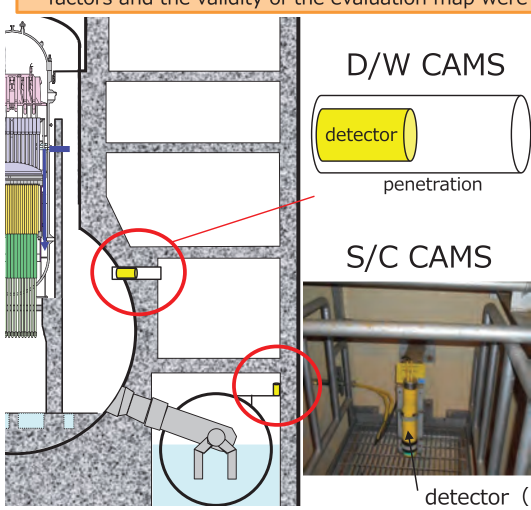
TEPCO 7

6. Re-evaluation on evaluation method of the core damage ratio of the Mark-I containment vessel

TEPCO 8

Overview

- In Unit 2, the core damage occurred after the D/W and S/C CAMS (containment atmosphere monitoring system) resumed measurement due to the restoration of power supply. In the third and fourth progress reports, we estimated the accident progression based on these measurements and evaluated the FP presence rate during the time when core damage and fuel meltdown progressed.
- Furthermore, focusing on the CAMS measurements, differences were found between the trends of the actual data and the time-dose maps for the evaluation of the core damage ratio (hereinafter referred to as "evaluation maps").
- Since the CAMS measurements are important data for understanding the accident progression, their factors and the validity of the evaluation map were discussed.



<Approach to the study>
- Compare the D/W and S/C CAMS measurements of Unit 2 with the evaluation map, and evaluate the reproducibility of the evaluation map.
- Examine from the viewpoint of evaluation method and geometric factors (location of D/W and S/C CAMS detectors, etc.)

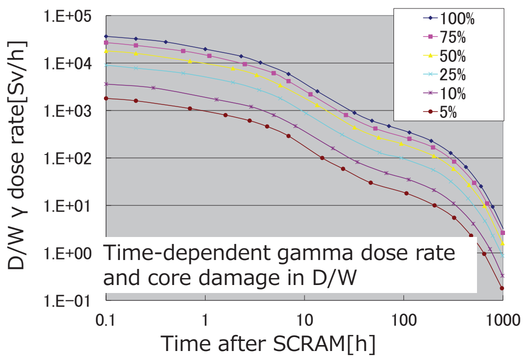
There is a discrepancy between the evaluation map and reality, mainly because the evaluation map does not take into account that the S/C wall (shielding) and the location of the CAMS detector (separation from the source) will be smaller.

Status of Reflection on Safety Measures at KKNPS : Validation of procedures for estimating core conditions from CAMS dose rates

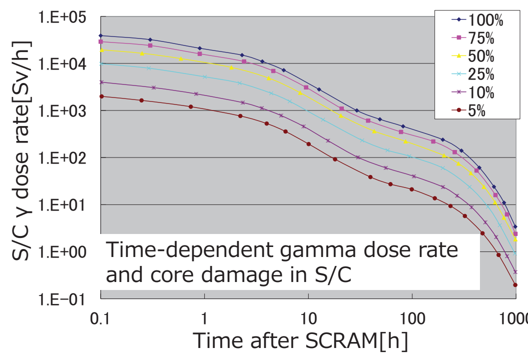
TEPCO 9

Fukushima Daiichi BWR4 core damage ratio map

- In the evaluation map for the BWR4 Mark-I containment used in the Fukushima Daiichi NPP accident, there is no significant difference in the CAMS dose rate for each damage fraction between D/W and S/C.
- The evaluation maps were prepared by considering the radiation from only noble gas released from the fuel, therefore it has been considered that the core damage ratio is conservatively evaluated when iodine and other FPs are released at the same time.



Time-dependent gamma dose rate and core damage in D/W



Time-dependent gamma dose rate and core damage in S/C

At the time of the accident, the core damage ratio was evaluated and reported using these evaluation maps. However, while current understanding is 100% core damage in all units, the evaluated values at that time were smaller, less than 100%.
（Unit 1 : 55%、Unit 2 : 35%、Unit 3 : 30%　2011/04/27)

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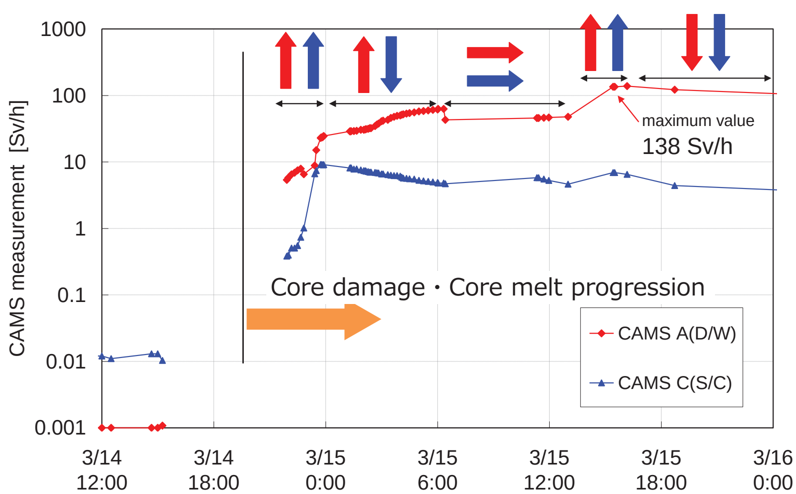
CAMS γ dose rate measurements for Unit 2



- The CAMS measurements after core damage resumed on the night of 3/14 are always about one order of magnitude lower in S/C.
- At the beginning of core damage, FP flows via SRVs from S/C to D/W, and a large amount of FP is expected to be present in S/C. However, the characteristics of the measured values are different.

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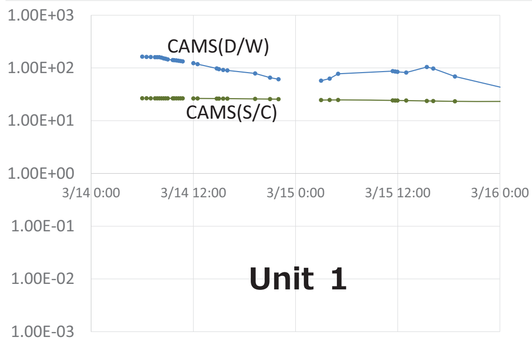
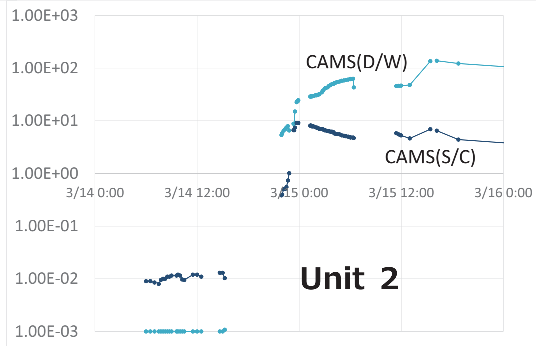
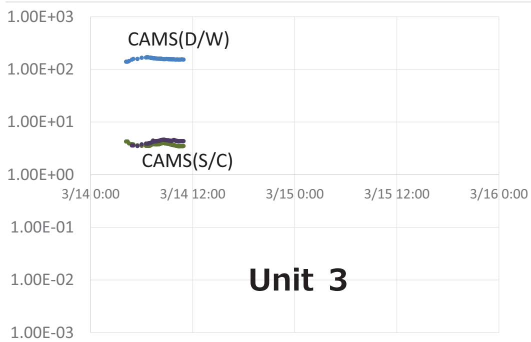
(ref.) CAMS γ dose rate measurements in unit 1 to 3







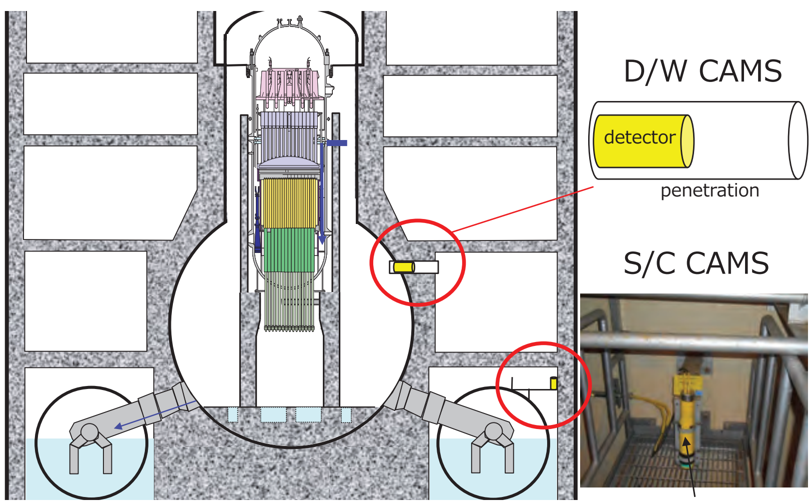
Unit : [Sv/h]

Accident progression differs for Units 1~3, but CAMS measurements show the same trend (D/W > S/C)

Note: Measured values in Units 1 and 3 are after venting

12

Location of CAMS detector



detector（installed in unit 5）

D/W CAMS: Located just outside of the D/W shell
S/C CAMS : Installed on the wall of the torus room at a short distance from S/C
⇒Different distances from the radiation source may affect the value of dose rate

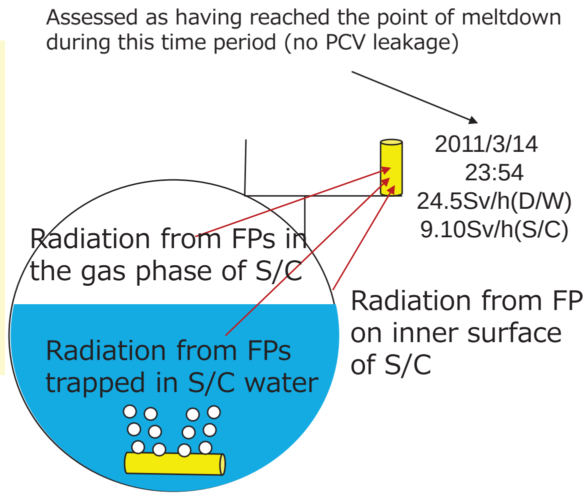
TEPCO 13

The late night of March 14, 2012
Migration and distribution of FPs to S/C

When FP released from the fuel is released into the S/C through the SRV

- noble gas FPs migrate directly to the gas phase of S/C.
- Volatile FPs such as iodine and Cs are mostly trapped in water.
- Some of the volatile FPs in the gas phase adhere to the inner surface of the S/C.

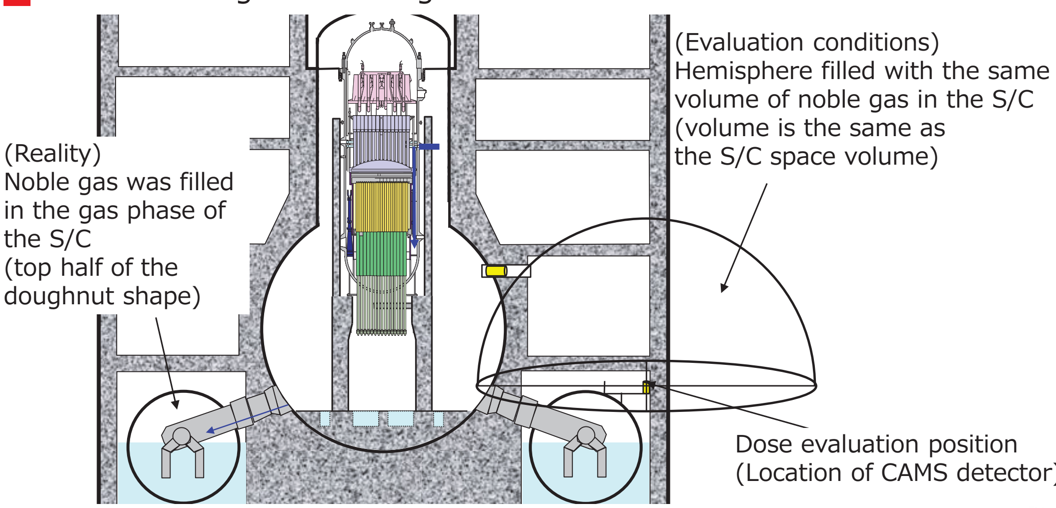
The measured value of 9.1 Sv/h on 3/14 23:54 is the sum of radiation from three sources



<u>The evaluation map</u> conservatively assumes that only the contribution of noble gases is taken into account, which is equivalent to **<u>460 Sv/h after 81.8 hours and 240 Sv/h after 197.4 hours in the case of 100% FP release</u>**, while the contribution of noble gas is estimated to be **<u>about 1.2 Sv/h at most</u>** out of the **<u>measured value of 9.1 Sv/h</u>**.
→In other words, the S/C CAMS measurements and the evaluation map are inconsistent due to the influence of the installation location.

TEPCO 14

Dose rate calculation method used in the methodology for evaluating core damage ratio



A simplified dosimetry calculation method assuming a hemispherical plume is used in the core damage ratio map for Mark I PCV.
The inconsistency between calculated and measured value is caused by the fact that the calculated values of CAMS are not taken into account for the S/C wall (shielding) and the CAMS installation position (separation from the source).
(However, at the time of the Fukushima Daiichi NPP accident, core meltdowns were also evaluated using D/W CAMS figures, so the effect of the underestimation is small.)

Summary

Re-evaluated the validity of the evaluation map to evaluate the core damage ratio, etc., using the CAMS measurements for Unit 2.

- It was found that the evaluation map, which was supposed to conservatively evaluate the core damage ratio, tended to underestimate the core damage ratio when evaluated using S/C CAMS.

- This was presumed to be due to the fact that it did not properly reflect the effects of the geometry of the Mark-I containment vessel and the location of the CAMS detector.

TEPCO 16

Lessons Learned and Safety Measures at KKNPS

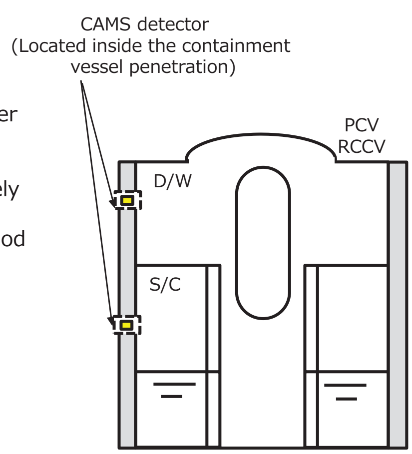
Lessons Learned : When estimating core conditions from CAMS dose rates, attenuation due to shielding and distance between the source and the CAMS detector must be properly considered.

■ **Confirmed the validity of the method for estimating core damage ratio using CAMS dose rate**

(1) Determination of core damage
The following confirms that there are no obstacles to the determination.

- In Units 6 and 7 of the Kashiwazaki Kariwa Nuclear Power Plant, both D/W and S/C CAMS detectors are located inside the containment vessel penetration.
- The dose rate to determine core damage is conservatively low to avoid delay in judgment.
- Since the dose rate increases significantly in a short period of time at the time of core damage, the influence of the uncertainty of the core damage determination curve on the determination time of core damage is small.

(2) Estimation of core damage ratio
The core damage ratio has not been used to determine the operation by the operator.
In the manuals referred to by organizations that provide technical support to operators, the conventional practice of calculating the core damage ratio has been abolished.



CAMS detector location in ABWR

TEPCO 17

7. 3号機圧力抑制室水位にかかる検討

- 3号機の13日9:00頃の原子炉減圧以降の事故進展（格納容器ベント、圧力容器・格納容器からの気相漏えい、水素爆発など）や燃料デブリの冷却状態などを推定する上で、格納容器圧力のデータを理解することが重要である。
- 3号機では3月11日17:15～12日20:00にかけて、S/C水位のデータが採取されている。
- このデータは、水素発生量の推定や、S/CからD/Wへの水の逆流有無にかかる推定に役立つもので、上述の事故進展を理解する上で重要な情報である。本検討では、3月13日9時頃のS/Cベント（以降、「第一回ベント」という）開始時のS/C水位に着目し、これを推定した。

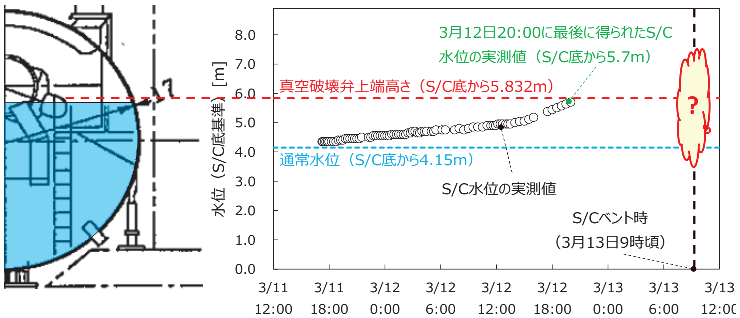


図　S/C水位の実測値

＜検討のアプローチ＞
以下の独立な２つの方法で第一回ベント時のS/C水位を評価し、両方の評価結果を総合してS/C水位を推定

(1)得られているS/C水位データに基づく評価
(2)格納容器圧力データに基づく評価

第一回ベント開始時点のS/C水位はS/C底から7m前後と、真空破壊弁を超えて高かったと推定した。
⇒13日20:40以降のD/W減圧時には、水がS/CからD/Wに逆流し、落下してきた燃料デブリの冷却に寄与した可能性がある。

柏崎刈羽原子力発電所安全対策への反映状況：真空破壊弁の水没対策

8. 3号機原子炉減圧後の事故進展について

- 3号機の13日9時頃の原子炉減圧以降の事故進展（格納容器ベント、圧力容器・格納容器からの気相漏えい、水素爆発など）や燃料デブリの冷却状態などを推定する上で、格納容器圧力のデータを理解することが重要である。
- 3号機13日9時過ぎの格納容器ベント時におけるS/C水位の推定や、既往の検討に基づき、13日9時～14日0時までの3号機の事故進展シナリオの更なる検討を行った。

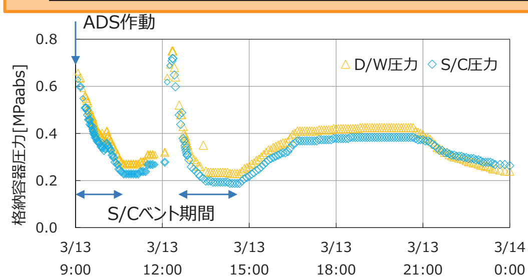


図　3号機ADS作動以降の格納容器圧力

＜検討のアプローチ＞
- 実測値の挙動や、既往の検討の結果等を踏まえて事故進展シナリオを構築。
- 構築した事故進展シナリオについて、実測値の挙動を再現する解析等を通じて定量的な側面からも検討。

検討成果：実測値の傾向を定量的に再現できる事故進展シナリオを推定（以下、主なもの）
- ADS作動とほぼ同時期に、圧力容器からD/Wへの気相漏えいが生じていた可能性。
- ADS作動後から12時頃までの間に、SRV6弁開は維持できなくなっていた可能性。
- 13日16時40分頃にはD/Wから気相漏えいが生じていた可能性。
- 13日20時40分頃からのD/W減圧には、RPV内下部プレナム水の枯渇が影響した可能性。

柏崎刈羽原子力発電所における関連する安全対策
減圧維持機能の強化、格納容器漏えい防止対策

9. 3号機RCIC運転中のプラント状態の検討

- 3号機の津波到達後のRCIC運転では、原子炉水位高でトリップさせないよう、水源のCSTへの戻りラインを活用し、さらに原子炉への注水量を調整することで、運転を継続した。
- この期間の原子炉圧力の挙動は、 RCICの特殊な運転がされる中でSRVが開閉する複雑な状況によるものと認識。
- この定性的な説明の妥当性を確認するため検討を行った。

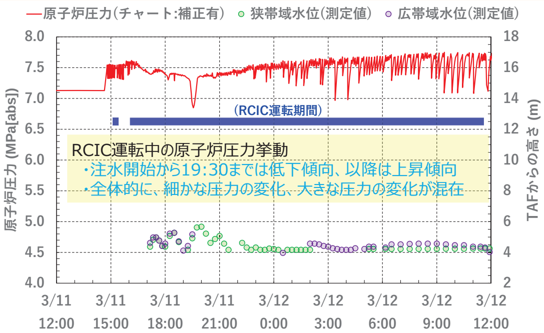


図 RCIC運転期間中の原子炉圧力及び原子炉水位

<検討のアプローチ>

- 当該期間の状況確認及び検討
  - ➢ RCICの動作実績と運転員の操作方法
  - ➢ 崩壊熱（エネルギーバランス）
  - ➢ 予想されるSRVの開モード
- 原子炉圧力の再現解析

- RCICによる原子炉への注水及びSRV開閉を模擬した再現解析を通じて、この期間のプラント挙動に関するこれまでの認識の妥当性を確認し、下記を示唆する結果を得た。
  - ➢ RCICから原子炉への注水による原子炉圧力の低下
  - ➢ RCICタービンへの供給蒸気だけでは崩壊熱を消費できないためSRV経由の蒸気放出があった（SRVは開ききるところ（全開）まではいかない程度の開放と考えられる）

柏崎刈羽原子力発電所における関連する安全対策：減圧維持機能の強化

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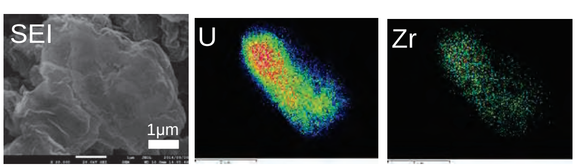
10. サンプル分析による事故状況の把握

- 1～3号機原子炉格納容器（PCV）内外で採取した分析サンプルからウラン（U）含有粒子を検出。
- 環境サンプルからは不溶性セシウム（Cs）粒子が検出され、その組成等と合わせ報告されている。
- これらの放射性微粒子は、事故時の高温の燃料に由来すると考えられ、その生成プロセスがわかれば、生成した時期の原子炉圧力容器（RPV）内環境（温度変化速度、水素/水蒸気比）等の情報が得られる。
- こうした知見は、燃料デブリの状態や事故進展過程の理解に活用する。



2号機オペフロ養生シート上のU含有粒子
SEM/WDSによる、SEI（二次電子像）、元素マッピング（U,Zr）

<検討のアプローチ>
放射性微粒子の生成プロセスを検討する。
（1）U含有粒子に着目した分析
- 燃料成分の混合状態について、サンプル間のU同位体比の分布から評価。
- U含有粒子の組成、結晶構造に着目し、粒子の生成プロセスを推定。Uが溶融凝固過程、蒸発凝縮過程のいずれを経たかによって分類。

（2）不溶性Cs粒子に関する検討
- 球形の不溶性Cs粒子の生成プロセスを推定。

<燃料デブリの状態にかかる知見>
- 滞留水中のα汚染源の多くは粒子状で存在し、ろ紙で9割以上除去できるものであった。Uは立方晶UO_2の形で化学的に安定であり、経年で変化する可能性は小さい。
- 試料中のU同位体比（U235/全U）にかかる分析の結果から、燃料溶融によりU同位体の混合が進んだものと考えられる。

<事故進展にかかる知見（放射性微粒子の生成プロセス評価より）>
- RPV/PCV内の化学的環境（水素／水蒸気比等）が時間や場所に応じて変化したと考えられる結果を得た。
- 1号機では水素が多い環境で生成したと考えられる粒子を確認しており、当該粒子は事故初期原子炉への注水が十分でなかったことと関連する可能性がある。
- 2号機では水蒸気が多い環境で生成したと考えられる粒子と、水素が多い環境で生成したと考えられる粒子を確認。不溶性Cs粒子の生成時期は、燃料の温度上昇初期と考えられ、生成時のRPV内環境の手がかりになると考えている。

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Thank you for your attention

C.2.3.5. Update on 1F1 Mid-and-Long-Term Investigation Plan

Update of Mid-and-Long-Term Plan for the Fukushima Daiichi Nuclear Power Station Accident Investigation

TEPCO

November 18, 2022
Kenji OWADA
Tokyo Electric Power Company Holdings, Inc.

Background

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- **Continuous acquisition of information** contributing to clarification of accident development ⇒ **deepening understanding** of accident progression and further **improving safety** of power reactors

- While steadily proceeding with decommissioning work, the site situation changes and valuable information may be lost ⇒ proceeding with on-site work after organizing and sharing items important from the viewpoint of accident analysis and investigation

Objective

- Formulation of **Mid-and-long-Term Plan for the 1F Accident Investigation** in order to **systematically and initiatively** proceed with the **investigation** in the future

Timeline of publication

- November 25, 2021 – Publication of the original plan
- August 25, 2022 – **Update of the plan and summary of investigation progress**

1

Outline

The plan is formulated and implemented as follows

- 4 categories:
Immediate (~1 year), **short term** (~3 y), **mid term** (~10 y) and timing **TBD**

- Accident investigation **items** sorted **by area** (R/B, PCV, RPV)

- Collection and organization of following **input information** appropriately reflected

  - ✓ D&D work significantly affecting investigation (removal of equipment, etc.)
  - ✓ Needs of internal/external stakeholders
  - ✓ Our external commitments

- Important investigations **actively planned** regardless of their relevance to D&D

- **Revised according to D&D progress** (consolidation of published results of investigation activities)

- Useful information may be also obtained through D&D work not listed in this document

2

Progress

- Efforts in FY2021
  - ✓ Units 1/2 R/B upper floors investigation : Completed
  - ✓ Units 1/2 bottom part of exhaust stack removal (and SGTS piping removal) : Ongoing (delaying original plan)
  - ✓ Unit 1 PCV internal investigation : Ongoing
  - ✓ Unit 2 shield plug perforations investigation : Completed
  - ✓ Unit 2 R/B stagnant water drawdown : Completed
  - ✓ Unit3 MSIV room stagnant water analysis : sampling completed, analysis ongoing
  - ✓ Unit 1~3 remaining gas examination and investigation : Ongoing
  - ✓ Unit 1~4 SGTS room investigation and filters radionuclide analysis : investigation completed

- Efforts in FY2022. (see yesterday's presentation)
  - ✓ Unit 2 FHM remote control room investigation before dismantling : Completed

- Reflecting the obtained findings for improving the safety of the nuclear reactors
  - ✓ Remaining gas confirmed in Unit 3 RHR piping
    ⇒additional procedures (such as venting operations of the system after use)

3

Unit 1 ▼ : Important D&D Step ☆ : Internal/external stakeholders needs, external commitments **Document 1**

			～2022	～2024	～2033	Timing TBD
R/B	Whole bldg.	Each area/eq./bldg. state and dose rate investigation (RB-3a,b,8) ☆		Unit 1 R/B upper floors investigation	▼ Unit 1 indoor/outdoor environment improvement	
		Nuclide analysis of indoor samples (RB-5,9a,14)	Implemented as appropriate			・PCV leak location investigation (RB-10)
		Investigations of penetrations for electrical conduits etc. (RB-9b)		Unit 1 R/B upper floors investigation		
		Stagnant water analysis (TRB-1)	▼ Unit 1 stagnant water analysis			
	O.F. ☆	※Included in general bldg. work			▼ Unit 1 rubble removal ▼ Unit 1 O.F. decontamination/shielding	
	Torus room					
	Specific equipment /systems	RCW investigation (RB-15)		Unit 1 R/B upper floors investigation		・HPCI investigation (RB-2) ・MSL investigation (RB-13) ・Instrumentation soundness investigation (TRB-9) ☆
		AC investigation (TRB-6)		Unit 1 R/B upper floors investigation		
		Remaining gas examination and investigation (TRB-11)				
T/B, yard	-	Vent line, SGTS investigation (RB-11) 【completed】				・SW investigation (TRB-10) ☆
		Units 1/2 exhaust stack FP nuclide analysis (TRB-7)	Units 1/2 SGTS piping removal	Units 1/2 bottom part of exhaust stack removal	▼ Units 1/2 bottom part of exhaust stack removal	
PCV	General	PCV liner, pedestal, etc. state confirmation, FP nuclide analysis (PC-3,9,10,16,RB-9b)	Unit 1 PCV internal investigation	▼ Unit 1 PCV internal investigation		
	Debris, sediments	Investigation of FD, sediments, etc. (PC-3,15,17,18,20,22)	Unit 1 PCV internal investigation	▼ Unit 1 PCV internal investigation		
	Specific equipment/ systems	PLR investigation (PC-4,11)	Unit 1 PCV internal investigation	▼ Unit 1 PCV internal investigation		・IC investigation (PC-2) ・MSL, SRV investigation (PC-5,6) ☆ ・Conduit cables/liner survey (PC-14)
		RPV instrumentation investigation (PC-7,8)	Unit 1 PCV internal investigation	▼ Unit 1 PCV internal investigation		
	RPV itself, peripheral	RPV, peripheral piping state confirmation (PC-3,12,13) ☆	Unit 1 PCV internal investigation	▼ Unit 1 PCV internal investigation		
RPV	-	※Consideration of the content and timing of the investigation based on the progress of D&D				

※This information is subject to change depending on the status of consideration and the progress of the investigation. 4

Unit 2 (1/2) ▼ : Important D&D Step ☆ : Internal/external stakeholders needs, external commitments **Document 1**

			～2022	～2024	～2033	Timing TBD
R/B	Whole bldg.	Each area/eq./bldg. state and dose rate investigation (RB-8) ☆		Unit 2 R/B upper floors investigation	▼ Unit 2 building environment improvement	・Stagnant water analysis (TRB-1)
		Nuclide analysis of indoor samples (RB-5,7,9a,14)	Implemented as appropriate			
		Investigations of penetrations for electrical conduits etc. (RB-9b)		Unit 2 R/B upper floors investigation		
	O.F. ☆	O.F. investigation (RB-4) ※Included in general bldg. work	Unit 2 FHM remote control room investigation before dismantling	▼ Unit 2 FHM remote control room dismantling/existing objects removal ▼ Unit 2 O.F. decontamination, shielding		
	Torus room	PCV leak location investigation (RB-10) (No openings are confirmed up to approx. T.P. -2800mm) ▼	Unit 2 R/B water level lowering (completed)			・PCV leak location investigation (RB-10) ・Torus room investigation (TRB-8)
		Torus room investigation (TRB-8) ▼	Unit 2 R/B water level lowering (completed)			
	Specific equipment/ systems	MSL investigation (RB-13)		Unit 2 R/B upper floors investigation		・RCIC investigation (RB-1,TRB-2,5) ・HPCI investigation (RB-2) ☆ ・Instrumentation soundness investigation (TRB-9)
		AC investigation (TRB-6)		Unit 2 R/B upper floors investigation		
		HPCI investigation (RB-2)	Unit 2 R/B basement floor investigation			
		Remaining gas examination and investigation (TRB-11)				

※This information is subject to change depending on the status of consideration and the progress of the investigation. 5

Unit 2 (2/2)

▼ : Important D&D Step ☆ : Internal/external stakeholders needs, external commitments

			～2022	～2024	～2033	Timing TBD
T/B, yard	-	Vent line, SGTS investigation (RB-11) 【completed】			▼ Unit 2 building environment improvement	・SW investigation (TRB-10) ☆
T/B, yard	-	Units 1/2 exhaust stack FP nuclide analysis (TRB-7)	Units 1/2 SGTS piping removal	Units 1/2 bottom part of exhaust stack removal	▼ Units 1/2 bottom part of exhaust stack removal	
PCV	Genaral	Pedestal state confirmation, FP nuclide analysis (PC-16)		▼ Unit 2 PCV internal investigation Unit 2 PCV internal investigation		PCV liner, etc. state confirmation, FP nuclide analysis (PC-3,9,10,RB-9b)
PCV	Debris, sediments	Investigation of FD, sediments, etc. (PC-3,22)		▼ Unit 2 trial debris retrieval Unit 2 trial debris retrieval, PCV internal investigation	▼ Unit 2 property analysis ▼ Unit 2 debris retrieval, property analysis Unit 2 small scale debris retrieval	・PCV internal water sampling (PC-15)
PCV	Specific equipment/ systems					・PLR investigation (PC-4,11) ・MSL、SRV investigation (PC-5,6) ☆ ・Conduit cables/liner survey (PC-14) ・RPV instrumentation investigation (PC-7,8)
PCV	RPV itself, peripheral piping					RPV, peripheral piping state confirmation (PC-3,12,13) ☆
RPV	Inside RPV	RPV internal state confirmation (RPV-2b,3,4a,4b)		Unit 2 RPV internal investigation	▼ Unit 2 PCV internal investigation	・RPV internal investigation (RPV-1,2a,3,4c,5)

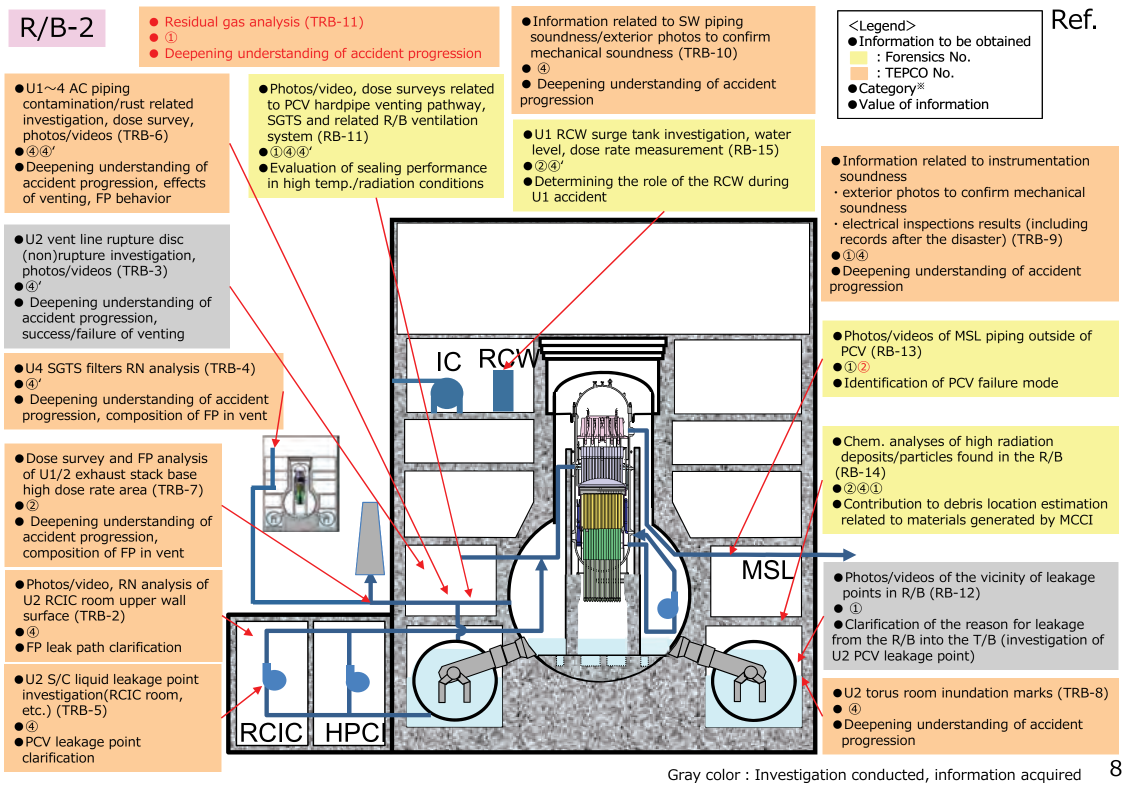
※This information is subject to change depending on the status of consideration and the progress of the investigation.

Unit 3

▼ : Important D&D Step ☆ : Internal/external stakeholders needs, external commitments

			～2022	～2024	～2033	Timing TBD
R/B	Whole bldg.	Each area/eq./bldg. state and dose rate investigation (RB-3a,b,8) ☆		Unit 3 R/B upper floors investigation	▼ Unit 3 indoor/outdoor environment improvement	・PCV leak location investigation (RB-10) ・Investigations of penetrations for electrical conduits etc. (RB-9b)
R/B	Whole bldg.	Nuclide analysis of indoor samples (RB-5,9a,14)	Implemented as appropriate			
R/B	Whole bldg.	Stagnant water analysis (TRB-1)	Unit 3 MSIV room stagnant water detailed analysis	▼ Unit 3 MSIV room stagnant water detailed analysis		
R/B	O.F. ☆	※Included in general bldg. work				
R/B	Torus room					
R/B	Specific equipment/ systems	MSL investigation (RB-13)		Unit 3 R/B upper floors investigation		・RCIC investigation (RB-1) ・HPCI investigation (RB-2) ☆ ・Instrumentation soundness investigation (TRB-9)
R/B	Specific equipment/ systems	AC investigation (TRB-6)		Unit 3 R/B upper floors investigation		
R/B	Specific equipment/ systems	Remaining gas examination and investigation (TRB-11)				
T/B, yard	-	Vent line, SGTS investigation (RB-11)	SGTS filter nuclide analysis		▼ Units 3/4 exhaust stack removal	・SW investigation (TRB-10) ☆
PCV	General	Unit 3 PCV internal investigation (PC-21)			▼ Unit 3 PCV internal investigation	・PCV top head area investigation (PC-1) ・PCV liner, pedestal, etc. state confirmation, FP nuclide analysis (PC-3,9,10,16,RB-9b)
PCV	Debris, sediments	Unit 3 PCV internal investigation (PC-21)			▼ Unit 3 PCV internal investigation	Investigation of FD, sediments, etc. (PC-3,15,22)
PCV	Specific equipment/ systems	Unit 3 PCV internal investigation (PC-21)			▼ Unit 3 PCV internal investigation	・PLR investigation (PC-4,11) ・MSL, SRV investigation (PC-5,6) ☆ ・RPV instrumentation investigation (PC-7,8) ・Conduit cables/liner survey (PC-14)
PCV	RPV itself, peripheral ☆	Unit 3 PCV internal investigation (PC-21)			▼ Unit 3 PCV internal investigation	RPV, peripheral piping state confirmation (PC-3,12,13)
RPV	-	※Consideration of the content and timing of the investigation based on the progress of D&D				

※This information is subject to change depending on the status of consideration and the progress of the investigation.



Details of Categorization

TEPCO

① Have to be investigated for D&D and are planned in near future

➢ Investigation items desirable for forensics activities that are included in the D&D plans or the relevant information/results are already available. (Investigation items indispensable for D&D and are planned in near future)

② Have to be investigated for D&D and are planned in near future, but investigation needs to be expanded

➢ Investigation is planned in the vicinity of the forensics target area, but with different purpose/scope, therefore it is necessary to coordinate and request additional investigation. (Investigation items indispensable for D&D and are planned in short or middle term but are to be investigated by plan expansion in order to satisfy forensics needs)

③ No survey planned until the D&D is completed (resources, technical issues, etc.)

➢ Items that do not need to be investigated for D&D purposes, therefore there is no intention of active investigation from the perspective of D&D (No investigation plan until the end of D&D due to no plan to approach the forensics target area within D&D work, lack of resources, etc.).

④ Items that have to be investigated for D&D but are not currently planned (including expansion of the investigation scope when implementing)

➢ There are merits in investigating these items in advance (due to possibility of information loss with time) and investigation is possible if resources are allocated.

④′ was initially ④, but became ① or ② as a result of investigation progress

9

Thank you for your attention

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Important input information ①

Contents in red letter have been updated since the presentation in 2021.

○D&D work significantly affecting investigation (1/2)

No.	D&D work content	Implementation	Target area
1	Unit 1 PCV Internal Investigation	Immediate	PCV (bottom of D/W)
2	Unit 1 indoor/outdoor environment improvement (Dose rate reduction/interferences removal, etc.)	Medium term	Whole R/B, Yard
3	Unit 1 Rubble Removal (Including FHM, overhead crane)	Medium term	R/B O.F.
4	Unit 1 O.F. decontamination/ shielding	Medium term	R/B O.F.
5	Unit 1 stagnant water analysis	Short term	Whole R/B
6	Unit 2 R/B water level lowering (lowering below the S/C RCIC nozzle)	Completed	R/B torus room
7	Unit 2 O.F. decontamination/shielding	Short term	R/B O.F.
8	Unit 2 building environment improvement (Dose rate reduction/interferences removal, etc.)	Short term	Whole R/B
9	Unit 2 FHM remote control room investigation before dismantling	Completed	R/B O.F.
10	Unit 2 trial debris retrieval/internal investigation	Short term	PCV debris, Whole PCV
11	Unit 2 debris property analysis (At the time of trial retrieval)	Short term	PCV debris

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Important input information ①

Contents in red letter have been updated since the presentation in 2021.

TEPCO

○D&D work significantly affecting investigation（2/2）

No.	D&D work content	Implementation	Target area
12	Unit 2 debris retrieval (Gradual expansion of retrieval scope)	Medium term	PCV debris
13	Unit 2 debris property analysis (At the time of gradual expansion of retrieval scope)	Medium term	PCV debris
14	Unit 2 RPV Internal Investigation	Short term	RPV debris
15	Unit 3 MSIV room stagnant water detailed analysis	Immediate	Whole R/B
16	Unit 3 indoor/outdoor environment improvement (Dose rate reduction/interferences removal, etc.)	Medium term	Whole R/B, Yard
17	Unit 3 PCV internal investigation	Medium term	Whole PCV
18	Units 1/2 bottom part of exhaust stack removal	Medium term	Yard
19	Units 3/4 exhaust stack removal	Medium term	Yard
20	Unit 1/2/3 residual gas examination and investigation (actions related to the residual gas confirmed in the Unit 3 RHR piping)	Short term (adjustment in progress)	PCV, R/B

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Important input information ②

Contents in red letter have been updated since the presentation in 2021.

TEPCO

○Needs of internal/external stakeholders (1/2)

No.	Organization	Site investigation item	Implementation	Target area
1	NRA	Units 1/2 SGTS filter train, AC system contamination	(TBD)	R/B specific equipment/system
2	〃	Dose rate measurement of vent lines, exhaust stack, etc.	Immediate	R/B specific equipment/system
		Dose rate measurement of vent lines, SGTS, etc.※1	Completed	
3	〃	Contamination distribution of R/B HVAC ducts, etc.	(TBD)	Whole R/B
4	〃	Contamination state of Units 1～3 shield plugs bottom sides, reactor wells, etc. (additional contamination investigation)※2	Short term	R/B O.F.
5	〃	Damage conditions investigation by 3D laser scanner	Short term	Whole R/B
6	〃	Contamination state inside R/B	Immediate	Whole R/B
7	〃	Shield plugs deformation investigation	(TBD)	R/B O.F.
8	NRA／TEPCO	Experiment to confirm generation of organic compounds by heating of cables, etc. (Confirmation of In-PCV combustible gases generation)※3	Immediate	R/B specific equipment/system

※1 : Already conducted in FY2020
※2 : Unit 2 shield plug investigation was conducted in cooperation with NRA.
※3 : Experiments on 3 types of cables, 1 type of paint, and 2 types of insulation materials were conducted by the end of FY2021. Experiment on one type of paint will be conducted by the end of FY2022.

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Important input information ②

Contents in red letter have been updated since the presentation in 2021.

TEPCO

〇Needs of internal/external stakeholders (2/2)

No.	Organization	Site investigation item	Implementation	Target area
9	TEPCO	Unit 1 R/B 2FL on-site reactor pressure gauge soundness	(TBD)	R/B specific equipment/system
10	〃	Units 1~3 SRV state Confirmation	(TBD)	R/B specific equipment/system
11	〃	Unit 1 T/B basement (check for damage to water circulation, auxiliary cooling, D/G cooling systems piping, etc. due to seismic motion)	(TBD)	T/B
12	〃	Possibility of leaks from RPV flanges	(TBD)	PCV RPV body

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Important input information ③

TEPCO

〇 Our external commitments (1/2)

No.	Partner/ setting	Site investigation item	Implementation	Target area
1	Our view on the NRA interim report	Expansion of knowledge of vent gas inflow route and mechanism via analysis of samples collected in SGTS filter trains investigations at off-site analysis facilities	Short term	R/B specific equipment/system
2	〃	Sorting of information related to discharge routes of hydrogen leaks, etc. and continuous expansion of knowledge based on information obtained from site investigations in accordance with the progress of D&D work.	(TBD)	Whole R/B

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Important input information ③

Contents in red letter have been updated since the presentation in 2021.

○ Our external commitments (2/2)

No.	Partner/ Setting	Site investigation item	Implementation	Target area
3	Document “Our view on the NRA interim report”	The following items are some of the field studies that may contribute to the expansion of vent gas inflow pathways and mechanisms, but will also be discussed in terms of feasibility.		
		Dose at the high-dose area near the root of the exhaust stack in Units 1/2	Medium term	R/B specific equipment/system
		Radionuclide analysis at the high-dose area near the root of the exhaust stack in Units 1/2※1	Completion	
		Contamination and rust investigation at Unit 1~4 AC piping (dose and photos/movies)	Completion	
		Dose and photos/movies of hardened vent path and SGTS investigation	Completion	
		Dose and photos/movies of R/B ventilation system	(TBD)	
		Unit 1/2 SGTS piping investigation after removal	Immediate	
4	Document “Partial removal of Unit 1/2 SGTS piping”	During the smear measurement, the orientation (location information) in the pipe should be recorded to confirm the effect of aerosol deposition.	Short term	Unit 1/2 SGTS piping

Unit 4

▼ : Important D&D Step ☆ : Internal/external stakeholders needs, external commitments

Document 1

			～2022	～2024	～2033	Timing TBD
R/B	Whole bldg.	Nuclide analysis of indoor samples (RB-5,6)	Implemented as appropriate			Each area/eq./bldg. state and dose rate investigation (RB-3C、8) ☆
	Specific equipment/ systems	SGTS filters nuclide analysis (TRB-4)				AC investigation (TRB-6)
T/B, yard	-				▼ Units 3/4 exhaust stack removal	

※This information is subject to change depending on the status of consideration and the progress of the investigation.

R/B-1

Ref.

<Legend>
● Information to be obtained
: Forensics No.
: TEPCO No.
● Category※
● Value of information

※ :
①Have to be investigated for D&D and are planned in near future
②Have to be investigated for D&D and are planned in near future, but investigation needs to be expanded
③No survey planned until the D&D is completed (resources, technical issues, etc.)
④Items that have to be investigated for D&D but are not currently planned (including expansion of the investigation scope when implementing)
④'was initially ④, but became ① or ② as a result of investigation progress

● Photos/videos of U1,U3 damaged walls, const. mat. (RB-3a, 3b)
● ②①④④'
● Explosion mechanism clarification

● U2 concrete nuclide analysis (RB-7)
● ①②
● RN residing mechanism, analysis models improvement

● Photos/videos of U2 walls (BOP, etc.)/equipment (RB-4)
● ①②④④'
● H_2 generation, depressurization cause

● Photos/videos of U4 damaged walls, const. mat. (RB-3c)
● ①
● Explosion mechanism clarification.

● U4 HVAC duct RN analysis (RB-6)
● ①④'
● U4 R/B explosion cause (H_2 source) clarification.

● Photos/videos of HPCI (RB-2)
● ④
● HPCI op. status in SA clarification.

● Photos/videos of RCIC (RB-1)
● ④
● RCIC op. status in SA and U2 RCIC trip reason clarification

● Photos/videos of places possibly affected by earthquake (RB-8)
● ①②④'
● Clarification of impact of earthquake on structures, equipment, etc.

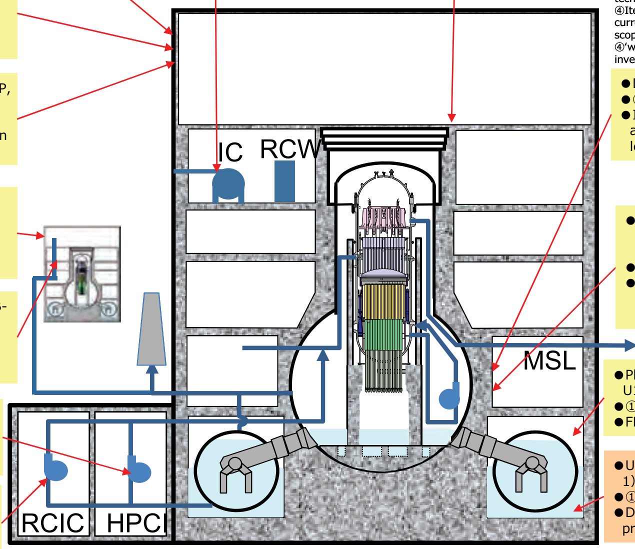
● RN analyses of samples from RBs (RB-5)
● ①②④'
● Leak path identification, dose evaluation benchmarking

● D/W concrete RN analyses (RB-9a)
● ④
● Identification of FPs' leaked amounts and leakage points from PCV, H_2 leakage points clarification

● Photos/videos of mech. seals, hatches, el. penetration seals. Dose surveys (RB-9b)
● ②①④'
● Identification of FPs' leaked amounts and places from PCV, H_2 leakage points clarification

● Photos/videos of U1vacuum breaker, U1~3 leakage points (RB-10)
● ①④
● FP, H_2 leak path clarification

● U1~3 stagnant water analysis (TRB-1)
● ①④
● Deepening understanding of accident progression



18

PCV-1

Ref.

<Legend>
● Information to be obtained
: Forensics No.
: TEPCO No.
● Category※
● Value of information

● RN analyses/sampling, photos/videos of IC (PC-2)
● ①④③
● Seismic damage, final valve position evaluation, insights about hydrogen transport

● Photos/videos of RPV lower head, structures, penetrations (PC-3e)
● ①②
● Damaged around the bottom of RPV, code evaluation of corium behavior, model improvement

● Photos/videos of PCV coatings (both D/W and S/C) (PC-9)
● ①②③④
● Clarification of the effects on coating

● Photos/video, RN analyses, sampling of pedestal walls and floors (PC-3c)
● ①②④
● Benchmark of code evaluation for RPV damage, morphology and composition of ex-vessel debris and MCCI estimation

● PCV liner examinations, photos/videos (vicinity of debris, U1 pedestal), metallurgical exams (PC-3b)
● ②(Metallurgical exams③)
● Improvements of models predicting liner failure and MCCI

● Photos/videos of debris, crust relocated to PCV, hot cell exams (PC-3a)
● ①②
● Obtaining knowledge on the amount, height, morphology, composition distribution, spreading, splashing and salt effects, etc. of fuel debris.

● Records of PCV head flange tightening, torque, bolt length. PCV head flange photos/vide (PC-1)
● ①④
● Clarification of how the PCV head lifted, peak temperatures, deterioration due to high temperature

● Examinations and operability assessments of ex-vessel sensors and sensor support structures (PC-8)
● ①②③
● Identification of vessel depressurization paths, clarification of the cause of failure of RPV pressure system B

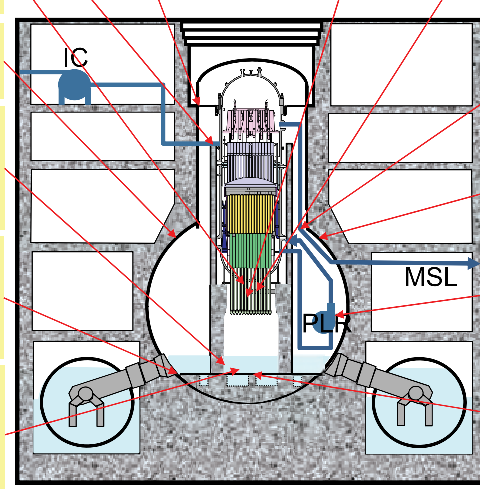
● Ex-vessel inspections and operability of in-vessel sensors and sensor support structures (RPV bottom head, U2 TIP, SLC) (PC-7)
● ①②③
● Identification of vessel depressurization paths, clarification of the cause of failure of RPV pressure system B

● Photos/videos of MSL, ADS lines to end of SRV tailpipes, including instrument lines (PC-5)
● U1③ 、U2,3 MSL②(ADS・SRV③)
● Investigation of RPV failure mode

● Visual inspections of SRV and MSL, interior valve mechanisms (PC-6)
● U1③ 、U2,3 MSL② (ADS・SRV③)
● Damage investigation of piping related to SRV

● Photos/videos of PLR lines and pumps (if molten fuel accumulates outside the shroud, it may enter PLR lines) (PC-4)
● ②③①
● Investigation of PCV failure mode and relocation path

● Concrete erosion profile, photos/videos, sampling and examination (PC-3d)
● ①②③④
● Benchmarking of MCCI code predictions



19

PCV-2

Ref.

<Legend>
●Information to be obtained
: Forensics No.
: TEPCO No.
●Category※
●Value of information

●RN analyses inside PCV (PC-10)
●①② (U3④)
●Dose code assessments, model improvements

●Samples of conduit cabling and paint for RN analyses (PC-14)
●③ (②)
●Dose code assessments, model improvements

●Analysis results of the black material from U2 CR exchange rail (including form, microstructure chemical composition) (PC-19)
●①
●Obtaining knowledge about the peak temperature of the structure and presence/absence of damage. Model improvement

●Photos/videos of melted, galvanized or oxidized structures (in and out of pedestal) (PC-16)
●①②③
●Obtaining knowledge about the peak temperature

●RN analyses of water from PCV (PC-15)
●①②
●Dose code assessments, model improvements

● Analysis of the black material on the structure from U1 D3 location (including form, microstructure chemical composition) (PC-20)
●③ (②)
●Estimation of presence/absence of MCCI based on the Si and debris content. Model improvement

●Photos/videos of insulation around piping and the RPV (PC-13)
●③ (②)
●Investigation of adverse effects on long-term cooling due to insulation deterioration

●Photos/videos of TIP, SRM, RIM piping outside of the RPV (PC-12)
● ①② (U3④)
●To determine if failure of TIP tubes and SRM/IRM tubes outside of the RPV led to depressurization

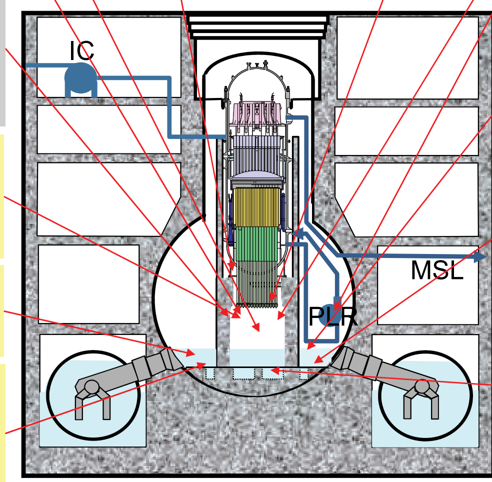
●Images from U3 PCV internal investigations (PC-21)
●①②
●Model improvement

●Photos/videos of PLR pump seal, etc. and potential RPV leak path (PC-11)
●③
●Performance assessment under high temperature/high pressure conditions

●Chem. analysis of sediments on U1 D/W floor, axial composition (PC-17)
●①②
●Estimation of presence/absence on MCCI based on the presence/absence of concrete oxides. Model improvement

●Properties of materials under the U1 D/W floor sediments (PC-18)
●②③
●Clarification of the material properties (possible difference with upper layer, possible debris). Estimation of presence/absence on MCCI based on the presence/absence of concrete oxides. Model improvement

●Debris samples analysis results at different axial and radial positions (including form, microstructure, chemical composition) (PC-22)
●② (③)
● Estimation of presence/absence on MCCI based on the presence/absence of concrete oxides. Knowledge about molten and relocated substances. Knowledge of fuel enrichment. Model improvement



Gray color : Investigation conducted, information acquired

20

RPV

Ref.

<Legend>
●Information to be obtained
: Forensics No.
: TEPCO No.
●Category※
●Value of information

●Photos/videos, sampling of separator (RPV-3)
●② (sampling③)
●Code evaluation and model improvement based on findings from soundness or displacement measurement, peak temperature evaluation, and metallurgical testing

●Photos/videos, samples of dryer (RPV-1a)
●②(sampling ③)
●Code evaluation and model improvement based on findings from soundness or displacement measurements, peak temperature evaluation, and metallurgical testing

●Photos/videos of MS piping, probe survey in piping, sample collection (RPV-1b)
●②④ (sampling, metallurgical exams ③)
●Code evaluation and model improvement based on knowledge about deformation, metallurgical tests, etc.

●Photos/videos, sampling of FDW sparger nozzle, water injection points (RPV-2b)
●② (sampling③)
●Evaluation of operation performance, effect of seawater injection including corrosion

●Photos/videos, sampling of upper structures, upper channel guides (RPV-1c)
●② (sampling, metallurgical exams③)
●Code evaluation based on deformations, etc., metallurgical exams and improvement of models predicting peak temperature, displacement and melting

●Photos/videos, sampling of CS nozzle, sparger, nozzle joints (RPV-2a)
●② (CS in shroud②、out of shroud②, sampling, metallurgical exams ③)
●Evaluation of operation performance, effect of seawater injection including corrosion

●Photos/videos of core support plate and related structures (RPV-4d)
●② (sampling, metallurgical exams③)
●Code evaluation and model improvement

●Photos/videos, sampling of shroud head (RPV-4b)
●② (sampling, metallurgical exams③)
● Code evaluation and model improvement based on findings obtained from soundness or displacement measurement and metallurgical tests

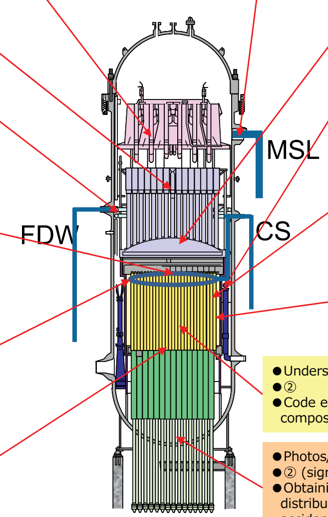
●Photos/videos, sampling of shroud (space between the shroud and RPV wall) (RPV-4a)
●② (sampling, metallurgical exams③)
●Code evaluation and model improvement based on findings obtained from soundness or displacement measurement and metallurgical tests

●Photos/videos, sampling of shroud (from core region) (RPV-4c)
●② (sampling, metallurgical exams③)
●Code evaluation and model improvement based on findings obtained from soundness or displacement measurement and metallurgical tests

●Understanding the state of the core (and between the RPV and shroud) using remote mapping (RPV-5a)
●③※1
●Code evaluation and model improvement

●Understanding the end state of the core and structures (RPV-5b)
●②
●Code evaluation. Improvement of models for prediction of debris composition, mass and morphology

●Photos/videos, sampling of in-vessel debris (TRPV-1)
●② (significant increase in the number of samples③)
●Obtaining knowledge on the mass, morphology, composition distribution, spread, etc. of fuel debris and deepening understanding of accident progression



※1 : Evaluation of debris location using muon tomography
(Unit 1 : February～May 2015, Unit 2 : March～July 2016, Unit 3 May～September 2017)

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① 福島第一原子力発電所の廃炉を安全かつ着実に進めることが最優先の課題であり、結果として、可能な限り速やかな廃炉を実現していくことが必要である。このため、福島第一原子力発電所の廃炉・汚染水対策に関する分析・調査を実施する必要があるが、その分析・調査は廃炉を安全かつ着実に進め得る取組みの範囲の中で実施すること。

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

② 一方で、福島第一原子力発電所の事故原因の究明や今後の原子力に関する安全性向上の観点からの分析・調査の実施も必要とされている。このため、福島第一原子力発電所の廃炉・汚染水対策に関する分析・調査は、福島第一原子力発電所の廃炉を安全かつ着実に進めるために行うことを前提として、事故原因の究明や今後の原子力に関する安全性向上の観点からの必要性を十分に考慮すること。

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

③ 福島第一原子力発電所の廃炉・汚染水対策に関する分析・調査は、地域の皆様、周辺環境及び作業員に対する安全確保を最優先に、現場の作業状況の厳しさを踏まえ、分析・調査の方法を具体化した上で計画すること。

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

① 福島第一原子力発電所の廃炉・汚染水対策に関する分析・調査は、その分析・調査により得られる情報が、何のために使われて、何に貢献するのかを明確にした上で、その意義とそれに伴う負担を熟慮し、廃炉プロジェクトとして合理的に許容できる範囲で行うこと。

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

⑤ 福島第一原子力発電所の事故を起こした我が国の国際社会に対する責任として、福島第一原子力発電所の廃炉・汚染水対策に関する分析・調査で得られた情報の積極的な発信を行うこと。また、それを超える情報を求める機関には、相応の負担を求める可能性があること。

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

C.2.4. NRA Investigations

NRA, Japan
Nuclear Regulation Authority

NRA's Investigation (Phase2) of Fukushima Daiichi Nuclear Accidents (2021~2022)

17th Nov. 2022, Reactor Safety Technology Expert Panel Forensics Meeting

Masaya YASUI
Nuclear Regulation Authority, Japan

NRA, Japan
Nuclear Regulation Authority

Item-1
Damages at the PCV Pedestal of Unit-1

2

TEPCO&IRID took video images inside the unit-1 PCV. Detailed information of the video and related equipment is described in the TEPCO's presentation.

Acknowledgment:
Unit 1 PCV internal investigation results were obtained by using the robot developed in Subsidy Program "Project of Decommissioning and Contaminated Water Management" by METI of Japan.

Configuration of the lower part of PCV

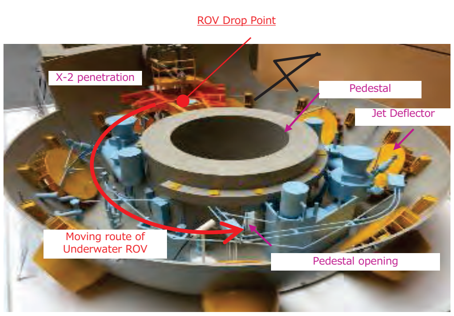


Fig.1 Structure model of lower PCV

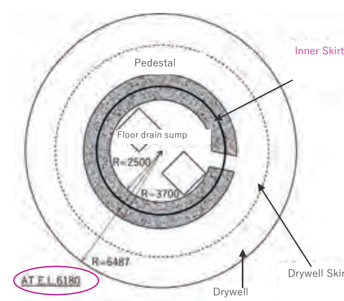
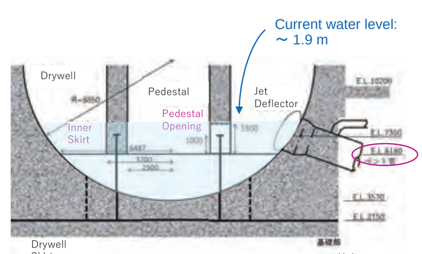


Fig.2 Horizontal cross-section of lower PCV



(source) IRID & Hitachi-GE Nuclear Energy, Ltd.

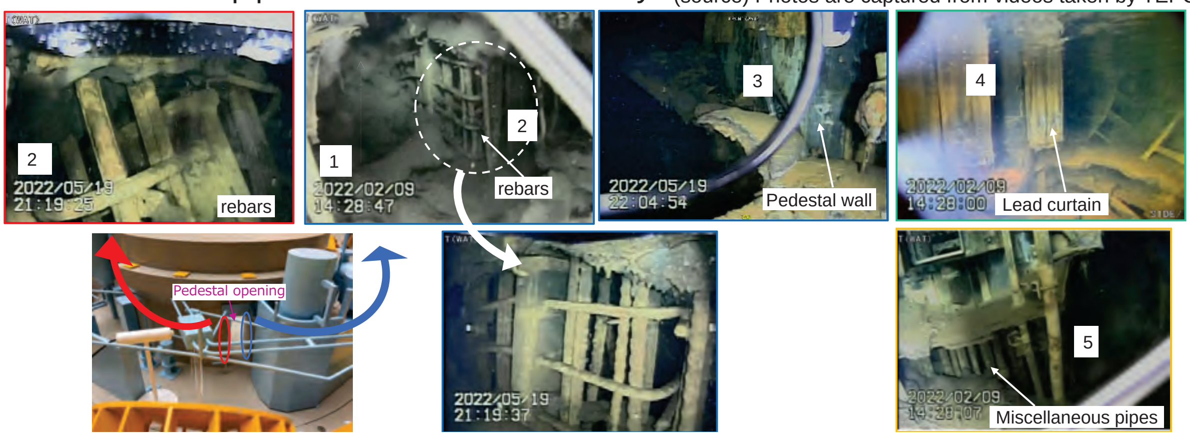
Fig.3 Vertical cross-section of lower PCV

Characteristics of the video images inside the unit-1 PCV

NRA, Japan
Nuclear Regulation Authority

1. Debris mound seems to exist near the pedestal opening. (not clear)
2. Loss of concrete at both sides of the opening. Rebars seems to be intact.
3. Concrete is lost only below the crust.
4. Lead net shield melted at a specific height.
5. Miscellaneous pipes are not affected severely.

(source) Photos are captured from videos taken by TEPCO



Characteristics of the video images inside the unit-1 PCV

NRA, Japan
Nuclear Regulation Authority

6. “Crust” exists from the area of opening to the other side.
7. “Crust” is the highest near the opening (~1.2m) and lowest at the other side (~0.3m).
8. “Crust” includes small bubbles. Its thickness is calculated around 3cm at a place.
9. Underside of the “Crust” seems to be smooth.

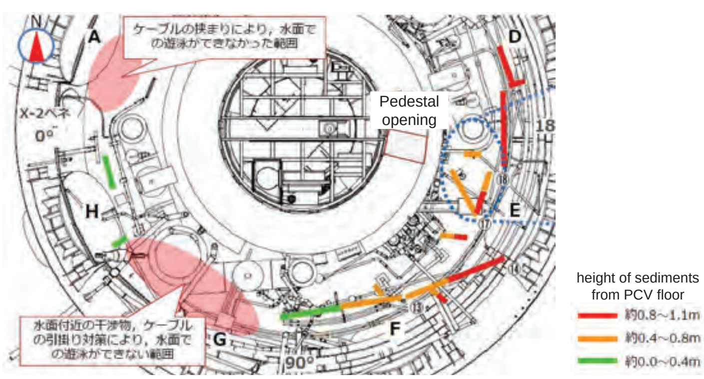


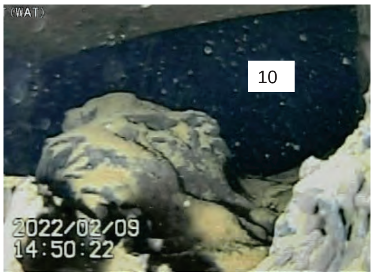
photo taken by TEPCO

(source) “Investigation on the inside of Unit 1 PCV” P.5, TEPCO, 2022 Sept. 6,
https://www.nra.go.jp/data/000403164.pdf

Characteristics of the video images inside the unit-1 PCV

NRA, Japan
Nuclear Regulation Authority

10. Large bubble-like “Crust” exists under the opening ceiling.
    (The thickness is not known.)
11. No Damage is found out at the outer walls of the PCV.
12. Bubble-like image is observed on the surface of debris or crust on the PCV floor.
13. Exact situation of the PCV floor is not known.



(source) Photos are captured from videos taken by TEPCO

7

Major Questions

NRA, Japan
Nuclear Regulation Authority

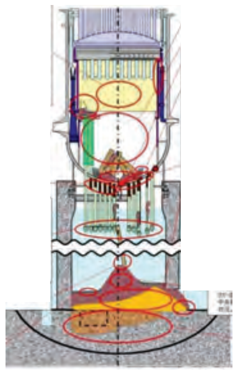
1. Why the debris dropped from the RPV did not spread out?
2. How was the concrete part alone of the pedestal wall damaged?
3. How was the “crust” formed?

8

Q1: Why the debris dropped from the RPV does not spread out?

NRA, Japan
Nuclear Regulation Authority

- If the molten core became metal-rich debris, the dropped molten core can be lower than 1200℃.
- Lower temperature of the dropped debris means higher viscosity.
- In that case, molten core drops to the pedestal floor relatively slowly.



JAEA reports that debris-mound exists to 3m height in Unit 3.

(source of the figure) debris wiki https://fdada-plus.info/wiki/

Q2: How was the concrete part alone of the pedestal wall damaged?

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Nuclear Regulation Authority

Various ideas have been proposed,
but the NRA has not reached any conclusion.

- Melted with high temperature （≧1200℃？）
- Heat Shock
- Silicone extraction （・・・・？）
- Phase changes （～800℃？）

➢ Concrete core sample will be provided by the TEPCO and will be tested.

Q3: How was the “Crust” formed?

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Nuclear Regulation Authority

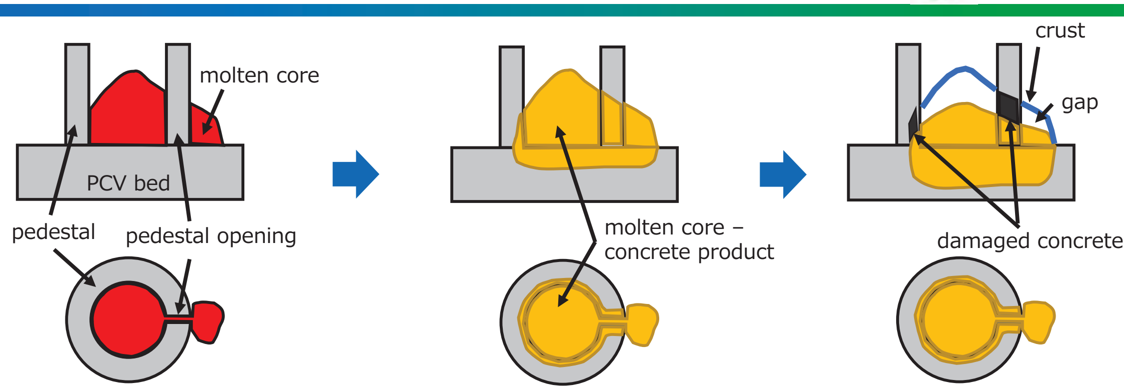
Tentative ideas (just proposed)

- Normal Crust (Osaka Univ. 1)
- Hydrothermal Reaction (Osaka Univ. 2)
- Inflated by gases (NRA)

➢ Any of them may not explain well the observations.
➢ New ideas from international world are welcome.

Image of “Normal Crust Scenario”

NRA, Japan
Nuclear Regulation Authority



- Molten core drop into a pedestal and spread on the floor

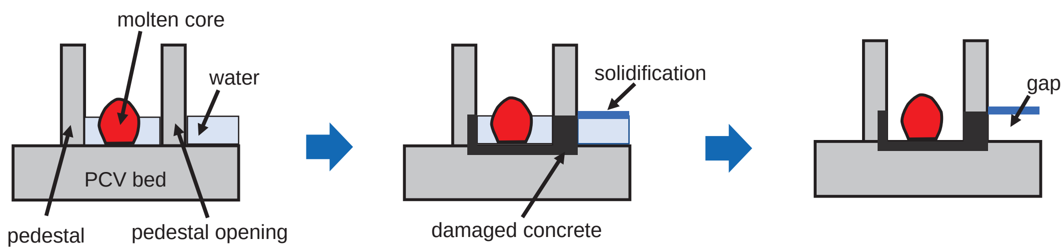
- Molten core and concrete react and form a product
- Concrete is eroded

- “Crust” is generated
- Erosion proceeds

(source) “Research on phenomena of concrete observed by TEPCO’s Fukushima Daiichi NPS Unit 1 PCV inside investigation” P.7, Osaka Univ. graduate school, 2022 Oct. 31, https://www.nra.go.jp/data/000408670.pdf

Image of “Hydrothermal Reaction Scenario”

NRA, Japan
Nuclear Regulation Authority



- Water (or vapor) exists at a pedestal
- Molten core drop into a pedestal
- Water temperature and pressure rise locally

- concrete components dissolve to high pressure water or react with vapor and concrete is damaged
- Concrete components and water form high viscosity liquid phase
- High water temperature and release of local high pressure facilitate evaporation of water and dissolved concrete components are solidified

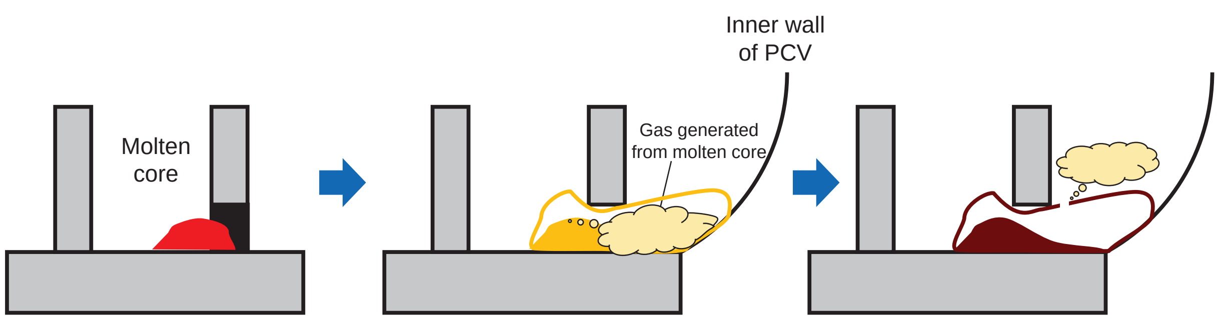
➡ Evaporation proceeds and a gap appears under the solids

(source) “Research on phenomena of concrete observed by TEPCO’s Fukushima Daiichi NPS Unit 1 PCV inside investigation” P.8, Osaka Univ. graduate school, 2022 Oct. 31, https://www.nra.go.jp/data/000408670.pdf

13

Image of “Gas Inflation Scenario”

NRA, Japan
Nuclear Regulation Authority



Gas generated from the molten core and concrete inflated a shell.
(Steam might be the gas)

The gas released from a crack on the shell and the shell remained as “Crust”.

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The NRA expects more information from TEPCO such as:

- Sampling and analysis of the "crust"
- Video image inside the pedestal and inside pedestal-wall
- Video image of under-side of the crust far from the opening
- Sampling and analysis of the material on the PCV floor

etc.

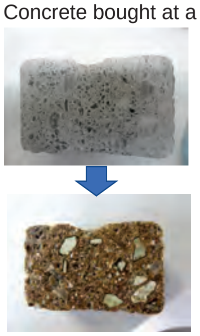
15

Concrete heat-up tests

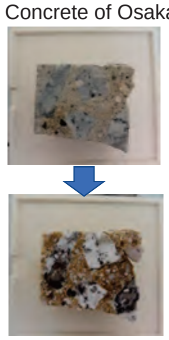
NRA, Japan
Nuclear Regulation Authority

- Osaka University conducted a preparatory experiment heating three types of concrete at 1200 ℃ for 8 hours

1. Concrete bought at a DIY shop



2. Concrete of Osaka Univ. Suita Campus A15 Building



Concrete of a DIY shop (1.) did not change its shape.

Concrete of Osaka Univ. iFEL* Building (3.) melted and changed its shape

*Institute of Free Electron Laser, Graduate School of Engineering, Osaka University

➢ Behavior of concrete when heated depends on each concrete sample

- TEPCO will extract core sample of concrete so that the NRA or other organization will conduct heat-up tests.

(source of analysis results) "Research on phenomena of concrete observed by TEPCO's Fukushima Daiichi NPS Unit 1 PCV inside investigation" P.12, Osaka Univ. graduate school, 2022 Oct. 31, https://www.nra.go.jp/data/000408670.pdf

16

Other Factors

- Water Levels
- Time and amount of injected water
- High Contamination of Reactor Cooling Water System (RCW)
  Piping, Heat Exchangers, Pumps
- Re-flooding attempts
- Indication of Thermo-couples after DC power recovery

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Item-2
How does Cs-137 move at the severe accident conditions?

- Characteristic Pollution Pattern in the SGTS* lines
  (Reported in the previous meeting)

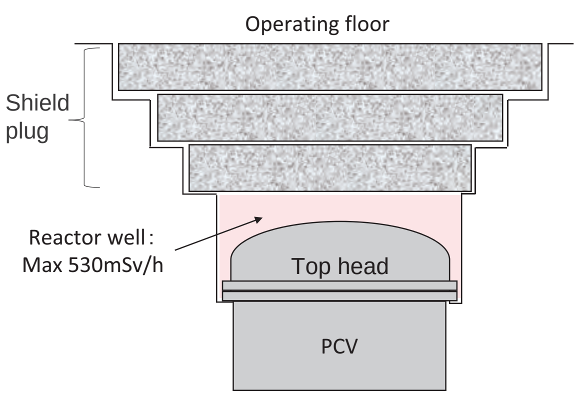
  *SGTS : Standby Gas Treatment System. SGTS lines are used as vent-gas exhaust lines.
- The related computer simulation was studied by the NRA (Dr. Tsukamoto)

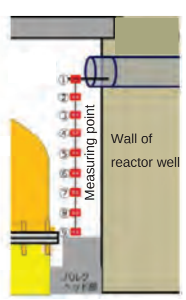
Result

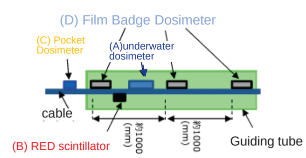
Fundamental mechanism of the pollution pattern could be understood mainly with condensation and condensed water location. (Other mechanisms of aerosol deposition seem not to be dominant.)

Radiation measurement in the Reactor well

Radiation field in the unit-2 reactor-well was measured by TEPCO in February 2021.
The result shows that the radiation level in the reactor-well is less than 530mSv/h.
(Reported in the previous meeting)







Measuring point	Distance (mm)	(A) underwater dosimeter		(B) RED scintillator	(C) Pocket dosimeter	(D) Film Badge dosimeter
		Dose rate (mSv/h)				
		Measured on May 20, 24		Measured on June 23th		
①	0	74.6	-		Approx. 315	Top: 220 Middle: 270 Bottom: 360
②	500	150	-			
③	1000	330	-			
④	1500	300	280	Under lowest limit		
⑤	2000	310	-			
⑥	2500	380	-			
⑦	3000	440	-			
⑧	3500	530 (Max)	370	400±100mGy/h		
⑨	4000	350	-			

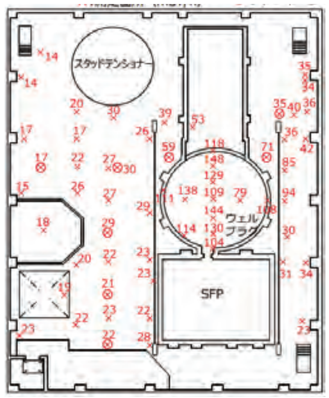
(source of the measurement results on the right side)The Committee of Accident Analysis of Fukushima Daiichi Nuclear Power Station 21th meeting (July 2021, TEPCO) Document 5-3, TEPCO https://www.nsr.go.jp/data/000358693.pdf

Observation result ①

NRA, Japan
Nuclear Regulation Authority

Three facts below almost certainly confirm the previous assumption.
= High contamination exists in the shield plugs.

① Radiation level above the Shield Plug is higher than the neighboring area.



<measurement condition>
- 1.5m height
- Dosimeter
- Without calibration

*Red figure means ambient dose rate (mSv/h)

(source of the measurement results) “Results of investigation on Unit 2 R/B operating floor after clean-up” P.14, TEPCO, 2019 Feb. 28 https://www2.nra.go.jp/data/000270192.pdf

Observation result ②

NRA, Japan
Nuclear Regulation Authority

② Surface contamination is fairly homogeneous on the operation floor including shield plugs.

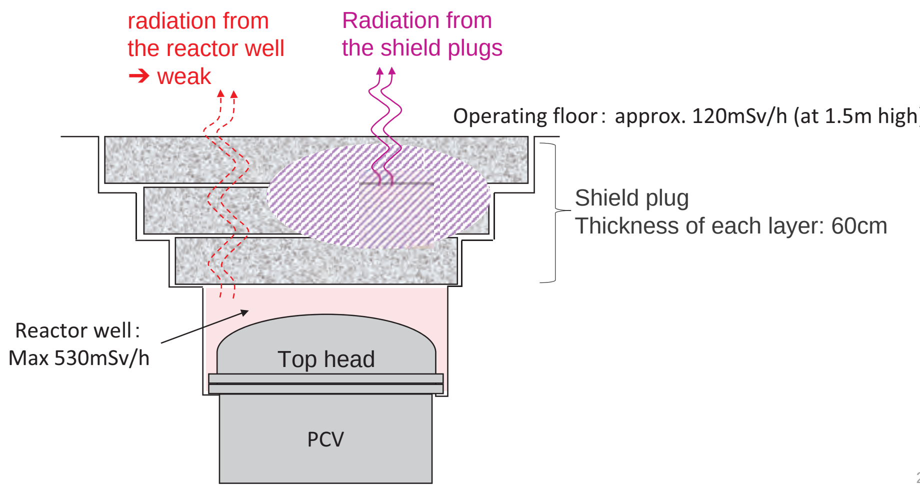
Measurement points		NRF54 (NRA2021.0414測定)	Smear (東電2019.5.20報告)	Gamma ray camera (NRA2020.1.30測定)
		Cs-137汚染密度 Bq/cm^2		
No. 1	West floor	3.3E+04	3.3E+05 位置(8)	−
No. 3		1.2E+04		−
No. 4		3.7E+04		−
No. 6		6.2E+04	6.5E+05 位置 (1)	7.00E+04
No. 7	Shield plug	8.3E+04	6.7E+05 位置 (11)	1.70E+05
No. 8		1.2E+04	6.4E+05 位置 (12)	8.00E+04
No. 9			9.7E+05 位置 (12)	1.70E+05
No. 10			5.1E+05 位置 (15)	1.00E+05
No. 11		1.6E+05	8.2E+05 位置 (13)	1.70E+05
No. 12	East floor	1.0E+05	1.0E+06 位置 (16)	−
No. 13			2.0E+05 位置 (17)	1.70E+04
No. 14		3.6E+04	2.9E+06 位置 (18)	1.70E+05
			()は引用資料内位置番号	東西に長いピクセル内平均の測定値 [−]は視野外

(source of the measurement result) “Result of investigation on Unit 2 operating floor cooperatively conducted with NRA” P.6, TEPCO, 2021 May 18, https://www.nra.go.jp/data/000352402.pdf

Observation result ③

NRA, Japan
Nuclear Regulation Authority

③ Radiation level in the reactor-well is too weak to bring about the radiation field above the shield plugs.



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JAEA's calculation on elastic deformation of the Shield plugs

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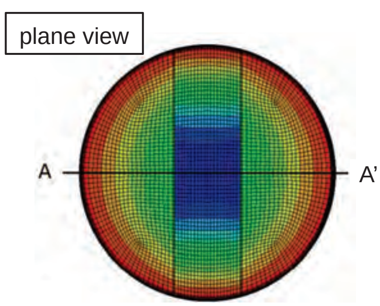
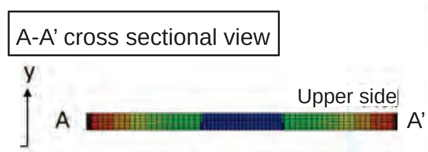
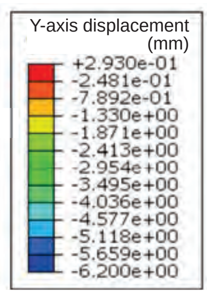




<u>Maximum downward displacement</u>
Central segment: 6.2 mm
Both side segments: 3.7 mm

<u>Properties of reinforced concrete</u>
Density: 2.4×10^{-9} t/mm
Young's modulus: 25700 MPa



<u>Evaluation results</u>

Analytical case	Maximum downward displacement	Rough evaluation result (area of the gap)
Evaluation with physical properties at room temperature	6.2mm	Vertical: 1.8×10^4 mm^2
		Horizontal: 5.2×10^4 mm^2
Reproductive evaluation (with a condition that Young's modulus is 4000 MPa)	40mm	Vertical: 1.1×10^5 mm^2
		Horizontal: 2.0×10^4 mm^2

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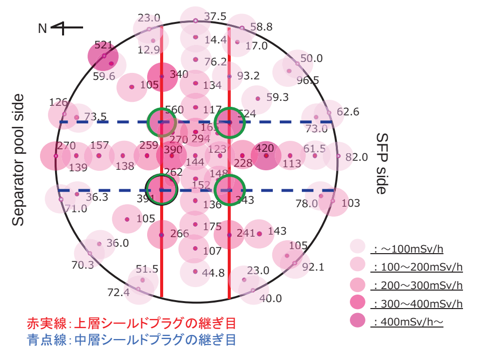
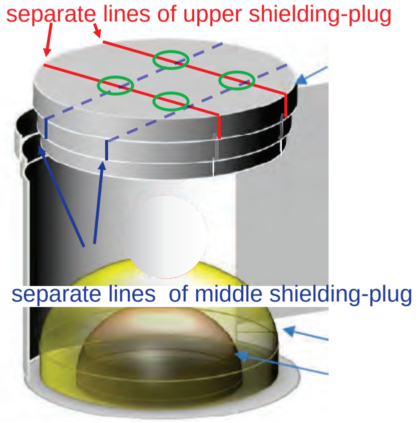
(source) "Evaluation of area of the gap between shield plugs" P.6, P.16 JAEA, 2022 Oct. 31 https://www.nra.go.jp/data/000408675.pdf

The result shows that sufficient flow pass exists at the vertical center lines of the Shield plugs.
These flow pass matches the observed radiation-field pattern.





(source of left figure) https://www.da.nsr.go.jp/file/NR000206348/000367850.pdf P.3 (Oct. 19th, 2021, TEPCO) *NRA added measurement results to each circle
(source of right figure) https://www.nra.go.jp/data/000358693.pdf P.1 (July 8th, 2021, TEPCO)

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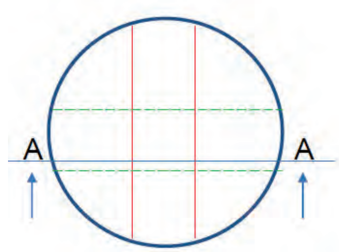
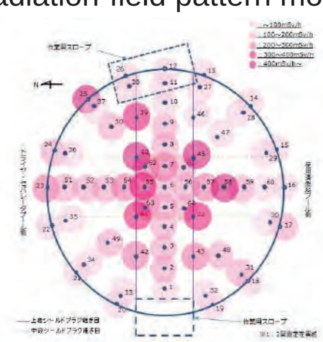
TEPCO's explanation on the flow pass can explain the radiation-field pattern more easily.





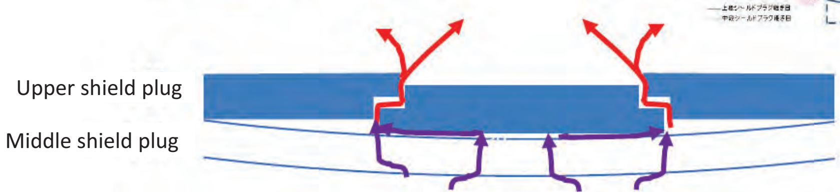


Figure A-A' cross-section flow pass of shield plug gap

(source of figures) "Contamination of Unit 2 Shield plug" P.1, TEPCO, 2022 Oct. 31 https://www.nra.go.jp/data/000408676.pdf

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"Condensation and condensed water location" assumption might be able to applied to wider observations. (Hypothesis)

Exemplary cases:

- High contamination of Unit3 shield plugs (30 PBq : estimated by NRA)
- Low contamination of Unit1 shield plugs (0.1-0.2 PBq : estimated by TEPCO)
- Condensed water and contamination in SGTS filters of Unit 2 and Unit 3
- Very low contamination at the upper half of Unit1&2 shared stack
- High contamination of Unit 1 RCW* heat-exchangers

*RCW: Reactor Cooling Water system

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Thank you for your attention.

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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 17-18, 2022
NEI
Washington, DC

Topics

- Key questions
- Current status – November 2022
- Major observations from 2022 activities
  1. 1F2 Shield Plug Deformation
  2. 1F2 Shield Plug Contamination
  3. 1F1 Reactor Pedestal Wall
  4. Structure Survey - Seismic
  5. Hydrogen Gas - Question

2

Key Questions

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

3

1F2 Shield Plug Deformation

- Compared shape of shield plug at three units
  - 1F2, 1F5, and Shimane 1, Tsuruga 1

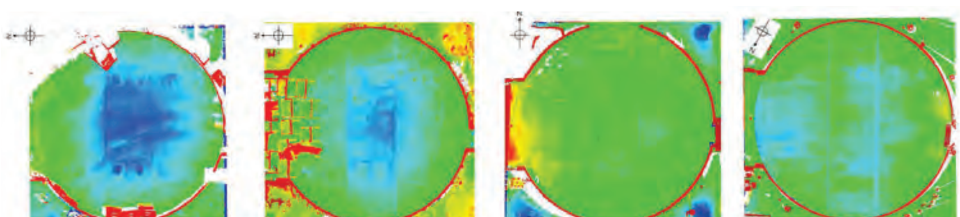


図2-1 1F2号機シールドプラグの変形状況 図2-2 1F5号機シールドプラグの変形状況 図2-3 島根1号機シールドプラグの変形状況 図2-3 敦賀1号機シールドプラグの変形状況

Source: Nuclear Regulation Authority
https://www.nsr.go.jp/data/000382268.pdf and https://www.nsr.go.jp data/000388505.pdf

- Result: 1F2 had the greatest sag in middle shield plug segment
  - This sag could contribute to flow path through shield stack
- No cracks observed in shield plugs at the three units
  - Therefore, deformation likely not due to external forces after fabrication

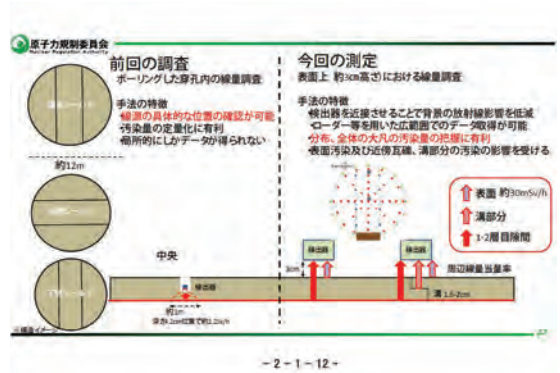
4

1F2 Shield Plug Contamination

- Boring into 1F2 shield plug & taking radiation measurements
  - Goal is to understand contamination distribution below top shield plug

- Measurement appears to be consistent/confirm previous estimate (several tens of PBq of Cs-137) ***Slides(2022-08-25,TEPCO) d220825_13-j***

Source: Nuclear Regulation Authority
https://www.nsr.go.jp/data/000376558.pdf

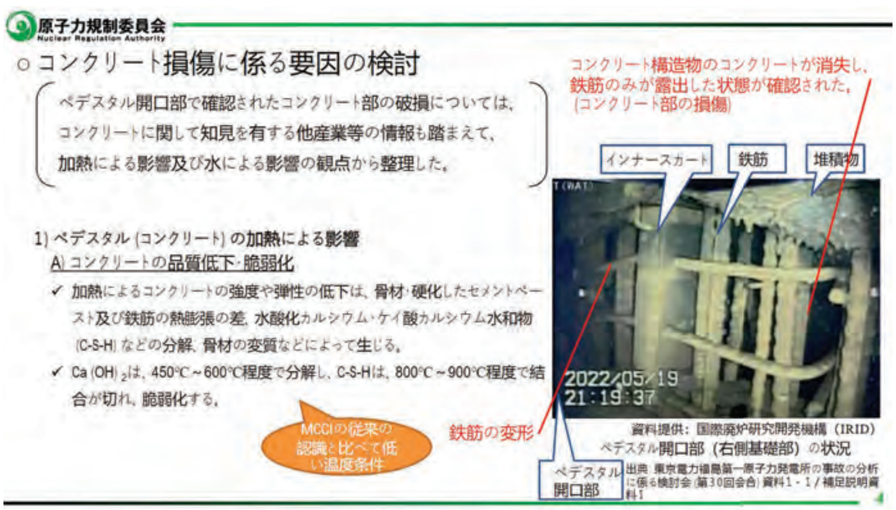


Source: Nuclear Regulation Authority
https://www.nra.go.jp/data/000376551.pdf

5

Unit 1 Reactor Pedestal Wall

- Core-concrete interaction significant near pedestal wall opening.



Source: Nuclear Regulation Authority
https://www.nra.go.jp/data/000403165.pdf

6

Structure Survey - Seismic

- Earthquake 3/16/2022
  - Detailed sequence of site evaluations
- Efforts to survey building structures for damage and impact of seismic activity (after 2011)
  - Visual, point displacement measurements , seismometers
- *K. Robb Impression:*
  - Relatively minor impacts of 3/16 earthquake
  - Did not see a note of impact/change that is of particular interest to the panel

7

Hydrogen Gas - Question

- *We reexamined the possibility of hydrogen gas remaining, focusing on cases similar to the Unit 3 RHR piping where hydrogen gas retention was confirmed in December 2021 (valve operation and water sealing at the time of the accident).*

  See: https://www.tepco.co.jp/decommission/information/committee/roadmap_progress/pdf/2022/d220825_13-j.pd

8

9

TEPCO/IRID 1F1 debris survey

Source: Nuclear Regulation Authority
https://www.nra.go.jp/data/000403164.pdf

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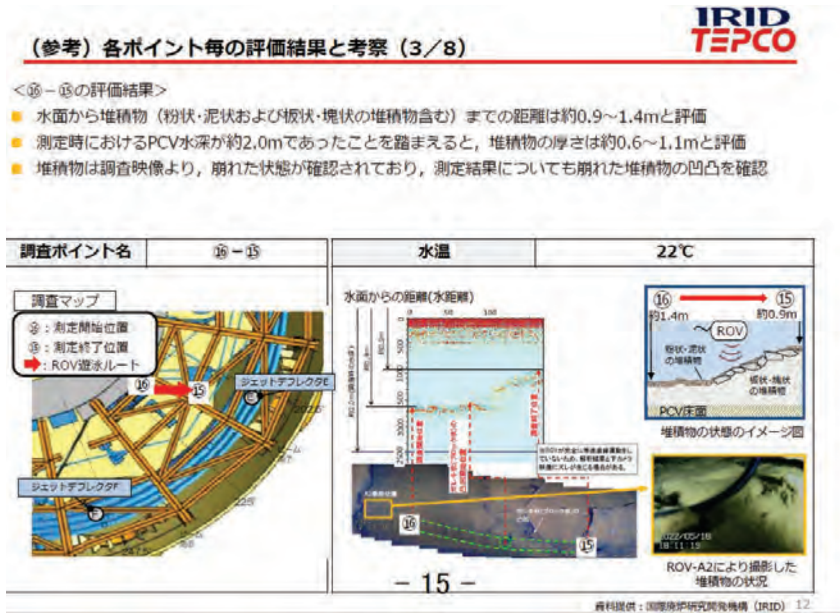
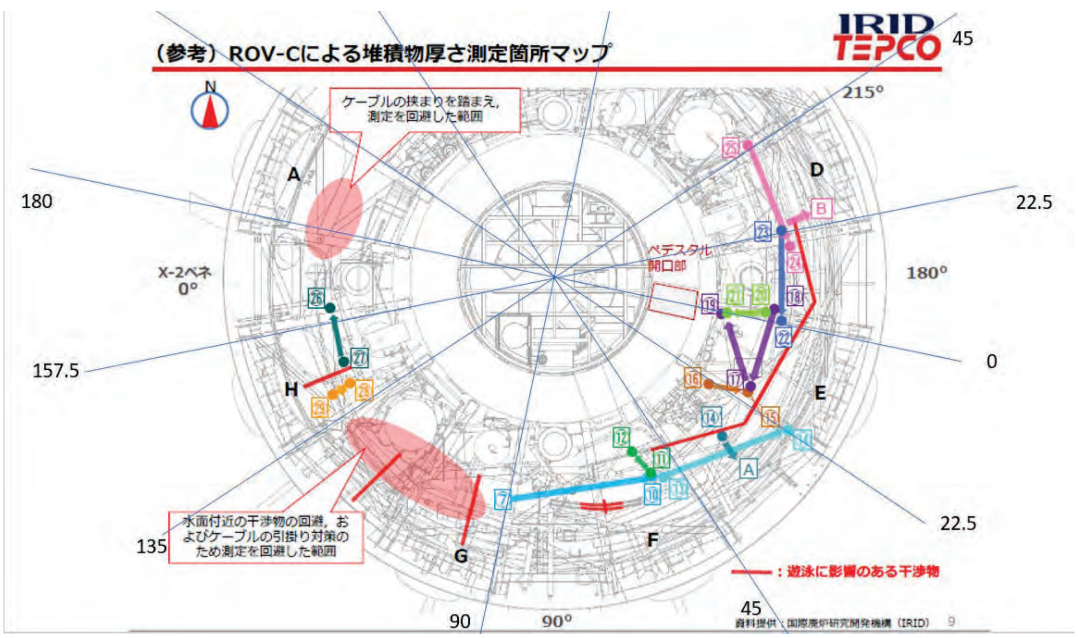
TEPCO/IRID 1F1 debris survey



Source: Nuclear Regulation Authority
https://www.nra.go.jp/data/000403164.pdf

11

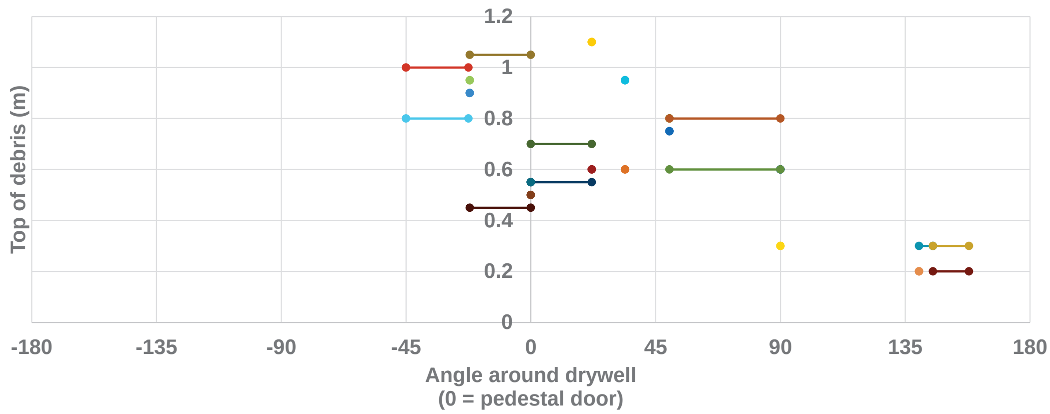
TEPCO/IRID 1F1 debris survey



Source: Nuclear Regulation Authority
https://www.nra.go.jp/data/000403164.pdf

12

Top of Debris - estimates



13

Enhanced Ex-Vessel Analysis for Fukushima Daiichi Unit 1: Melt Spreading and Core-Concrete Interaction Analyses with MELTSPREAD and CORQUENCH
K. Robb, M. Farmer, M. Francis, ORNL/TM-2012/455, Feb 2013.

- MAAP and MELCOR - melt pour scenarios
- MELTSPREAD – corium spreading, concrete ablation
- CORQUENCH – concrete ablation/debris cooling

Table 11. Initial Collapsed Melt Thickness

Case	Initial Collapsed Melt Thickness (cm) in each Region					
	1 Sumps	2 Inner Pedestal	3 Edge Pedestal	4 Doorway	5 Door Exit	6 Drywell
MELCOR-1	187	63	46	29	21	4
MELCOR-2	69	60	47	26	22	18
MELCOR-3	166	44	39	16	13	12
MELCOR-4	177	58	60	53	21	4
MELCOR-5	139	17	16	16	16	15
MELCOR-6	18	18	18	17	18	17
MAAP-HP-1	140	16	16	29	21	13
MAAP-HP-2	139	17	16	28	21	13
MAAP-HP-3	140	16	17	29	22	13
MAAP-HP-4	140	16	16	29	21	13
MAAP-LP-1	139	15	16	29	20	13
MAAP-LP-2	140	17	16	28	19	13
MAAP-LP-3	139	16	17	30	21	13
MAAP-LP-4	139	15	16	29	20	13

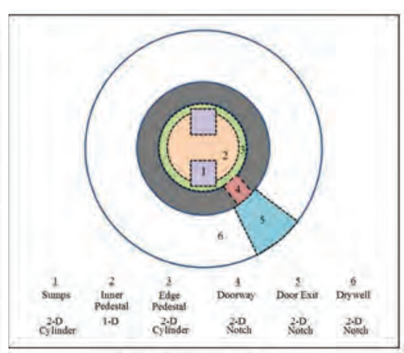


Fig. 30. Containment Discretization.

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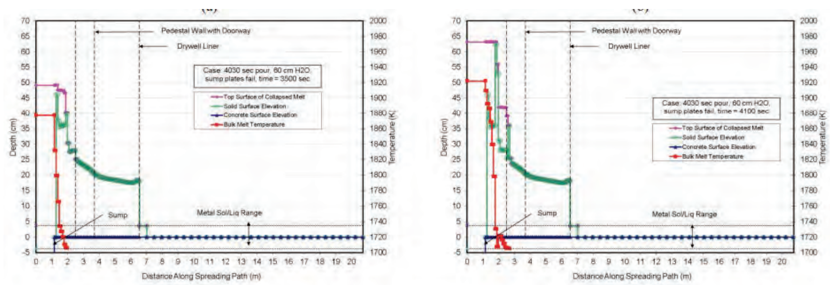
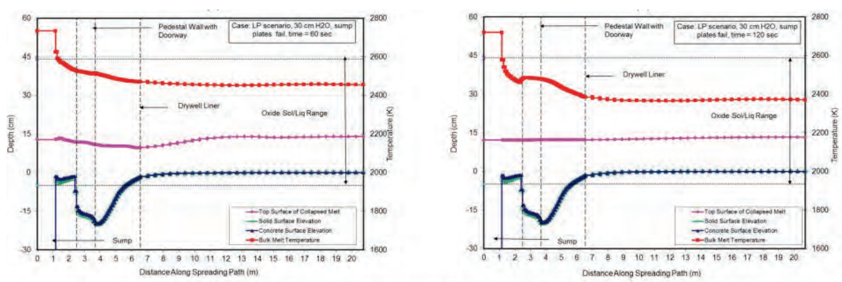
Enhanced Ex-Vessel Analysis for Fukushima Daiichi Unit 1: Melt Spreading and Core-Concrete Interaction Analyses with MELTSPREAD and CORQUENCH
K. Robb, M. Farmer, M. Francis, ORNL/TM-2012/455, Feb 2013.



(c) (d)

Fig. 15 (contd.). (a-d) Melt Temperature-Depth Profiles at 700, 2600, 3500, and 4100 sec for MELCOR Best Estimate Case.



(c) (d)

Fig. 18 (contd.). (a-d) Melt Temperature-Depth Profiles at 25, 30, 60, and 120 sec for MAAP-LP Best Estimate Case.

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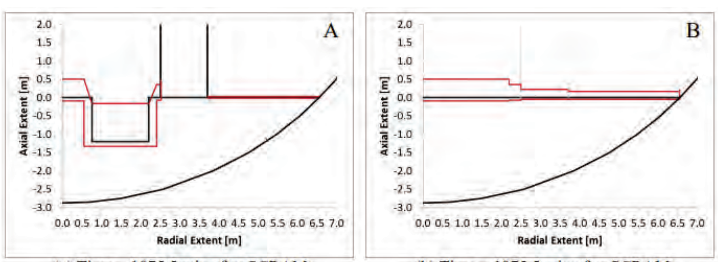
Enhanced Ex-Vessel Analysis for Fukushima Daiichi Unit 1: Melt Spreading and Core-Concrete Interaction Analyses with MELTSPREAD and CORQUENCH
K. Robb, M. Farmer, M. Francis, ORNL/TM-2012/455, Feb 2013.



(a) Time = 1075.5 min after SCRAM (b) Time = 1075.5 min after SCRAM

Fig. 35. MELCOR-1-1 Cavity Profile for Cross Section A and B after 60 min. of CORQUENCH simulation time and end of simulation.

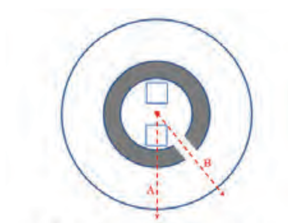
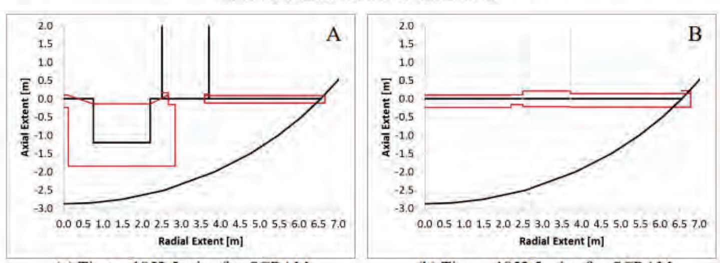


Fig. 32. Cross Sections of Containment Ablation.



(a) Time = 1852.5 min after SCRAM (b) Time = 1852.5 min after SCRAM

Fig. 39. MAAP-HP-3-1 Cavity Profile for Cross Section A and B after 60 min. of CORQUENCH simulation time and end of simulation. *These profiles do not take into account the concrete ablation that occurred during the spreading process as shown in Figure 25.

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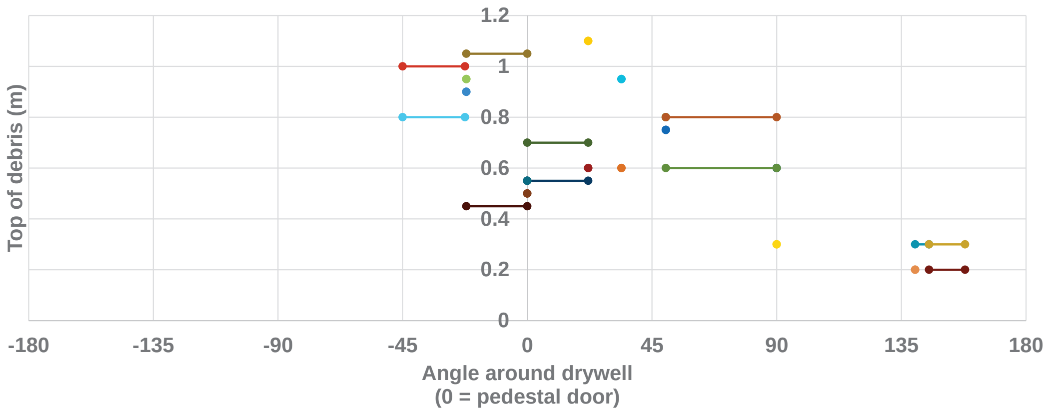
Top of Debris – estimate comparison

- Previous MELTSPREAD-CORQUENCH analysis all had much lower debris levels in drywell



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Enhanced Ex-Vessel Analysis for Fukushima Daiichi Unit 1: Melt Spreading and Core-Concrete Interaction Analyses with MELTSPREAD and CORQUENCH
K. Robb, M. Farmer, M. Francis, ORNL/TM-2012/455, Feb 2013.

- Several modeling scenarios evaluated
  - Example: crust anchoring allowed
    - CORQUENCH predicted crust anchoring in some locations (highlighted gray below)
    - All cases had crust anchoring in doorway

Table A-23. CORQUENCH Case 6 - Total Axial Concrete Ablation (cm)

Case	1 Sump	2 Inner Pedestal	3 Edge Pedestal	4 Doorway	5 Door Exit	6 Drywell
MELCOR-1-6	**51.2**	**126.4**	**21.1**	**17.6**	3.7	1.3
MELCOR-2-6	**15.4**	**116.4**	**22.7**	**15.3**	3.7	3.8
MELCOR-3-6	**48.4**	**68.9**	**13.3**	**9.8**	2.8	3.2
MELCOR-4-6	**48.1**	**112.6**	**42.5**	**44.0**	3.6	1.3
MELCOR-5-6	**62.8**	6.7	6.4	**8.3**	6.3	8.4
MELCOR-6-6	**9.1**	6.5	6.8	**9.1**	6.7	7.3
MAAP-HP-1-6	**86.7**	**21.3**	14.7	**18.8**	**19.4**	10.7
MAAP-HP-2-6	**86.2**	**22.7**	14.7	**18.4**	**18.8**	10.7
MAAP-HP-3-6	**86.8**	**22.1**	15.6	**19.3**	**20.6**	11.4
MAAP-HP-4-6	**86.7**	**21.3**	14.7	**18.8**	**19.5**	10.7
MAAP-LP-1-6	**85.3**	**20.6**	14.1	**18.5**	**17.7**	10.0
MAAP-LP-2-6	**85.3**	**22.1**	14.3	**18.0**	**16.2**	10.1
MAAP-LP-3-6	**85.2**	**21.2**	15.2	**19.2**	**18.9**	10.6
MAAP-LP-4-6	**85.1**	**20.2**	14.0	**18.5**	**17.7**	10.1

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Upper internals

- Melt pour scenarios didn’t include failure of upper internals
  - Melting of upper internals could add ~30% more mass, ~40% more volume to melt pour
    - Robb, Kevin R. "Heat up and failure of BWR upper internals during a severe accident." *Nuclear Engineering and Design* 314 (2017): 293-306.

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U.S. Efforts in Support of Examinations at Fukushima Daiichi - November 2021 Meeting Notes with Updated Information Requests

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

Location	1F1	1F2	1F3
X-100B PCV penetration[b]	Possible melted shielding material [36] No damage observed on outside [37]	NA	NA
X-51 PCV penetration[c]	NA	No damage observed: pressurized water could not penetrate blockage in standby liquid cooling system line [38, 39]	NA
X-53 HPCI steam supply penetration (1F2/1F3)[d]	High dose rate measured [40]	No damage observed [41]	No damage observed [42]

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U.S. Efforts in Support of Examinations at Fukushima Daiichi - November 2021 Meeting Notes with Updated Information Requests

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

Location	1F1	1F2	1F3
X-6 PCV penetration (CRD hatch)	NA	Melted material. [43, 44] Melted material expected to be from O ring and cable coating [Appendix C.1.3.2]	No damage observed from inside [45]
Equipment hatch	NA	NA	Water puddle [46, 47] unknown source
Personnel hatch and nearby penetrations	No damage observed [48]	NA	NA
HPCI pipe penetration[e]	No damage observed, but high dose rates measured: traces of flow and white sediment observed [40, 48,49]	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) [40,50]	Dose surveys do not indicate leakage from PCV through TIP guides. High dose levels in samples of materials from TIP indexer [51]	NA
WW vacuum breaker line	Leakage on expansion joint of one line (X-5E) [52]	NA	NA
DW/WW vent bellows	Water leakage attributed to vacuum line above [52]	No leakage observed [53]	
DW sand cushion drain pipe	Leakage [54]	No leakage observed [53]	NA
SC water level	Almost full [19]; increased leakage observed following February 2021 seismic event [55]	Middle [19]	Full [19]; increased leakage observed following February 2021 seismic event [55]
DW Water level	~2 m[19]	~0.2 m[19]	~6 m[19]
Torus room	Partially flooded [56, 57] Rusted handrails/equipment [36] NA	Partially flooded [58] Non-rusted handrails/ equipment [36,59] Some room penetrations tested, no leakage observed [61]	Partially flooded [58] Non-rusted handrails/ equipment [36,60] NA
MSIV room	Limited view obtained [62]	Water leakage cannot be observed. [63] Deterioration of HVAC ducting with sediment observed. Reactor well vent line confirmed open, but intentionally by operator prior to accident [Appendix C.1.3.3]	Leakage in Line D near bellows [64]
DW shield plugs	Reactor well shield plug displaced [65]	Possible leakage [66]	Leakage likely due to radiation measurements at head and presence of H_2 burn [19,67]

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

Location	1F1	1F2	1F3
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] Paint peeling observed. [Appendix C.1.3.2]	Paint peeling observed. [Appendix C.1.3.2]	NA
RCIC or other low SC piping	NA	Suspected leak location, not confirmed [36]	NA
RPV upper head	NA	NA	NA
RPV lower head	Ex-vessel debris images, dose surveys, and sample examinations indicate failure [19,68,69]	Ex-vessel debris and images confirm failure [67]	Ex-vessel debris images confirm failure [67]
SGTS vent path	High dose levels in vent path confirms rupture disk (RD) operation [70]	High dose levels in vent path, without RD disk operation, indicates backflow from 1F1 vent piping into 1F2 vent piping [70]	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 [70]

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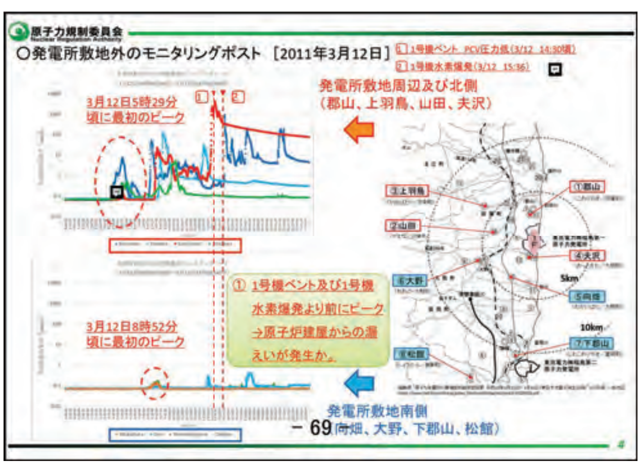
C.3.2. Topic Area 2- Radiation Surveys and Sampling

Task 2: Radiation Surveys and Sampling/Dose Calculation Insights and Comments on Future Examination Plans

Offsite Dose Rate – March 12, 2011



- 69

Source: Nuclear Regulation Authority
https://www.nra.go.jp/data/000403166.pdf

- Dose rate measured outside the power plant at the monitoring post on March 12, 2011.
- ~0400-0700: This left red-dashed oval shows peak dose. It is speculated that the peak may be from noble gases. Prior to the 1F1 venting and h2 explosion, there may have been leakage from reactor building.

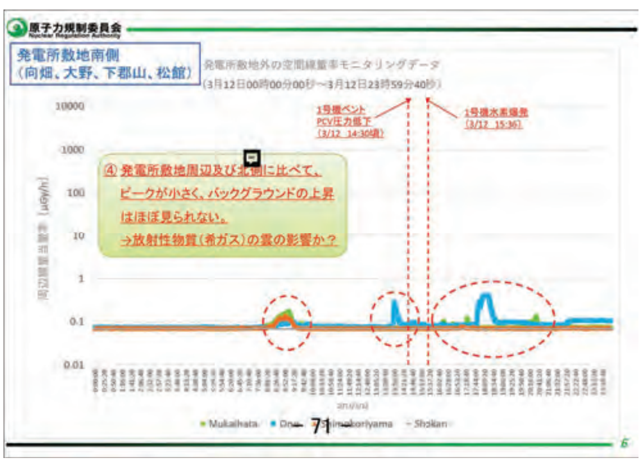
2

Koriyama, Yamada, Kamihatori, Ottozawa Site Detail

- After the spike, there's no increase in background levels. After venting (3/12 at 14:30 and the H2 explosion on 3/12 at 15:36, background levels increase). This has led NRA to speculate there's Cs and I release that is on the ground.

Source: Nuclear Regulation Authority
https://www.nra.go.jp/data/000403166.pdf

Mukaihata, Ono, Shimokoriyama, Shokan Site Detail

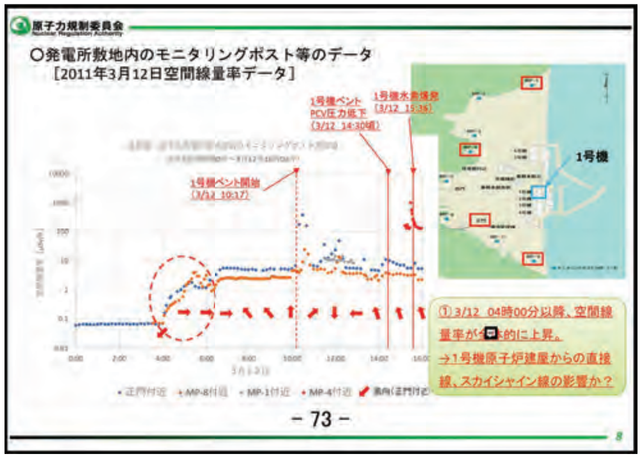


Source: Nuclear Regulation Authority
https://www.nra.go.jp/data/000403166.pdf

- Consider other locations in the vicinity of the power station site and the north side, smaller peaks, higher background is almost invisible. →Is it the difference due to the dispersion /transport of clouds of radioactive materials (noble gases)?

Site Dose Measurements – March 12, 2011 (1)

- After 04:00 on 3/12, space line overall increase in quantity. → Directly from Unit 1 Reactor Building line, is it the influence of the Skyshine line?



Source: Nuclear Regulation Authority
https://www.nra.go.jp/data/000403166.pdf

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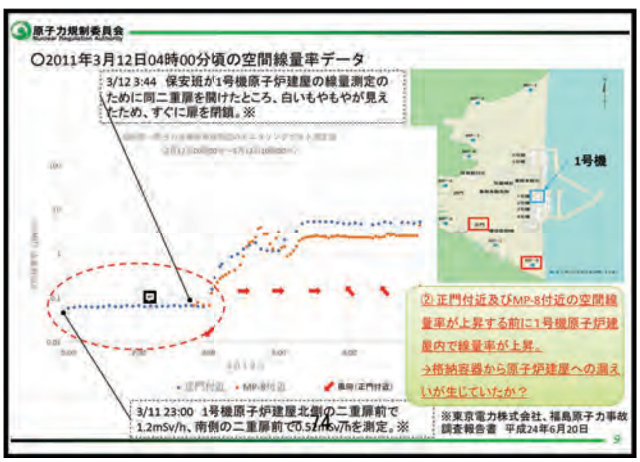
Site Dose Measurements – March 12, 2011 (2)



- During the time shown in the red dashed oval, the dose rate near the entrance of 1F1 after 4 am increases this may be when a TEPCO employee opened a 1F1 reactor building door. When they saw gas (a white haze) being omitted, he shut the door.

Source: Nuclear Regulation Authority
https://www.nra.go.jp/data/000403166.pdf

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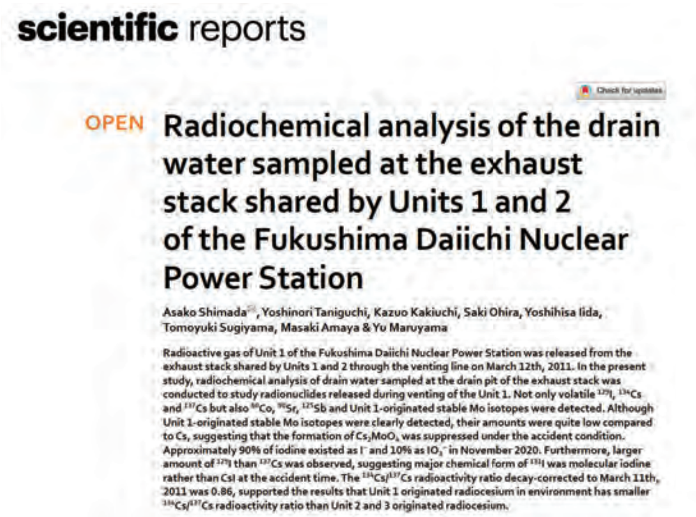
Radionuclide Speciation – Drain Water (1)



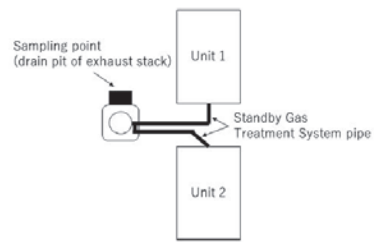
scientific reports

OPEN

Radiochemical analysis of the drain water sampled at the exhaust stack shared by Units 1 and 2 of the Fukushima Daiichi Nuclear Power Station

Asako Shimada, Yoshinori Taniguchi, Kazuo Kakiuchi, Saki Ohira, Yoshihisa Iida, Tomoyuki Sugiyama, Masaki Amaya & Yu Maruyama

Radioactive gas of Unit 1 of the Fukushima Daiichi Nuclear Power Station was released from the exhaust stack shared by Units 1 and 2 through the venting line on March 12th, 2011. In the present study, radiochemical analysis of drain water sampled at the drain pit of the exhaust stack was conducted to study radionuclides released during venting of the Unit 1. Not only volatile ^{129}I, ^{134}Cs and ^{137}Cs but also ^{60}Co, ^{90}Sr, ^{125}Sb and Unit 1-originated stable Mo isotopes were detected. Although Unit 1-originated stable Mo isotopes were clearly detected, their amounts were quite low compared to Cs, suggesting that the formation of Cs_2MoO_4 was suppressed under the accident condition. Approximately 90% of iodine existed as I^- and 10% as IO_3^- in November 2020. Furthermore, larger amount of ^{129}I than ^{137}Cs was observed, suggesting major chemical form of ^{131}I was molecular iodine rather than CsI at the accident time. The $^{134}Cs/^{137}Cs$ radioactivity ratio decay-corrected to March 11th, 2011 was 0.86, supported the results that Unit 1 originated radiocesium in environment has smaller $^{134}Cs/^{137}Cs$ radioactivity ratio than Unit 2 and 3 originated radiocesium.

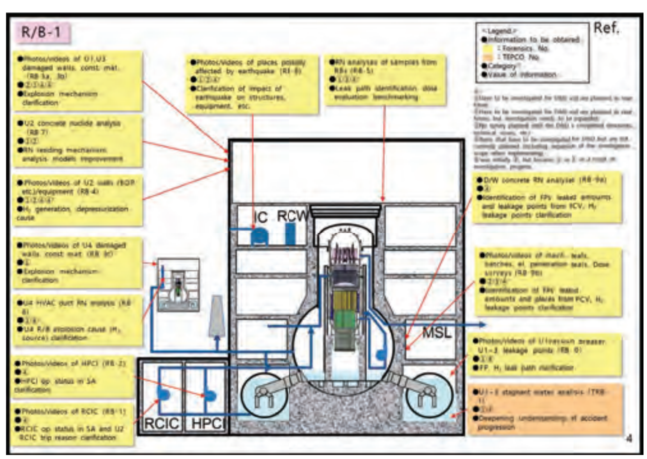


7

Images courtesy of Springer Nature. See: https://doi.org/10.1038/s41598-022-05924-2

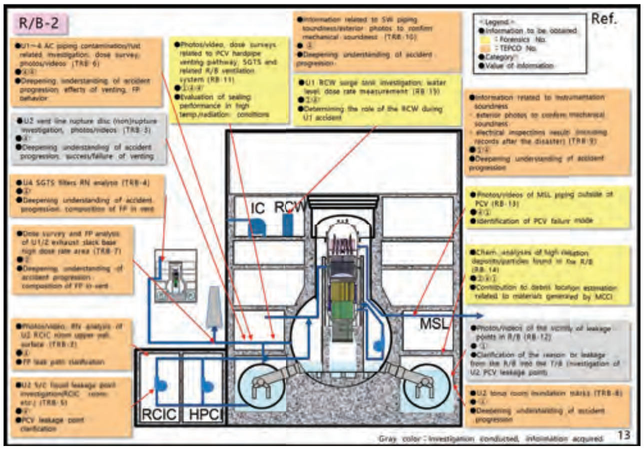
R/B -1

- Dose surveys and RN analyses
- Depressurization mechanisms
- H_2 and FP leak paths from containment to RB
- Stagnant water analysis



Source: M. Cibula, TEPCO Holdings, "Mid-and-Long-Term Plan for the Fukushima Daiichi Nuclear Power Station Accident Investigation", *presentation at DOE Forensics Meeting,* November 2021.

R/B - 2

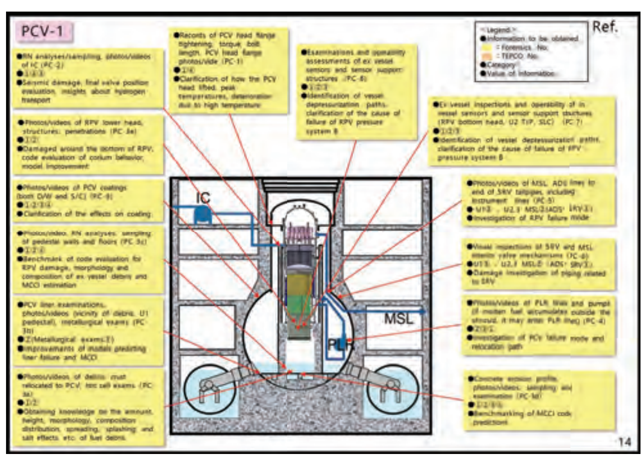


- Dose surveys and RN analyses
- Potential leak paths
- PCV failure modes

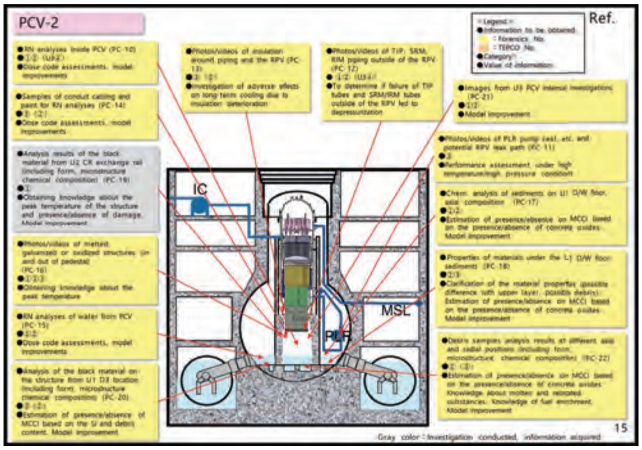
Source: M. Cibula, TEPCO Holdings, "Mid-and-Long-Term Plan for the Fukushima Daiichi Nuclear Power Station Accident Investigation", *presentation at DOE Forensics Meeting,* November 2021.

PCV - 1

- Dose surveys and RN analyses
- Primary depressurization mechanisms (MSL and SRV states)
- Final valve positions (e.g., leak paths)
- Debris characteristics



PCV -2

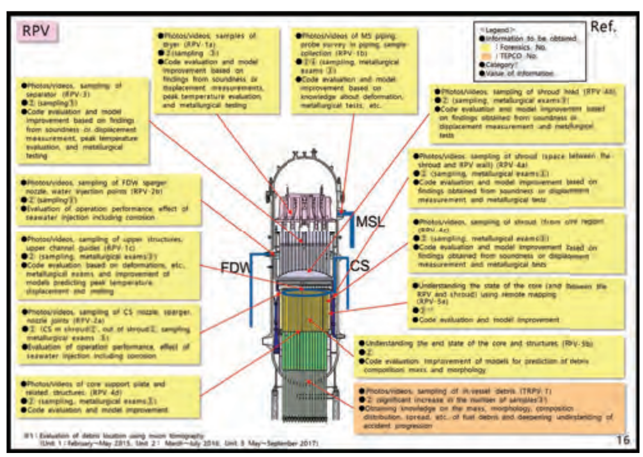


- Dose surveys and RN analyses
- Lower head state and failure mode indications
- Ex-vessel debris characteristics
- MCCI characteristics

Source: M. Cibula, TEPCO Holdings, "Mid-and-Long-Term Plan for the Fukushima Daiichi Nuclear Power Station Accident Investigation", *presentation at DOE Forensics Meeting,* November 2021.

RPV

- Dose surveys and RN analyses
- State of the upper vessel structures
- Evidence of depressurization mechanism (MSL piping, etc)
- In-vessel debris characteristics
- Debris end-states
- Core plate state and failure mode indications



Source: M. Cibula, TEPCO Holdings, "Mid-and-Long-Term Plan for the Fukushima Daiichi Nuclear Power Station Accident Investigation", *presentation at DOE Forensics Meeting,* November 2021.

Summary

- Integral observations from the Fukushima Daiichi accidents are generally consistent with existing MELCOR analyses
- Many valuable observations that reaffirm the MELCOR state-of-practice
- Future observations will support further refinement of existing models and severe accident progression assumptions hypothesized to date

C.3.3. Topic Area 3- Debris Endstate

C.3.3.1. Fukushima Fuel Debris Property Needs for Interim Storage

Fukushima Fuel Debris Property Needs for Interim Storage

Presented to the Reactor Safety Technology Expert Panel Forensics Meeting
Washington, DC, November 17, 2022

Dr . Martin G. Plys
Vice President and Chief Technology Officer
plys@fauske.com

Fukushima Fuel Debris Property Needs for Interim Storage

Introduction

- While forensic investigation of structures and debris is important to guiding improvement of severe accident models and mitigation techniques, an immediate practical application is to support the technical and licensing basis for debris retrieval, packaging, and interim storage.
- The information needs for packaging and interim storage depend both upon the methods of debris retrieval and the choice of any post-retrieval processes.
- This presentation provides my technical opinion and lessons learned, based upon about 28 years of experience in experiments and modeling for Hanford, Sellafield, and Fukushima.
- FAI is under contract to TEPCO to perform experiments and modeling to support interim storage choices.

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Fukushima Fuel Debris Property Needs for Interim Storage

Process Choices

- The major process choices are:
  1. Debris drying followed by interim storage in sealed containers (TMI fuel and debris, Hanford fuel and damaged fuel scrap).
  2. Active interim storage of wet debris in actively vented containers whose exhaust is HEPA filtered (retrieval phase of Hanford spent fuel sludge).
  3. Passive interim storage of wet debris in filter vented containers in a building whose exhaust is HEPA filtered (Hanford spent fuel sludge).
- In addition to the process, shielding must be provided; options are:
  1. Thick-walled, self-shielded containers (Sellafield FGMSP)
  2. Thin-walled containers within a shielded enclosure off-limits to personnel (Sellafield MSSS, Hanford)

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Fukushima Fuel Debris Property Needs for Interim Storage

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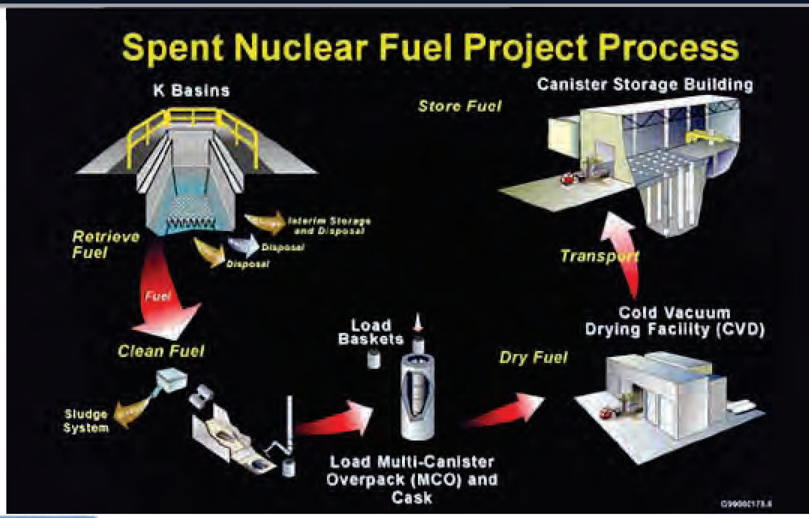
Interim Storage of Fukushima Fuel Debris – Hanford



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Interim Storage of Fukushima Fuel Debris – Hanford Fuel

- Hanford fuel was cleaned on a sorting table in the K West basin to remove sludge and to separate nearly intact fuel rods from scrap pieces.
- Fuel and scrap were vacuum dried in a custom-built facility, and the welded containers (known as MCOs) were placed in the Canister Storage Building.
- Fundamental technical issues with vacuum drying were:
  1. The need to clean fuel meticulously to remove fine particulate,
  2. The need for some other process to handle the sludge stream,
  3. The need for a quantitative proof of a high bound for water-bearing materials in the container to be dried, and
  4. The need for quantitative proof-of-dryness of a waste package.

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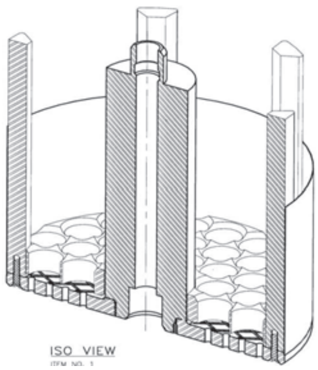
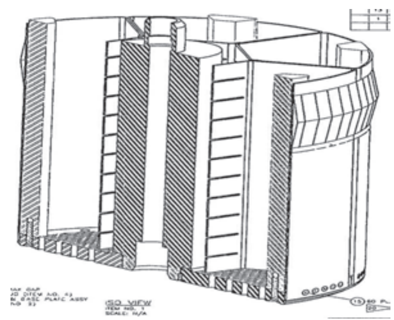
Interim Storage of Fukushima Fuel Debris – Hanford Fuel

Fuel basket, MCO, and scrap basket



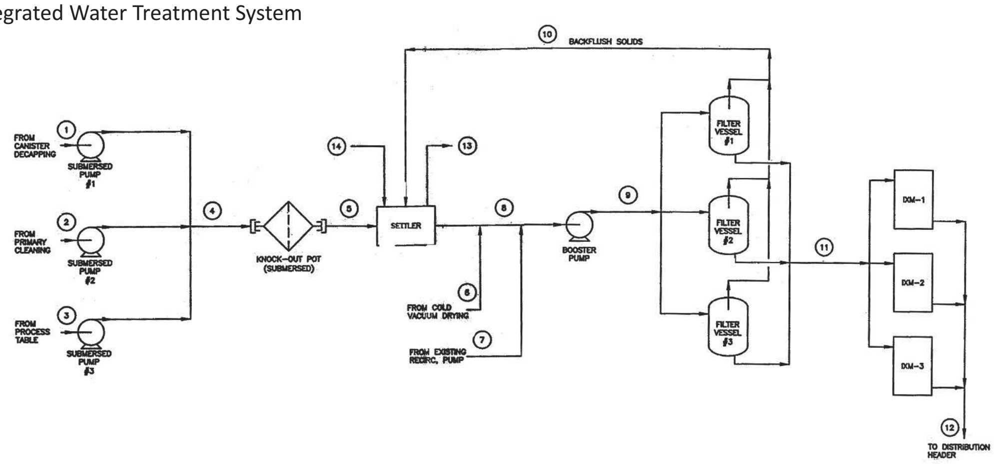


Interim Storage of Fukushima Fuel Debris – Hanford Sludge

- About 20 to 30 m^3 of fine sludge were collected in the Hanford K basins.
- Sludge was retrieved into containers known a Sludge Transport and Storage Containers (STSCs). A water layer was added over the sludge to prevent dryout, and there is provision for inspection of the water level and periodic water addition.
- The STSC lids have two vents of unequal height in order to promote natural circulation and remove hydrogen generated by chemical reactions of metal remaining in the sludge and by radiolysis.
- STSCs were placed in racks in process cells at a re-purposed reprocessing facility, T Plant. The process cells have concrete lids and are actively vented. This is what allows the STSC vents to be unfiltered.

Interim Storage of Fukushima Fuel Debris – Hanford Sludge

K West Integrated Water Treatment System



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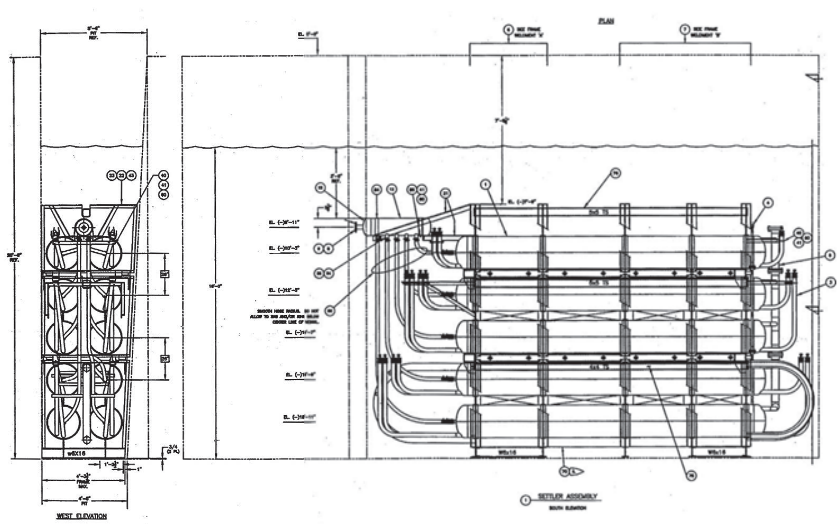
Interim Storage of Fukushima Fuel Debris – Hanford Sludge

Settler tanks for fine particulate (sludge)



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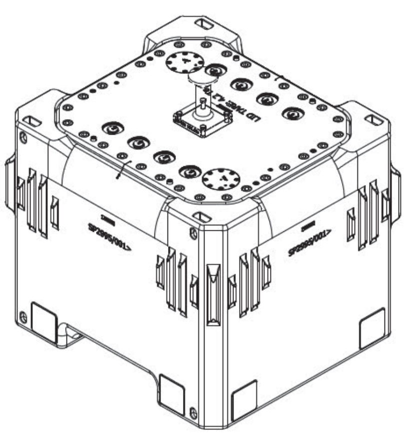
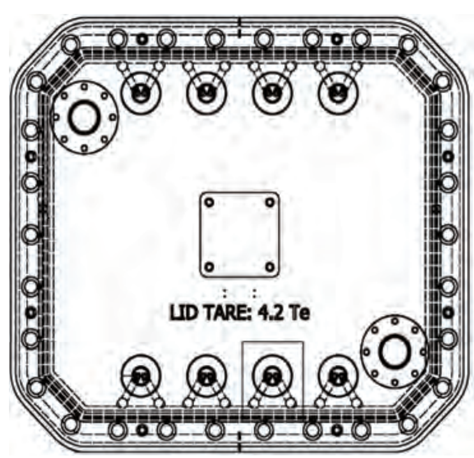
Interim Storage of Fukushima Fuel Debris – Sellafield FGMSP

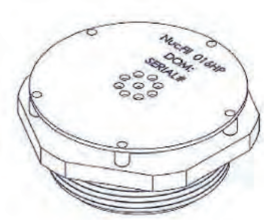
- Much of the fuel at Sellafield FGMSP is nearly completely decladded, exposing the uranium metal fuel to pond water, due to oxidation of the Magnox cladding to create magnesium hydroxide sludge. Fuel is held in containers known as skips.
- The current process for FGMSP cleanup is to place the skips into Self-Shielded Boxes (SSBs) that have filter vents for passive hydrogen and water vapor removal. This process is far less expensive and more safe than previously considered alternatives: Vacuum drying, encapsulation, and vitrification.
- The passive filter vent system was invented and experimentally verified by Fauske & Associates, and SSBs are currently being delivered to FGMSP.

Interim Storage of Fukushima Fuel Debris – Sellafield FGMSP

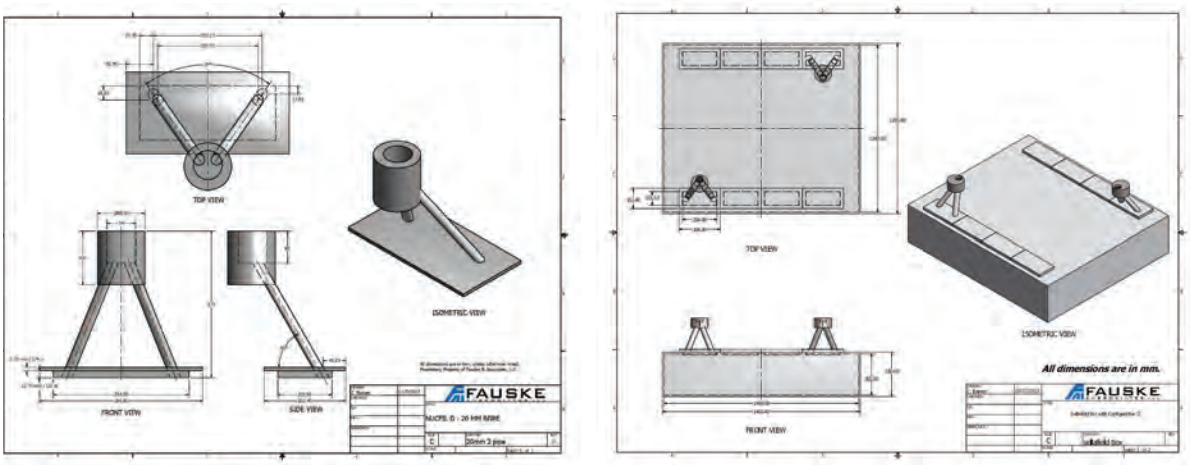




SSB, lid showing filter vent holes, and filter



Interim Storage of Fukushima Fuel Debris – Sellafield FGMSP



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FAI experiments for SSB lid filter vents

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Interim Storage of Fukushima Fuel Debris – Sellafield MSSS

- MSSS holds Magnox cladding removed from fuel, carryover of uranium metal, damaged fuel bars, and special containers with miscellaneous materials. The silo contents are dominated by sludge from degraded Magnox, but still also contain chemically reaction Magnox, and of course the uranium metal is chemically reactive.
- The original plan for remediation of MSSS involved a large facility for mixing the silo sludge and other materials with grout, which would then be poured into containers of somewhat smaller than an SSB.
- In the current process, which has just begun, MSSS waste is loaded into filter-vented skips placed in filter-vented 3m^3 boxes for passive interim storage – saving about 1 billion GBP.
- FAI provided experiments and modeling to support the current process.

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Interim Storage of Fukushima Fuel Debris – Process Conclusions

1. The clear trend for interim storage of highly damaged fuel debris is to explore passive means with minimal process steps.
2. Debris information needs for a drying process are far greater than information needs for processes without drying.

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Fukushima Fuel Debris Property Needs: Interim Storage of Wet Debris

- The drivers for information needs for interim storage of wet debris, whether actively or passively vented, are:

1. Initial water inventory, rate of water loss, and eventual time for dryout. The dryout time is important to provide a technical basis for limited container corrosion potential. Even debris with very low decay power will eventually dry out in just a few years in a properly designed system.
2. Debris temperature. This depends predictably upon decay power, container geometry, debris thermal conductivity, and the container external boundary condition.

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Fukushima Fuel Debris Property Needs: Interim Storage of Wet Debris

- Based upon our cumulative experience and our current experimental and modeling results obtained for TEPCO, these are the property needs for a process that avoids a drying step:

1. Particle size distribution (PSD), which is based upon the retrieval methods. Attempt to create coarse particulate but expect a fraction of fine particulate for any cutting/chiseling operation, which to first order will have a Rosin-Rammler PSD.
2. Residual saturation of retrieved debris. We wish to distinguish between the drainable and undrainable debris because this influences the drying time.

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Fukushima Fuel Debris Property Needs: Interim Storage of Wet Debris

3. Expected range of decay power within a container. This is directly related to the expected range of fuel fraction that would exist in a container (typically holding 10 to 100 kg debris) which is a small sample of the total debris inventory, and hence variable. High power = high temperature, low power = long drying time.
4. Debris bed thermal conductivity. This depends upon the intrinsic thermal conductivity of the debris particles, void fraction, and the local bed saturation (wet, partially wet, dry). FAI models have shown that we can tolerate reasonable uncertainty in debris thermal conductivity and maintain acceptable storage temperatures in a properly designed system.

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Fukushima Fuel Debris Property Needs: Interim Storage of Wet Debris

- We do not really need specific information on the following properties, because these have been shown by FAI to be inconsequential to the prediction of drying time and temperature:

1. Precise composition of debris. This is because we must tolerate a range of fuel fraction (decay power) and a reasonable range of debris bed thermal conductivity. This is good because it means a single process can be designed for storage of all retrieved debris.
2. Density and specific heat of debris. These only affect prediction of highly transient scenarios (accidents).

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Fukushima Fuel Debris Property Needs: Debris Drying and Sealed Interim Storage

- In addition to the information required for wet debris, we require the following for a drying process:

1. Quantification of pore water and chemically bound water inventories (chemically bound water can be converted to hydrogen during interim storage).
2. Method for proof-of-dryness and demonstration for a wide range of debris particle size distributions, including sludge (which is for practical purposes impossible to safely dry using the same process as for coarse fuel particulate).
3. Fission product release rate data under process conditions.
4. Technical basis for process safety under both normal and accident conditions.

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C.3.3.2. Core Debris Location Evaluations

WE START WITH YES.

TOPIC 3 - CORE DEBRIS LOCATION EVALUATIONS

MITCH FARMER
Nuclear Science & Engineering Division
Argonne National Laboratory

Fukushima Forensics Meeting, November 17-18, 2022
Hybrid Virtual (Webex) and In-Person at NEI, 1201 F Street, NW, Suite 1100, Washington, DC

PRESENTATION OUTLINE

- Review of most recent TEPCO/IRID/NRA[1] observations/questions of ex-vessel debris characteristics for 1F1.
- High level review of past MACE and OECD/MCCI test results focused on providing potential insights to some of these questions based on test results.
- CORQUENCH scoping calculations aimed at further addressing some of these questions, but from a modeling viewpoint.
- ~~Revisiting previous idea for potential *in-situ* core-debris water ingression measurement for 1F2~~. – see backup slides

1. Masaya YASUI, "NRA's Investigation (Phase 2) of Fukushima Daiichi Nuclear Accidents (2021-2022)," Reactor Safety Technology Expert Panel Forensics Meeting, Nuclear Energy Institute, Washington, DC, 17th November 2022.

QUESTIONS RAISED[1] FROM RECENT 1F1 INVESTIGATIONS

1. Why did the debris released from the RPV not spread out?
   – *Assumption on my part*: this is based on the elevated height of the crust material in doorway opening in relation to the volume of core debris discharged.
2. How was the pedestal wall concrete damaged but not the rebar?
3. How was the "suspended crust" material formed?

- Other questions of interest:

1. What is the source of the white powder that appears to cover some surfaces?

3

Argonne

INSIGHTS FROM MACE AND OECD/MCCI TESTS RELEVANT TO 1F1 OBSERVATIONS

- MACE tests examined debris coolability under ***early*** cavity flooding conditions.
  – All tests 1-D except the MACE Scoping Test (M0), which was 2-D.
- The OECD/MCCI (or 'CCI') tests were intended to provide additional data on 2-D MCCI behavior as well as debris coolability.
  – All tests flooded late except for CCI-6 that featured early flooding.
- MACE tests (with LCS and SIL concrete types) all exhibited behavior in which the upper crust formed by water cooling would 'anchor' to test section sidewalls.
- The 'anchored' crust would eventually separate from the melt due to: i) reduced gas sparging as the test progressed, causing the voided melt height to decrease, and ii) concrete densification upon melting (i.e., 'slumping').
  – In all tests, this led to suspended 'bridge crusts' anchored to test section sidewalls and separated from the underlying melt by an intervening gap (illustrations to follow).

4

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INSIGHTS FROM MACE AND OECD/MCCI TESTS RELEVANT TO 1F1 OBSERVATIONS

- The extent that core debris will slump due to concrete densification upon melting is given by the equation:

$$\frac{V_f}{V_0} = (1 - \chi_{gas})\left(\frac{\rho_{con}}{\rho_{slag}}\right)$$

- Here, χ_{gas} is the mass fraction of decomposition gases in concrete (H_2O and CO_2), ρ_{con} is the original concrete density, and ρ_{slag} is the density of the slag produced by melting (i.e., gases leave, and $Ca(OH)_2$, $CaCO_3$, and $MgCa(CO_2)_3$ are decomposed into the simple oxides CaO and MgO).
- For CORCON default 'Basalt' concrete, χ_{gas} = 0.0708, and based on CORQUENCH thermo-physical property subroutines, ρ_{con}= 2431 kg/m^3 and ρ_{slag}= 2542 kg/m^3.
- With this information, $\frac{V_f}{V_0} = 0.889$ for Basalt concrete, implying that for every 10 cm of erosion, 1.1 cm of surface elevation reduction will occur during ablation in 1-D.
- More slump will occur in 2-D erosion cases, but volume reduction equation is more complicated as it depends on extent of lateral vs. axial ablation.

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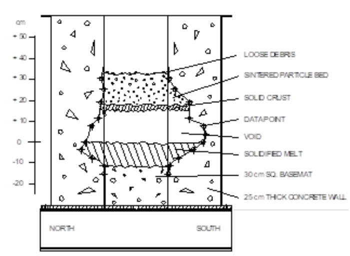
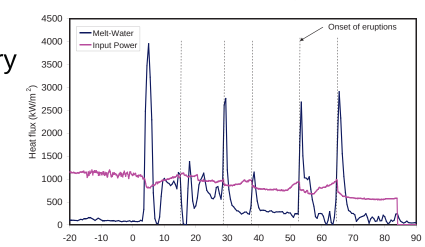
MACE SCOPING TEST[2]

LCS concrete, 30 cm x 30 cm square test section, 130 kg corium mass, high power density test

- Top crust anchored to sidewalls, and was mechanically stable for rest of test.
- Due to high power density (700-1400 W/kg fuel), relatively thin conduction-limited crust formed.
- Periods of high melt void fraction occurred in which the melt re-contacted the crust, leading to melt eruptions and particle bed formation.
- Anchored crust viewed as 'non-prototypic' and the test section design was changed to refractory sidewalls and larger scales in subsequent tests in an effort to achieve a floating crust boundary condition.





2. M. T. Farmer, D. J. Kilsdonk, and R. W. Aeschlimann, "Corium Coolability under Ex-Vessel Accident Conditions for LWRs," *Nuclear Eng. Technology*, Vol. *41*, pp. 575-602, June 2009.

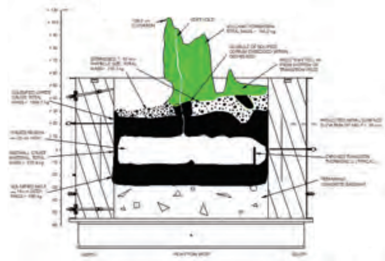
6

Argonne

MACE TEST M3B[2]

LCS concrete, 120 cm x 120 cm square test section, 1-D (refractory walls), 2000 kg corium mass, normal power density

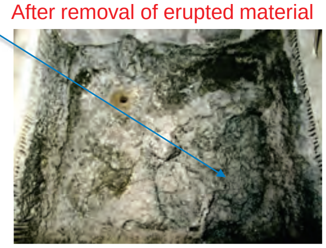
- Crust stress analyses[3,4] indicated that a test section size of 2 meters or more would be needed to achieve a floating crust boundary condition.
- Practical considerations limited the maximum test section size to 1.2 m.
- Test showed similar phenomenological behavior as M0 (anchored crust, periodic eruptions), but evidence of crust structural failure was also noted at this increased test scale.



As found

After removal of erupted material

Section of crust that structurally failed



3. Z. Feng, R .L. Engelstad, E. Lovell, M .L. Corradini, "Stress Analysis and Scaling Studies of Corium Crusts," Proceedings of the Second OECD (NEA) CSNI Specialist Meeting on Molten Core Debris-Concrete Interactions, KfK 5108 NEA/CSNI/R(92)10, April 1992.
4. J. H. Ptacek, Z. Feng, R.L. Engelstad, E.G. Lovell, M.L. Corradini, and B.R. Sehgal, "Modelling of the MCCI Phenomena with the Presence of a Water Layer," Proceedings of the Second OECD (NEA) CSNI Specialist Meeting on Molten Core Debris-Concrete Interactions, KfK 5108 NEA/CSNI/R(92)10, April 1992.

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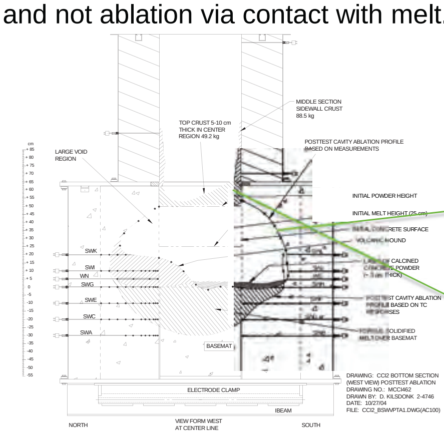
OECD/MCCI TEST CCI-2[2]

LCS concrete, 2-D test section with 50 x 50 cm initial cavity size; 6 hours of dry cavity ablation, followed by cavity flooding

- No apparent crust anchoring, but large accumulation of core debris in upper portion of test section due to deposition from highly swelled melt pool height.
- Extensive ablation above the solidified debris over the basemat; likely due to radiation heat transfer from upper surface of melt during the test.
- Relatively thick (3 cm layer) of calcined concrete found on top of debris, supporting the idea that this material was produced by radiation heat transfer and not ablation via contact with melt.



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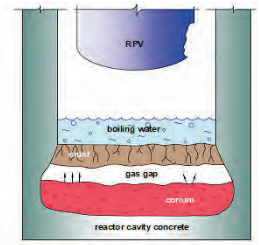
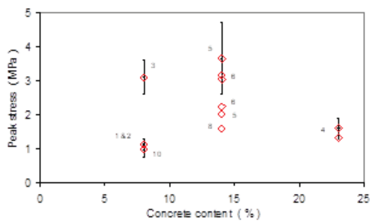
CRUST ANCHORING IN RELATION TO PLANT CONDITIONS

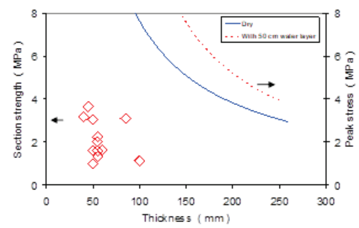


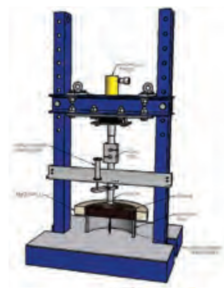
- Occurrence of anchoring in MACE tests raised concerns that if this happened at plant scale, it may prevent debris cooling by forming an insulating gap between the crust and remaining melt.
- On this basis, load tests were conducted on sectioned corium ingots from SSWICS tests to measure tensile strength of core debris containing inherent crust cracking and porosity.
- Results indicated that crusts are weak, and that sustained anchoring at plant scale is unlikely given the 6 m cavity span in many plants.
- Rather, periodic crust anchoring and 'breach' would likely occur, leading to renewed pathway(s) for water to re-contact and ingress into the debris.

. Max centerline stress before fracture for ingot sections under point load at room temp



Measured section strength and calculated peak stress in a 6 m OD self supported crust





MAIN INSIGHTS FROM MACE AND CCI TESTS

1. In MACE tests featuring early cavity flooding, crust anchoring leaving behind a suspended bridge crust occurred in all tests.
   - Exacerbated by the fact that tests were flooded early when pool swell due to gas sparging was the highest.
   - Evidence of crust 'breach' observed in one test (M3B, largest @ 1.2 x 1.2 m).
     - *These observations seem to be consistent with 1F1 findings of suspended crust material and occurrence of crust shelves attached to structure.*
     - *Based on crust strength measurements, likely limited to doorway opening and drywell areas due to the tighter dimensions in these locations*.
2. For dry cavity tests with extensive ablation, high melt void fractions (>50%) periodically observed leading to deposition of crust material at high elevations in the test section. (Note: also occurred in ACE/MCCI Tests L1 and L5).
   - *Might partially explain elevated debris heights in 1F1; i.e., the debris did actually spread, but this is masked by elevated crusts formed by foaming and solidification behavior, as in the MACE/CCI tests.*
   - *Researchers attributed this high void fraction behavior to melt 'foaming,' and published models for this process[5].*

5. B. Tourniaire , E. Dufour, and B. Spindler, "Foam Formation in Oxidic Pool with Application to MCCI Real Material Experiments," Nuclear Engineering and Design, Vol. 239, pp. 1971-1978 (2009).

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MAIN INSIGHTS FROM MACE AND CCI TESTS

3. The dry cavity tests also showed evidence of ablation above the collapsed melt height, leaving calcined powder buildup on top of the crust material. The vertical extent of this ablation was limited by bridge crust material attached to structure above the melt. The bridge crust apparently acted as an insulator preventing ablation above the crust.
    - *Consistent with observations of pedestal wall ablation below crust material attached to the pedestal walls in 1F1.*
4. The above mentioned ablation behavior also led to buildup of calcined concrete layer over the solidified debris.
    - *Could partially explain the occurrence of white powder layers covering material in various locations in 1F1.*
    - *The fact that the powder was not incorporated into the melt, but rather accumulated on top of the upper crust, indicates that this material was probably formed by radiation heat transfer from the melt as opposed to direct contact with the core debris.*

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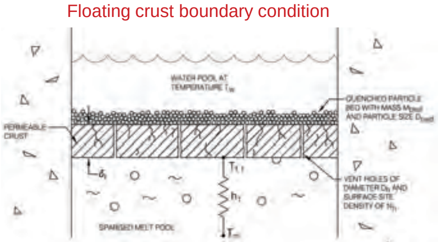
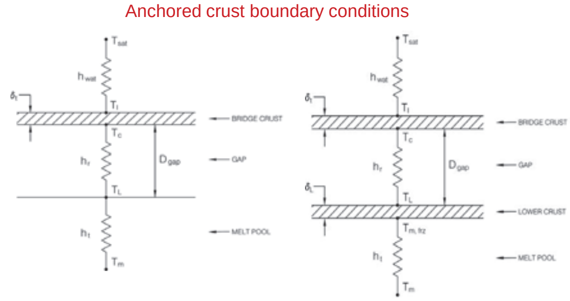
CORQUENCH SCOPING CALCULATIONS OF CRUST ANCHORING BEHAVIOR IN 1F1

- A simple crust anchoring model was integrated into CORQUENCH in the mid 90's and validated against MACE test results.
  - Motivation: need for an analysis tool to better understand crust anchoring behavior as observed in MACE tests.
- Models still present and executable in current version of CORQUENCH, but are rarely used except for code validation efforts.
- Full documentation in reference 6 if you are interested.

Floating crust boundary condition



Anchored crust boundary conditions



6. M. T. Farmer, "The CORQUENCH Code for Modeling of Ex-Vessel Corium Coolability under Top Flooding Conditions: Code Manual-Version 4.1-beta," Argonne National Laboratory, ANL-18/22, August 2018.

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SCOPING CALCULATIONS CONTD.

Basic modeling assumptions:

1. The crust will continue to 'float' over melt as long as it's mechanical strength is < than that which can support the combined loads of the crust and particle bed weights, as well as the weight of overlying water layer (as applicable).
2. If the crust grows to a thickness where it can support those loads, then it is assumed to anchor at it's current position.
3. Thereafter, whether or not the melt remains in contact with the crust depends on the fixed crust elevation vs. the time-dependent voided melt height.
   - If the melt swells to maintain or re-contact the anchored crust, then normal cooling mechanisms can proceed.
   - If the voided height is < than the bottom of the fixed crust, then a radiation heat transfer resistance is introduced between the melt and crust.
4. As time progresses, the crust strength is continuously checked against the applied loads, and if the loads exceed the strength, the crust is assumed to 'fail' and is placed back on top the melt as a 'floating' crust.

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SCOPING CALCULATIONS CONTD.

- Explicitly, the anchoring criterion is as follows:

$$\underbrace{g\left(m_{bed} + \rho_{t,c} A_b \delta_{t,\min} + m_{wat}\right)}_{applied\ load\ on\ crust} \leq \underbrace{C_{geom}\, \hat{\sigma}_{t,f}\, \delta^2_{t,\min}}_{crust\ mechanical\ strength}$$

- Once anchored, this equation is also used to determine if the crust subsequently fails; the crust is then placed back atop the melt pool.
  - Mechanisms that can lead to failure after anchoring are: i) increased crust size (by lateral ablation), and/or change of loading on top of crust (increase area via radial ablation; reduced crust thickness via re-melting while suspended; water addition…)
- Whether or not a gap forms is determined simply by tracking voided melt height relative to the anchored crust position bottom surface position; i.e.,

$$D_{gap} = \left\langle EL_{anchor} - EL_{m,v}(t) + \delta_{t,anchor} - \delta_t(t),\, 0\right\rangle$$

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SIMULATED CASES

- Two cases were executed: one to mockup up behavior in the pedestal doorway region, and a second to mock up behavior in the larger drywell annulus region. Geometry for two cases as follows:

1. *Pedestal doorway opening*: 0.851 m wide door x 1.28 m pedestal wall thickness; ablation into pedestal walls adjacent to the doorway; adiabatic on other two sides.
2. *Drywell Annulus*: 2.55 m radial slice between exterior of pedestal wall and drywell liner; width of slice assumed to be 2 m. Ablation into pedestal wall and PVC liner modeled; other two sides treated as adiabatic.

- 30 cm uniform melt depth assumed after vessel failure in pedestal/drywell regions.
  - For 140 MT pour mass, equivalent to filling the pedestal sumps with corium and spreading material out the door to cover 112 degrees of the drywell area.
  - MELCOR melt composition and initial temperature the same as in Ref. 7.
- Concrete type assumed to be CORCON Basalt.
- Based on crust strength measurements made as part of OECD/MCCI program, a tensile strength of 3 MPa is assumed (see pg. 12).
- Brockmann correlation used to predict melt void fraction; melt foaming ***is not*** modeled.
- Calculation ran out to 14 days, which includes 11.25 days of dry cavity ablation, followed by cavity flooding to a uniform depth of 2 m (current condition).

7. K. R. Robb, M. W. Francis, and M. T. Farmer, "Ex-vessel Core Melt Modeling Comparison Between MELTSPREAD-CORQUENCH and MELCOR 2.1," Oak Ridge National Laboratory, ORNL/TM-2014/1, March 2014

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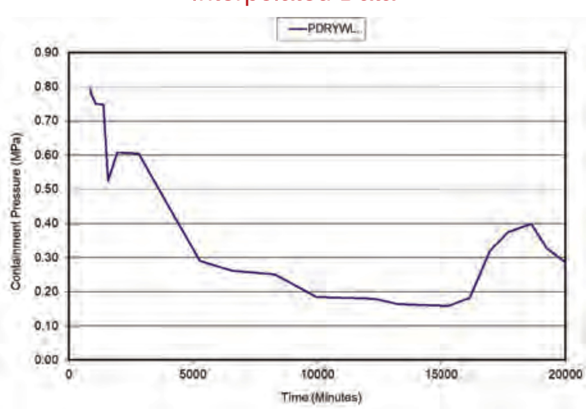
ASSUMED CONTAINMENT PRESSURE

Important as this impacts superficial gas velocity from concrete decomposition and melt void fraction

- Estimated by using TEPCO data where it exists and interpolating using MELCOR results[8] in regions where data does not exist….

Interpolated Data



8. N. Andrews, R. Gauntt, et al., "BSAF Phase 2 Sandia National Lab Activities," presentation SAND2017-0178PE.

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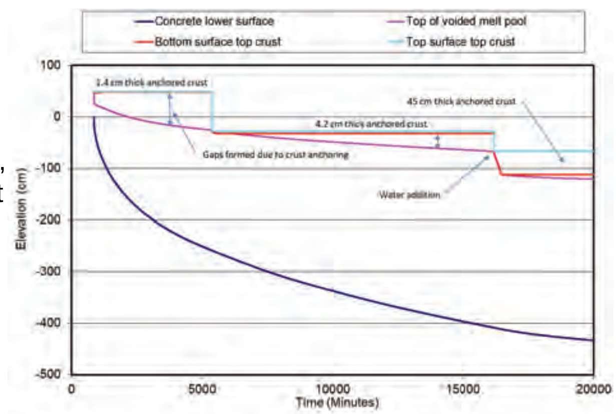
DOORWAY RESULTS

Surface elevation of crust and voided melt height

- Results indicate possibility of three crust anchoring and two crust failure events in the region spanning the pedestal doorway over the calculated time interval.

- Crusts are anchored over a major fraction of the time.
- Water addition caused 2nd failure.
- Although not explicitly modeled, failure events would leave crust ledges on concrete walls (1st 1.4 cm thick, the 2nd 4.2 cm thick).
- Large time intervals during dry phase in which gaps formed would allow lateral ablation by radiation heat transfer to exposed concrete.



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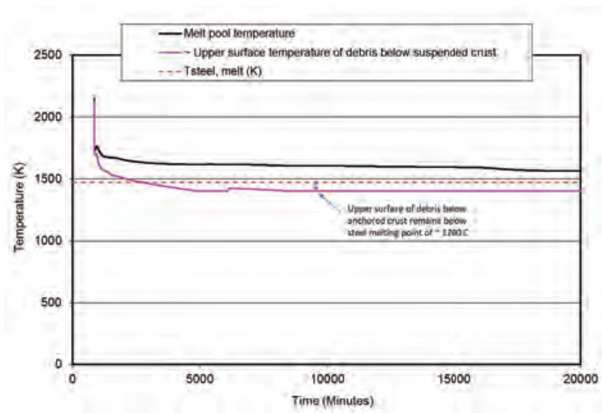
DOORWAY RESULTS

Melt and crust temperature results

- During extended periods of crust anchoring before flooding, the upper surface temperature of the debris below the bridge crust remains below steel melting temperature (assumed to be 1200 C = 1473 K here).

- Thus, the rebar exposed by radiation-driven concrete ablation in the doorway sidewalls would not have been ablated according to this modeling result.



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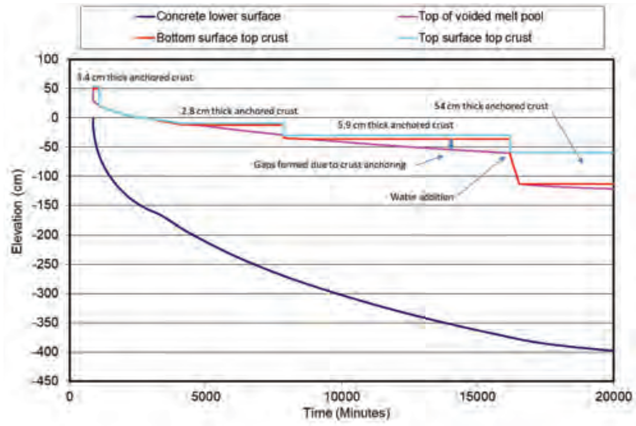
DRYWELL ANNULUS RESULTS

Surface elevation of crust and voided melt height

- Overall, trends and observations are similar to doorway results. However, a few differences:
- Four anchoring events as opposed to three for the doorway case.
- Early on, there was an extended period (~ 2 days) in which the upper surface was initially crust free, and then was covered with a crust that floated.



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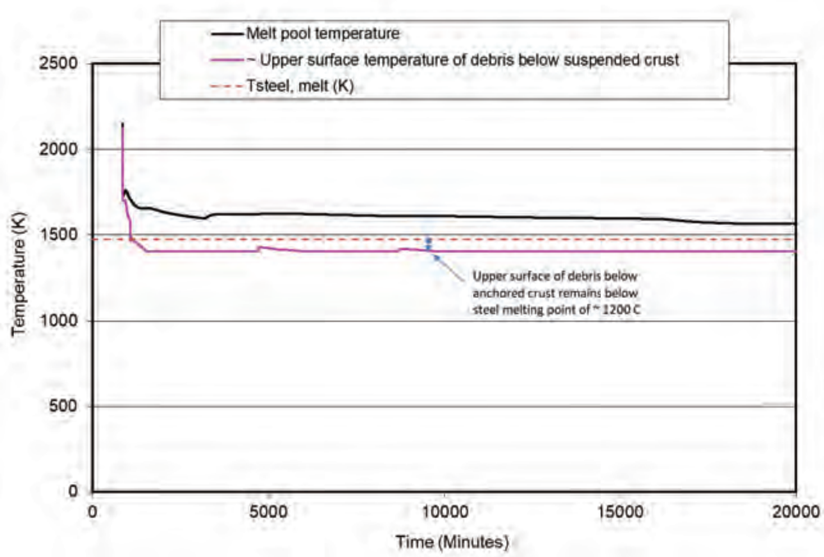
DRYWELL ANNULUS RESULTS

Melt and crust temperature results

- General trends and conclusions similar to the doorway case.



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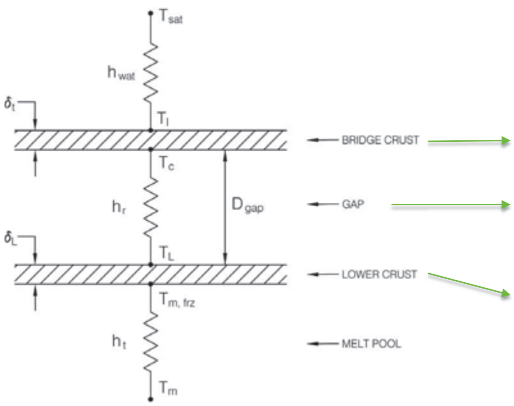
PREDICTED END STATE CONDITIONS AT 11 DAYS, JUST PRIOR TO CAVITY FLOODING



Parameter	Drywell Annulus Case	Doorway Case
Suspended Bridge Crust Thickness (cm)	5.9	4.3
Gap Thickness Between Crusts (cm)	23.0	33.6
Bottom Crust Thickness over Melt (cm)	2.9	3.8
Radiation View Factor to Sidewalls	0.10	0.20

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MAIN INSIGHTS FROM CQ SIMULATIONS

1. Using crust anchoring models as currently deployed in CQ and the estimated crust tensile strength of ~ 3 MPa from previous measurements, the results indicates that crust anchoring would likely occur over the first 11 days in which the cavity remained dry.
   – *Multiple crust failure events are predicted, which would leave crust ledges in the range of 1-5 cm attached to sidewall materials*.
2. Crust ledges consistent with some of the observations reported on the basis of video data from 1F1.
3. Predictions of the occurrence of anchored crusts as well as the temperature evolutions in the gap between the anchored crust and melt were consistent with the idea that concrete around the rebar would have been ablated, but not the rebar itself.
   – *Also consistent with some of the reported observations.*

5. B. Tourniaire , E. Dufour, and B. Spindler, “Foam Formation in Oxidic Pool with Application to MCCI Real Material Experiments,” Nuclear Engineering and Design, Vol. 239, pp. 1971-1978 (2009).

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CLOSING COMMENTS-CQ ANALYSES

- Note that the CQ crust anchoring calculations presented here are scoping in nature; this is first time these models have been used for a long-duration real plant accident scenario including extended dry and wet phases.
  - Note that there were a few instances (in time) for both cases where the code was not able to meet specified convergence criteria, and I did not have time to chase down the reasons why or debug.
- This work also revealed some modeling shortcomings.
  - Crust strength calculation based on a simple plate (i.e., MCCI surface area) model. For places like the annulus, a beam strength model would be more appropriate because the behavior is essentially 1-D in the radial direction.
  - The model does not leave crust ledges when the crust fails, as observed in 1F1.
  - The code also pessimistically assumes that once the crust anchors, water is not able to flood below the crust to continue cooling. In essence, 'crust breach' is not modeled.

CLOSING COMMENT-GENERAL

- Much work was done in the MACE and OECD/MCCI programs addressing the issue of crust anchoring and whether or not this type of behavior would be applicable to plant sequences.
  - Concerns of whether anchored crust(s) would inhibit debris coolability.
- The results (both analytical and experimental) indicated that for tight cavity regions (a few meters) this may occur, but it was argued that even if the crusts did anchor, they would not be completely stable. In particular, 'crust breach' would occur, allowing water to flood below the anchored crust and thereby maintain debris cooling.
- The results from 1F1 seem to support this vision of crust anchoring and breach behavior, thereby allowing the debris to cool. This is a beneficial confirmatory observation from the viewpoint of reactor safety!!

ACKNOWLEDGEMENTS

- Thanks to all organizations within Japan for ongoing interactions on this project.
  - Findings from Daiichi have provided many insights into severe accident progression and management, and considerably reduced knowledge gaps in this area.
- Thanks to DOE and Program Manager Mr. Damian Peko for continued support; US national lab involvement would not be possible without it.
- Thanks to Dr. Rempe for leading these efforts and providing the information in a format that we can access, analyze, and assess.

BACKUP SLIDES

REVISITING THE POTENTIAL FOR *IN-SITU* CORE DEBRIS WATER INGRESSION MEASUREMENT FOR 1F2

- 1F2 examinations have revealed extraordinary information on ex-vessel core debris distribution within the pedestal, including data on water injection characteristics.
- Specifically, video indicates that injected water penetrates the core debris (50-70 cm in depth) and passes through that material during passage to drywell annulus where water level is constant at ~30 cm. This is clear evidence of water ingression.
- ***If conditions allow***, it would be advantageous to obtain video footage while injection flowrate is increased in a step-wise manner until water begins to accumulate on the surface and spill over directly into the annulus through the pedestal doorway.
- This information could be used to estimate debris permeability and dryout limit for an actual prototypic core debris accumulation, which is valuable for safety evaluations.
- A white paper was prepared describing this procedure.

1F2 in-pedestal debris distribution

Image courtesy of TEPCO: http://www.tepco.co.jp/decommission/information/committee/roadmap_progress/pdf/2020/d200327_05-j.pdf

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OVERVIEW OF METHOD

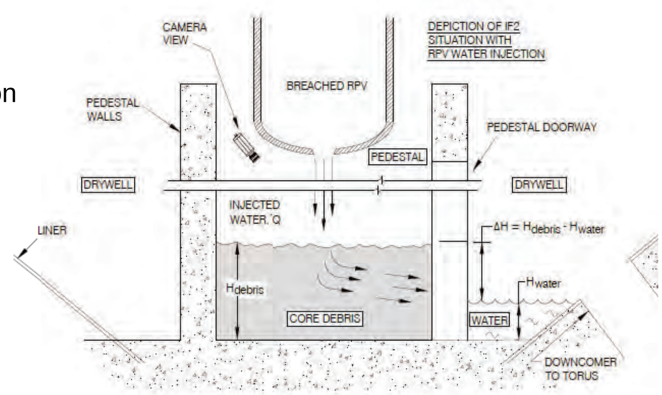
- Water injection flowrate would be gradually increased until it pools above the debris and begins to flow out the doorway, this would be flooding limit, $\dot{Q}_{limit}$.
  - Camera footage would be needed to determine when this point is reached.

- With flooding limit known, debris permeability can be estimated using the equation (based on Darcy's law):

$$\kappa = \frac{\mu L \dot{Q}_{limit}}{\rho g \Delta H A}$$

- The debris dryout limit can then be estimated from:

$$q" = \frac{\kappa \, \rho_g h_{fg} \left(\rho_f - \rho_g\right) g}{2\mu_g}$$



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C.3.4. Topic Area 4 - Combustible Gas Effects

C.3.4.1. Thoughts on Cable Degradation Testing and Proposed Future Examinations

Fukushima Forensics: Combustible Gas Effects - **Thoughts on Cable Degradation Testing and Proposed Future Examinations**

Wison Luangdilok
H2Technology LLC
The Joint Graduate School of Energy and Environment (JGSEE), KMUTT

DOE Reactor Safety Technology Expert Panel Forensics Meeting
Nuclear Energy Institute, 1201 F Street NW, Suite 1100, Washington, DC
November 17-18, 2022

H2TECHNOLOGY LLC

Outline

- Experimental results on combustible gas generation from hydrocarbon polymeric materials under high temperature steam or hydrogen atmosphere
- Comments on NRAJ proposed combustion experiments
- Suggestion on the range of hydrogen concentration for combustion experiments
- Suggestion on the geometric-specific fluid dynamics aspect of the 1F3 combustion problem

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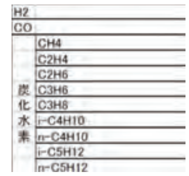
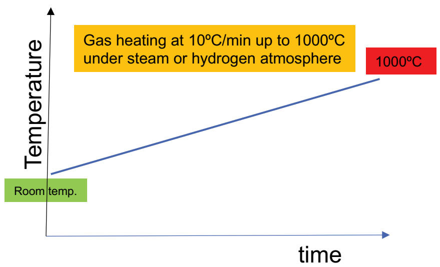
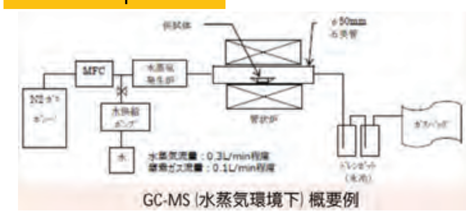
TEPCO experiments

On thermal decomposition of hydrocarbon polymeric materials under heated atmosphere filled mostly with steam or hydrogen and ~25% nitrogen as remaining gas.





TEPCO experiment



Source: Nuclear Regulation Authority
https://www.nra.go.jp/data/000403170.pdf (9/6/2022)

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TEPCO's Experiments on Combustible Gas Generations from Hydrocarbon Polymeric Materials

4 Source: Nuclear Regulation Authority
https://www.nra.go.jp/data/000403170.pdf (9/6/2022) H2TECHNOLOGY LLC

Summary of TEPCO's Test Results as of April 26, 2022

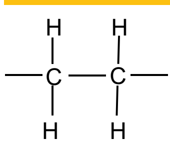
Results of TEPCO experiments on combustible gas generation from cables, paints and insulations subjected to heated steam up to 1000 [1]

	Gas release (m^3/ton)			Gas release (g/kg)		
	H2	CO	CH4 equivalent	H2	CO	CH4 equivalent
Urathane Thermal insulators for piping	264	116	56.6	23.57	144.96	40.42
CV Cable Polyethylene + heat-resistant vinyl	101	19.5	60.2	9.02	24.37	42.99
PN Cable Polyethylene + Polypropylene	398	162	57.4	35.53	202.45	40.99
Co-axial Cable ETFE/polyethylene	33.7	11.7	27.2	3.01	14.62	19.42
Polyimide Thermal insulators for piping	632	394	26.9	56.41	492.37	19.21
Epoxy Paint Top coating	131	20.5	28.7	11.69	25.62	20.49
Zinc-rich paint-Bottom coating KGB cable- silicone rubber Lubrication oil – PLR motors	Testing in progress					

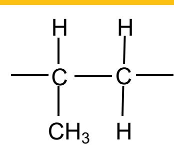
Source: Nuclear Regulation Authority [1]https://www.nra.go.jp/data/000388502.pdf (4/26/2022)

Reaction of Hydrocarbon Polymers with High Temperature Steam

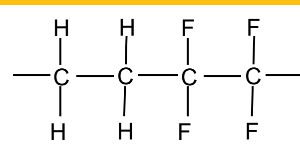
Steam oxidation of hydrocarbon polymers: $C_nH_m + n\ H_2O \rightarrow n\ CO + (n+m/2)\ H_2$



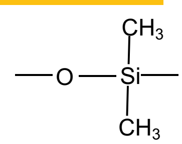
Polyethylene



Polypropylene



Ethylene tetrafluoroethylene
ETFE



Silicone

Hypalon40 : $C_{85}H_{157}Cl_{13}SO_2$. is widely used as cable insulation in US nuclear plants (NUREG-5950)

Estimation of ex-vessel H2 generation from thermal decomposition (no steam oxidation) of 2427-kg Hypalon in Fermi-2 was ~ 43 kg using data presented by M. Salay in 2021.

Potential Gas Generation in the 1F3 Accident

		Potential gas generation (kg)			Total potential gas generation (kg)
	TEPCO's Estimated Mass in 1F3 (kg)[1]	H2	CO	Hydrocarbons CH4 equivalent	H2 equivalent
CV Cable	3000	27.05	73.11	128.97	86.97
PN Cable (+ KGB cable)	830	29.49	168.03	34.02	57.83
Co-axial Cable	320	0.96	4.68	6.22	3.95
Polyimide	6	0.34	2.95	0.12	0.64
Epoxy Paint	442	5.17	11.32	9.06	9.90
Urathane	280	6.60	40.59	11.32	14.74
Total		70	301	190	174

Source: Nuclear Regulation Authority

[1]https://www.nra.go.jp/data/000388502.pdf (4/26/2022)

Potential Sources of H2 Generation in the 1F3 Accident

Core Component	Potential source (kg)	Potential H2 production[1] (kg)	Potential H2 production %
Zr in Fuel cladding	29000	1272	40.8%
Zr in Channel box	18000	789	25.3%
Fe in Absorber	12800	641	20.6%
B4C in Absorber	960	243	7.8%
Steam oxidation of hydrocarbon polymers	4878	174	5.6%
Total (kg)		3119.00	100.0%

New addition

[1]Data for core components based on information in Luangdilok, Nuclear Eng. & Design 362 (June 2020) 110536

The potential ex-vessel generation amount of ~174 kg of H2 equivalent from steam oxidation of cables and paint can be marginal to substantial for 1F3 given large uncertainties

BSAF Phase 2 Analyses of 1 F3 Accident

BSAF Phase 2 Participating organization	SA code used in the analysis for BSAF Phase 2	Calculated in-vessel H2 generation (kg)	Calculated ex-vessel H2 generation (kg)	Total calculated H2 generation (kg)	Calculated RPV failure time (hr)	RPV failure mode
VTT	MELCOR	1220	1200	2420	43.3	penetration
SNL	MELCOR	1010	700	1710	58	user specified
JAEA	THALES/KICHE	790	875	1665	46.5	vessel melt
IAE	SAMPSON	790	500	1290	55.2	creep
PSI	MELCOR	1180	0	1180	73.1	penetration
IRSN	ASTEC	1150	0	1150	55.4	creep
NRA	MELCOR	910	100	1010	49.4	penetration
CRIEPI	MAAP5	600	0	600	102	penetration

[1]Data for H2 gas generation based on information in Luangdilok, Nuclear Eng. & Design 362 (June 2020) 110536

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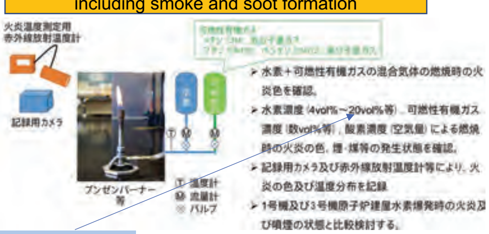
NRAJ Proposed Combustion Experiments[1]

as of September 6, 2022

Source: Nuclear Regulation Authority
[1]https://www.nra.go.jp/data/000403170.pdf (9/6/2022)

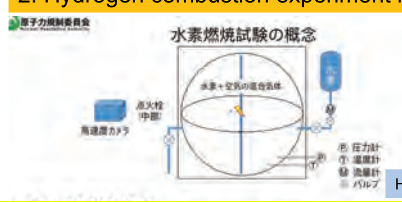
1. Proposed mixed gas flame color experiments including smoke and soot formation



H2: 4~30 vol.%

CH_4 & C_4H_{10}: 0~5%

2. Hydrogen combustion experiment in a sphere



H2: 4~20 vol.%

Proposed range of hydrogen concentrations for investigation is too narrow. The hydrogen-rich side of the flammability limit is the concentration range that is relevant to 1F3 explosion.

3. Mixed hydrogen-hydrocarbon gas combustion experiment in a sphere

H2: 4~20 vol.%

CH_4 & C_4H_{10}: 0~5%

This is an experiment to study the chemistry aspect of the combustion of mixed gases. The fluid dynamics aspect of the problem also needs to be addressed.

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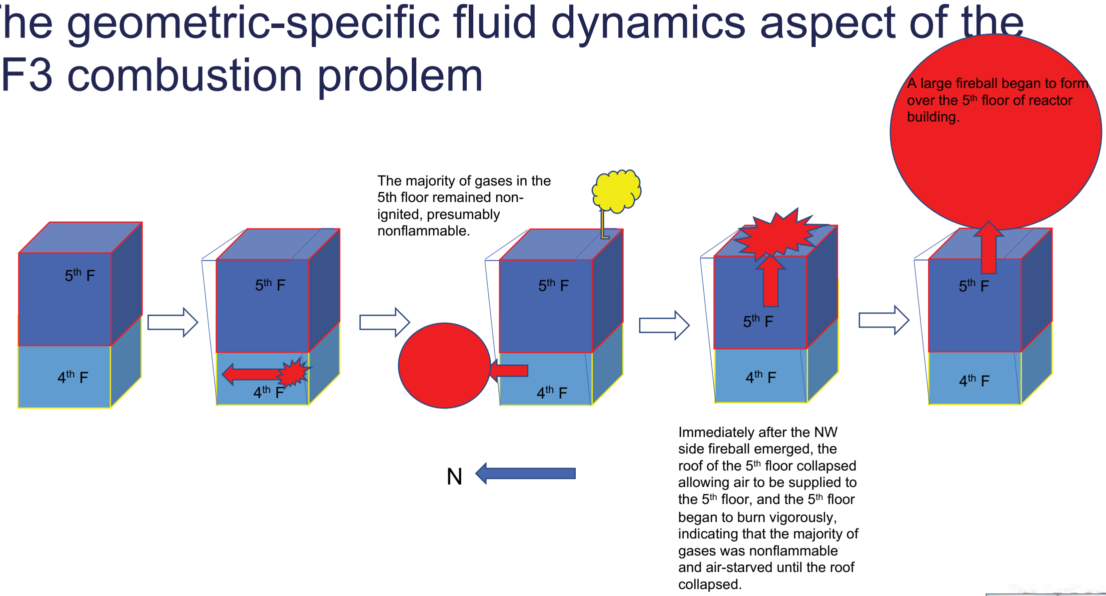
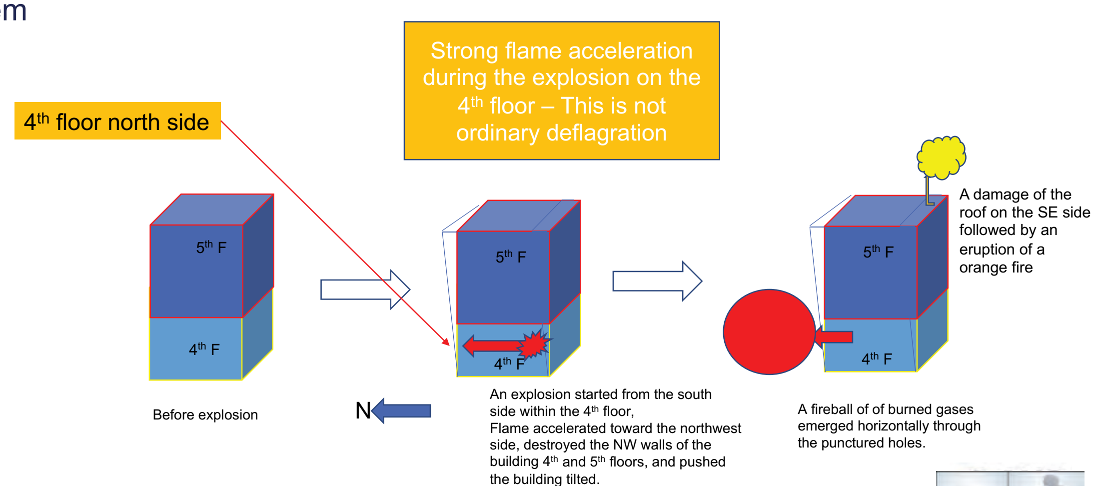
Comments on NRAJ Proposed Combustion Experiments

- Proposed range of hydrogen concentrations for investigation is too narrow. The hydrogen-rich side of the flammability limit is the concentration range that is relevant to 1F3 explosion.
- Proposed experiments are good to study the chemistry aspect of the combustion of mixed gases.
- The geometric-specific fluid dynamics aspect of the problem also needs to be addressed.
- Consider a combustion experiment to investigate the concentration of mixed gases that would produce a dynamical aspect with a characteristic of the 1F3 explosion on the 5th floor.
- Investigate the strong flame acceleration pointed toward the 1F3 North wall in the 4th floor explosion (if there is an idea!)

The geometric-specific fluid dynamics aspect of the 1F3 combustion problem



The geometric-specific fluid dynamics aspect of the 1F3 combustion problem



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Hydrogen Concentration Range for 1F3 Explosion

Long-stem, Mushroom-shaped Fireball Evolution Φ> 1 Experiments

Guo, et al. 2015, Effect of ignition position on vented hydrogen-air explosions, Int. J. Hydrogen Energy 40 (2015) 15780-15788

Figure shows the length of the mushroom stem increases with an increasing equivalence ratio.

Conclusion: The 1F3 explosion on the 5th floor was highly likely an event with **Φ> 1.**

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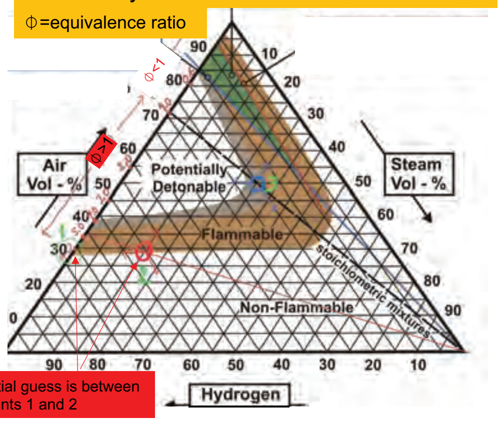
Hydrogen Concentration Range for 1F3 Explosion

Range Φ<1 → provide basic understanding
Range Φ>1 → provide answers to 1F3
Uncertainty → Steam concentration

Φ=equivalence ratio



Initial guess is between points 1 and 2

A satellite photo of steam over 1F3 taken 3 minutes after the 1F3 explosion is not necessarily evidence that there is too much steam at the time of the explosion

The RPV drywell head had been leaking for 6 minutes prior to the explosion at 11:01 am. During this time the drywell pressure dropped from 5.2 bar to 4.8 bar. This initial leak would have taken out stratified hydrogen at the top of the drywell out first before steam.

From 11:02 to 11:15 am, the drywell pressure dropped further from 4.8 bar to 3.9 bar. The satellite picture was taken during this time when a significant leak of steam was going on.

Important questions that should be answered by more research on the geometry-specific fluid-dynamics aspect of the problem

- Can an explosion similar to this type cause water splashing out of the spent fuel pool ?
- What damages on the RB 5th floor were caused by the 5th floor fireball vs the explosion initiated on the 4th floor.
- Where was the center of the 5th floor explosion? What factors determine it?
- Is there a condition that could lead to the center of the fireball explosion close to the spent fuel pool?

C.3.5. Topic Area 5 - Operations and Maintenance

C.3.5.1. BWR Industry Updates

BWR Industry Updates on Terry Turbines and Innovations Spawned from the Fukushima Accident Response and Decommissioning

November 17-18, 2022

Randy Bunt, Southern Nuclear

Actions to Prevent and/or Mitigate Fuel Damage for BDBEs

- U.S. Industry Diverse and Flexible Coping (FLEX) Program
  - Plant sites maintain additional equipment for water injection, power restoration ,and debris removal
  - Similar equipment at two national response centers
- Improved spent fuel pool (SFP) level water level instrumentation and strategies to address challenges to SFP cooling
- Hardened containment wetwell vent (BWR I and II containments)
- Alternate venting and water addition strategies
- Revised procedures and guidance and updated training
  - BWROG computer-based Severe Accident Interactive Learning (SAIL) training and guidance.

Terry™ Turbine Testing and Follow-on Actions

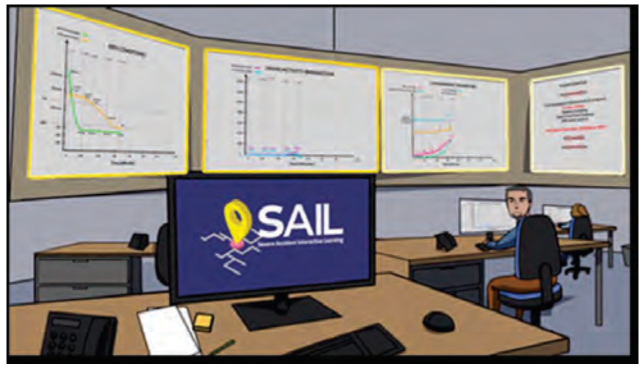
- BWROG-led Terry™ Turbine Expanded Operating Band Project (TTEXOB) expanded and defined actual operating limitations of Terry™ turbine systems (i.e., RCIC/TDAFW)
- Concluded the 1F3 RCIC operation based on available plant instrumentation data is repeatable based on insights from bi-lateral Japan/U.S. Terry™ turbine testing program
- Provided information for RCIC system Terry™ turbine and HPCI system performance models
- New RCIC models benchmarked using Tennessee Valley Authority data in which RCIC system ran on April 27, 2011 after a tornado
- Provided data on no impact on performance for higher bearing oil temperatures
- Provided data to support basis of relaxing the low pressure start up test for Terry™ turbines (eminent submittal for BWROG topical report)

Graphics courtesy of Texas A&M University

Emergency Procedure Lessons Learned and Application

- The post Fukushima EOP changes to Level 8 trip inhibits important in maintaining RCIC and HPCI system performance and in reducing SRV cycling
- The DAEC event re-emphasizes need for symptom-based procedures for EPG/SAGs and FLEX
- Procedures and proficiency important to restoring systems out of service for testing or maintenance and for returning failed systems to operation during a LOOP
- Modeling of event assumptions needs to be consistent with actual plant operations or conditions
- RCIC testing provides specifics about turbine and pump operation that improve modeling
- Plant transient response was as expected and agreed with simulator training for LOOP response

BWROG Emergency Procedures SAIL Training



Graphics courtesy of BWR Owners Group

New technologies facilitate 1F D&D

- Muon tomography
- Special-purpose robots, drones, and Unmanned Aerial Vehicles (UAVs)
- Portable gamma-ray imaging camera
- Infrared thermography
- Real-time monitoring with 2Dor 3D visualization of radiation levels and temperatures
- Plastic scintillation fiber monitors
- Centralized data system to optimize worker exposure

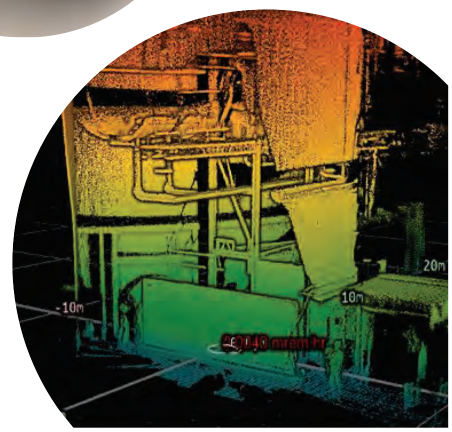
These new technologies offer the potential to improve plant operations and maintenance

Graphics Courtesy of TEPCO, IRID, and JAEA

Examples of RP New Tech Uses from BWROG TP19-1-062r1

- "Drones have the capability of viewing a space or area without disturbing the floor surface contamination or exposing an individual to potentially high dose rates in an area. Drones can access areas where scaffolding, confined space, or other fall protection may be required, increasing safety margin. Drawbacks associated with drone use can include a potential to create airborne radioactivity in spaces with fine, loose surface contamination, a FAA pilot's license is required if used outdoors, radio-sensitive equipment interference, the potential for a drone to physically impact plant equipment (collision), loss of radio signal within a space may require entry in order to retrieve the drone, and cost can be prohibitive in some circumstances."

- "LIDAR: LIDAR technology allows for the laser and 3-D modeling for rooms in which engineering can perform measurements via software. This eliminates the use of scaffolds and/or entrances into LHRA/HRA areas to get measurements for mods and equipment fixes. Can also be used as a valuable training tool to give new team members a general idea of how the room is laid out."



Graphics courtesy of BWR Owners Group

C.3.5.2. Update on Efforts to Launch New Technologies Research Effort

U.S. DEPARTMENT OF ENERGY
Nuclear Energy

New Technologies for Advanced and Existing Reactor Maintenance and Recovery Activities (including Decommissioning)

1

U.S. DEPARTMENT OF ENERGY
Nuclear Energy

Objective

To launch a joint U.S. – Japan research program to develop and deploy new technologies to support activities beneficial for advanced and existing nuclear reactor maintenance and recovery activities (including decommissioning)

2

U.S. DEPARTMENT OF ENERGY
Nuclear Energy

Background

- Japan developing and deploying new technologies for surveying, characterization, stabilization, decontamination, and waste minimization
- These technologies could benefit routine plant O&M activities
  - Potentially offering increased efficiency, reduced cost, reduced waste generation, improved schedule, and improved safety)
  - Applicable to operating LWRs and new LWRs and non-LWRs

3

U.S. DEPARTMENT OF ENERGY
Nuclear Energy

FY21 Activities

- FY 21 Report 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.

- DOE and BWROG developed draft document that:
  - Identifies new technologies/measures (with sample two-page brochures for selected technologies)
  - Presents information to characterize effectiveness and areas where future research would be beneficial for routine O&M activities

4

FY22 Activities

- JAEA presented updated research results at FY22 meeting
- Expert panel agreed the DOE/BWROG draft document should be updated to consider information presented by Japan.
- FY 22 Report Recommendations:

  The draft information highlighting new D&D should be included as an appendix to this FY2022 report. [This information appears as Appendix E.3 of report]

  Additional efforts should be devoted to facilitate deployment of new D&D technologies from 1F for routine O&M activities. It is suggested that this workscope be funded under the DOE LWRS program.

5

FY22 Activities (continued)

- MEXT/DOE developed draft proposal for joint research program on new technologies to support advanced and existing nuclear reactor maintenance and recovery activities (including decommissioning)
  - Proposal emphasizes new technologies to support measurement, analysis and visualization techniques
  - BWROG provided input on possible activities where substantial costs and dose savings could be obtained with such technology development.
- Draft proposal socialized with various organizations in the US (e.g., INL, DOE, etc.) and Japan (e.g., JAEA, NDF, etc.)

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Next Steps

- **Continue emphasizing need for this research program**
  - **Requires lead organization (e.g., university, national laboratory, industry), industry support (possible pilot deployments?) and commercial industry partner**
  - **Combined Owner Group (PWROG and BWROG), DOE NE-6, and EPRI support could be helpful**

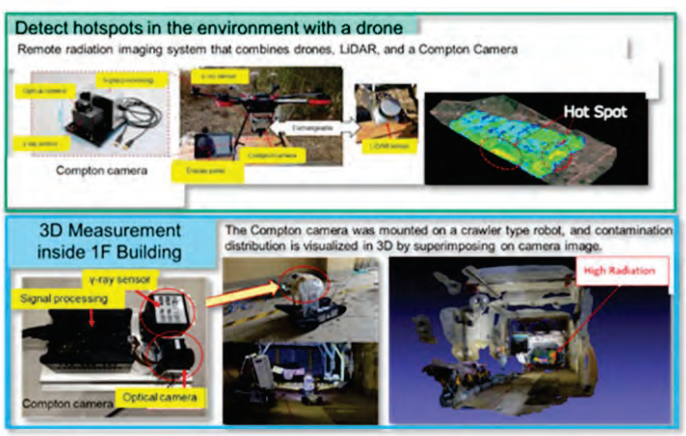
- **Propose FY23 Report recommendation:**

  Additional efforts should be devoted to facilitate deployment of new D&D technologies from 1F for routine O&M activities. A joint US/Japan NEUP program should be initiated.

Backup

U.S. DEPARTMENT OF ENERGY
Nuclear Energy

Gamma Cameras provide real-time radiation measurement and detection to facilitate D&D



Graphics courtesy JAEA

- ***Compact portable systems for remote detection***
- ***Can deploy with special-purpose robots, drones, and UAVs***
- ***Can combine with software for real-time monitoring in easy-to-understand 2D and 3D visualizations showing photos, radiation levels, and temperatures***

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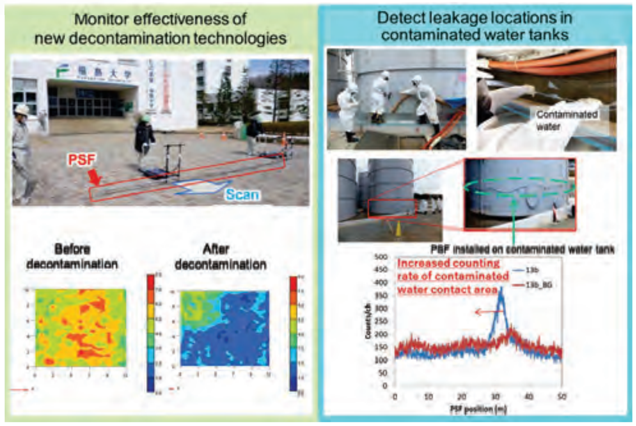
Plastic Scintillation Fibers (PSFs) provide option for real-time radiation detection and monitoring



Graphics courtesy JAEA

- ***Remote method for real-time detection of radiation (contaminated water leakage and D&D effectiveness)***
- ***Simulataneous detection of β -and γ –radiation***
- ***Can deploy with special-purpose robots, drones, and UAVs***
- ***Can combine with software for real-time monitoring in easy-to-understand 2D and 3D visualizations***

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

C.4.1. Related EPRI Activities

Related EPRI Activities

Recent MAAP Enhancements, PCMQ Methods, and Info Request

Matt Nudi
EPRI, Risk & Safety Management

Reactor Safety Technology Expert Panel Forensics Meeting
November 18, 2022

www.epri.com

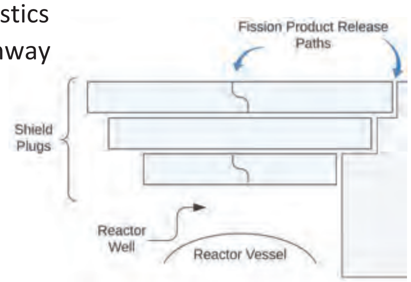
Outline

- Additional details for shield plug contamination investigation
- Related MAAPv5.06 Enhancements
  - MAAPv5.06 is the latest production release of the MAAP code
- Predictive Capability Maturity Quantification (PCMQ) methods

2 EPRI

EPRI Data Needs for Shield Plug Contamination Analysis

- Intention is to develop a one-way coupling for MAAP & GOTHIC to assess ability to analytically reproduce shield plug contamination observed
  - MAAP best-estimate analysis to provide DW boundary conditions and FP release characteristics
  - GOTHIC detailed 3D analysis of FP release pathway through shield plug
- Important data inputs needed:
  - nominal gap area in shield plug plates
  - estimate gap area following shield plug deformation



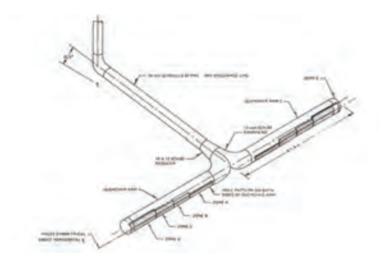
Related MAAP 5.06 Enhancements

Overview of major related MAAPv5.06 enhancements

- Extend ex-vessel relocation model to PWR
- Improve corium jet fragmentation heat transfer models
- **Modeling of X-Quencher in BWR Suppression Pools**
- **Improvements to fission product scrubbing model in the core**
- **Implement modeling capability for user-specified values for gamma induced water radiolysis (G-value) in the code**

Modeling of X-Quencher in BWR Suppression Pools

- Fukushima Daiichi suppression pool was a T-quencher design.
- Full scale plant testing at the Monticello plant and instrument readings from the Fukushima Daiichi units indicated the T-quencher design had a propensity for thermal stratification in the suppression pool during RPV depressurization. Modeling of T-quenchers was added in MAAPv5.04.
- Other plant designs like advanced Mark II designs improved on the T- quencher design through the addition of the X-quencher design
- Data from Fukushima Daiini demonstrated that better mixing was achieved with this design.



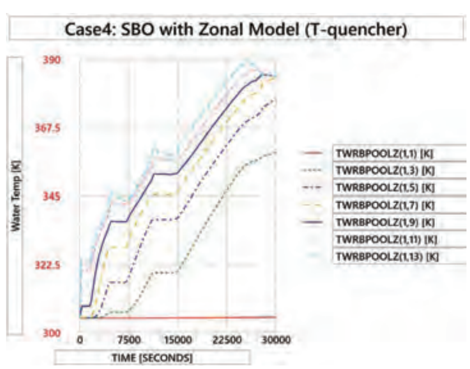
Ref: MAAPv5.06 Volume 3, Benchmarking – Volume 3C Monticello SRV Discharge Test Benchmark

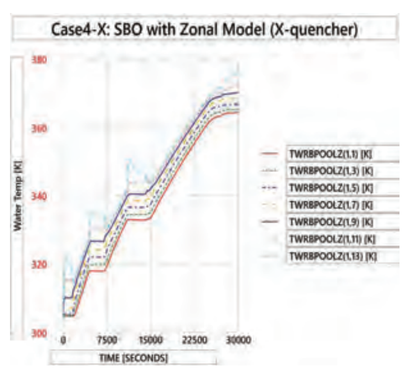
Modeling of X-Quencher in BWR Suppression Pools (cont'd)

- A simple X-quencher model is implemented to allow the user to account for the improved mixing characteristics of SRV discharge via X-quenchers.
- X-quencher is modeled when the vessel is at a pressure significantly higher than the suppression pool and there is discharge through the X-quencher.
- When the model is activated, the water in the nodes at the discharge elevation and above will be mixed at a rate calculated by subroutine REMIXZ.
  - the user will have the option of overriding the Ricou-Spalding entrainment coefficient for plume entrainment with separate user input parameter

Impact of X-Quencher compared to T-Quencher (cont'd)

SBO case with 3 node suppression pool (TDBATT= 4 hr) with the zonal model





New Pool Scrubbing (Water on top of damaged core)

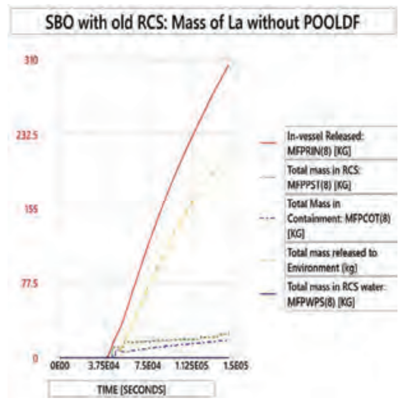
- In pre-MAAP5.06 codes, fission products released from a damaged core are added to the upper plenum regardless of whether the core is covered or not.
- In MAAP5.06, pool scrubbing is added when the core is submerged to reduce the fission product release to the environment
  - This is expected to affect low-volatile fission product release when flag to allow low volatile fission product release from molten pool enabled
  - Model approach is similar to pool scrubbing on top of corium pool in containment
    - All fission products released from the damaged core are treated as aerosols with a size of XRDB (0.01 micron)

New Pool Scrubbing (Water on top of damaged core)

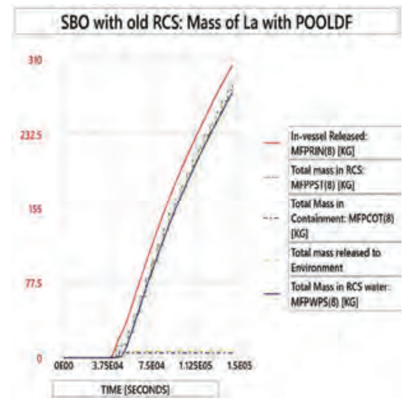
- During the recovery of damaged core, two conditions in the core can be considered:
  - For intact fuel rods covered by water, the fuel rods will be cooled quickly, and no fission product will be released because the fuel temperature is too low.
  - For a damaged core or in-core molten pool, most of volatile fission products are already released. Release of fission products from the in-core molten pool is allowed based on user input parameters
    - When enabled, release of low volatile fission products from the water-covered molten pool is allowed and the pool scrubbing model will reduce the overall release to the gas region.
- **Impact:** Reduced low volatile fission product release from the vessel to containment when core is submerged.

POOL Scrubbing (SBO with In-vessel Water Injection at 41,500 s)

- La release without pool scrubbing



- La release with pool scrubbing



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New User-Input G-value Radiolysis model

- A new, alternate radiolysis model was added to the MAAP5.06 BWR code (in-vessel and containment) and PWR code for containment only
- New model is based on
  - User input G-value (Hydrogen yield rate due to beta and gamma radiation)
  - User-input radiation absorption fraction (E)
  - MAAP calculated decay heat
    - Decay heat associated with air-borne and deposited fission products
    - Decay heat in the core
    - Decay heat in corium
    - Decay heat from fission products in water
- Model important for evaluating long-term effects

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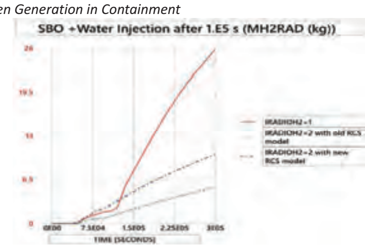
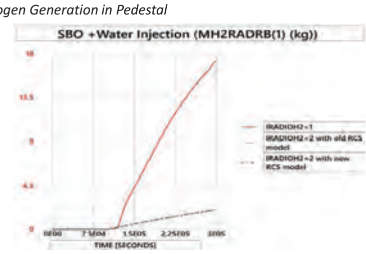
Hydrogen Generation by Radiolysis (sample SBO)

- IRADIOH2=1 : Use existing model
- IRADIOH2=2 : Use user-input G-value model
- G-value for containment = 0.06
- Radiation absorption fraction for decay heat in the water and on submerged heat sinks = 1.0
- Radiation absorption fraction for decay heat in the submerged core or submerged corium= 0.1
- Radiation absorption fraction for decay heat in the gas region = 1.0

Hydrogen Generation in Containment



Hydrogen Generation in Pedestal



Predictive Capability Maturity Quantification (PCMQ) Methods

Introduction

- In code development, V&V, and application, there are always questions about:
  - Which experiments/benchmarks are most applicable to a particular application?
  - Is the available benchmarking sufficient?
  - What improvements or additional benchmarks would provide the **most value**?
- Currently investigating the feasibility of methods for answering these types of questions using the existing testing suites for MAAP & GOTHIC.

Background

- Several approaches have been proposed to assess the maturity of Modeling and Simulation (M&S) tools,
  - EMDAP (Evaluation Model for Development and Assessment Process)
  - CSAU (Code scaling, Applicability, and Uncertainty)
- Included as part of Reg. Guide 1.203 – *Transient and Accident Analysis Methods*
- Credibility assessment to evaluate the adequacy and trustworthiness of the software, assumptions, inputs, and methods (Evaluation Model) for an intended application

Motivation

- While structured, these determinations tend to be primarily qualitative in nature
  - Generally, relies on expert knowledge and engineering judgement for determining adequacy
  - May work for well-established applications, but not first-of-a-kind (FOAK) instances where expertise may be lacking.

- Predictive Capability Maturity Quantification (PCMQ) has the potential to:
  1. Characterize the domain covered by the experiments, software V&V and application.
  2. Quantify the overlap in these domains

Potential Benefits

- **End User**
  - Need to identify which experiments and software benchmarks are most applicable to their intended application
  - Provide the justification that the evaluation model (software, inputs, assumptions, etc.) is applicable.

- **Software Developers**
  - Evaluating sufficiency of coverage of their existing V&V and identifying gaps

- **Product Managers**
  - Evaluating and prioritizing areas for improvement or additional experiments/ benchmarks that will close gaps and provide the most value in improving the credibility of the evaluation model for various intended applications.

PCMQ Approach

- Show evidence that support the claim that the code is adequate/mature to simulate specific phenomena.
- This evidence is supported by
  - Direct attributes
  - Process Quality Assurance (PQA) attributes
- The direct attributes-evidence includes the assessment of the data uncertainty, scaling, relevance/applicability, data coverage, and the accuracy relative to figures of merit (FOM).
- The Process Quality Assurance considers the personnel, M&S tools/techniques, evidence level of details, credibility and scaling methodology, and validation metrics.

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Technical Approach

- Work in 2022 provided a proof-of-concept to
  - Demonstrate the value of PCMQ methods to the nuclear industry
  - Transition these methods from the R&D stage to the application stage
- Case Study: 2D Thermally Driven Cavity
- Steps:
  - PIRT table
  - Maturity Level Assignment for each evidence (E_i)
    - Validation Results
    - Data Applicability (evaluates relevance, uncertainty and scaling)
  - Evidence Integration: Formulating of the Decision model using Bayesian network
  - Sensitivity Analysis

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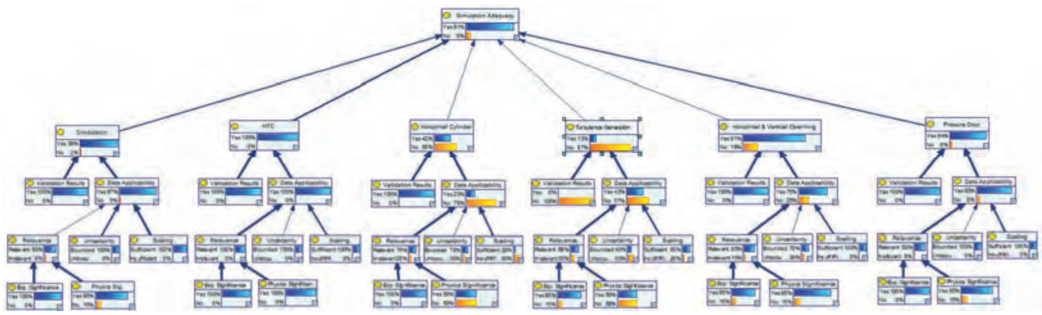
Decision Model – Bayesian Network



- The arrow's thickness reflects the node's influence on the target state

The PCMQ results were found consistent with expectations

Future Work

- Implementing the PCMQ approach on integral application over multiple systems.
- Defining approach to make qualitative elements more quantitative, or at least less dependent on the expert opinion.
- Sensitivity to qualitative "big-picture" elements of the PCMQ method (e.g., phenomena ranking and identification)
- Proof-of-concept application of PCMQ on other codes using the GOTHIC case results.

Conclusion

- Successfully demonstrated use of PCMQ methods on a simple and well-established GOTHIC application
- The current work provides a proof-of-concept to
  - Demonstrate the value of PCMQ methods to the nuclear industry
  - Transition these methods from the R&D stage to the application stage
  - Provide a framework for evaluating which experiments and benchmarks are most important for an application and guiding which additional benchmarks would be most valuable
- Relative to MAAP and Fukushima-related activities
  - Future work will evaluate use of PCMQ methods on larger, integral MAAP & GOTHIC applications
  - PCMQ process expected to provide quantitative method to identify highest priority benchmarks and model development items

C.4.2. MELCOR Considerations

Overview

Severe accident knowledge evolution in light of Fukushima Daiichi

- Areas of focus given influence identified in uncertainty analyses
- Areas of interest to evolve state-of-practice in severe accident modeling
  - Extensive involvement in international programs

2

Insights from Severe Accident Uncertainty Assessment

What do our existing uncertainty assessments teach us about important areas of focus?

November 15, 2022

3

Objectives of the SOARCA Uncertainty Assessments

Considering one accident scenario specific to each of the Peach Bottom, Surry and Sequoyah plants:

- Identify the uncertain input parameters potentially influential to accident progression and source term
- Define informed distributions for the possible values of the uncertain parameters
- Randomly exercise for the specific scenario, thru Monte Carlo sampling, a MELCOR model of the plant across the possible values of the uncertain parameters generating a distribution of source term outcomes
- Determine from the distribution of outcomes the importance of the uncertain parameters relative to the metrics of Cs and I release to the environment
- Identify the variations in accident phenomena driving differences in the Cs and I release metrics
- Identify the linkages between the uncertain parameters and the driving phenomena
- Develop insight into overall sensitivity of results and conclusions from the original SOARCA studies to uncertainty in model inputs

4

Peach Bottom Uncertainty Assessment

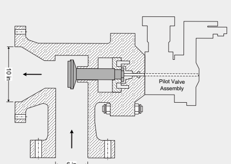
Expanded upon one particular severe accident scenario addressed in SOARCA

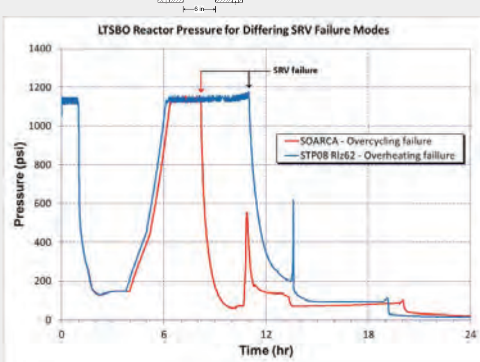
- Long-term station blackout involving one of the two Peach Bottom (BWR/4 Mark I) units

Considered releases to the environment out to 48 hr from the loss of all AC power

Involved 900 MELCOR simulations 865 of which completed successfully each taking 2+ days of CPU time

Important Peach Bottom Modeling – SRV Failure





Threats to an SRV

- Excessive cycling
- Excessive temperature
  - Differential thermal expansion
  - Plastic deformation

SRV cycles for hours in LTSBO

Failure assumed to be seizure in the open position

- Fully open in the case of over-cycling
- Fractionally open (0 to 1) given overheating

Failure in the closed position not addressed

- Assumed that pressure relief would simply move up to the next of 11 total valves

Images courtesy of Sandia National Laboratories

Peach Bottom Uncertain Parameters – Source Term

Uncertain Parameter (21 total)	Distribution type	Mode/α,β	Lower bound	Upper bound
Sequence-Related Parameters				
SRV stochastic failure to reclose (SRVLAM)	Beta	α 0.494 β 133.2	0	0
Battery Duration (BATTDUR), hr	Log triangle	4	2	8
In-Vessel Accident Progression Parameters				
Zircaloy melt breakout temperature (SC1131(2)), K	Triangle	2,400	2,098	2,550
Molten clad drainage rate (SC1141(2)), kg/m-s	Log triangle	0.2	0.1	1
SRV thermal seizure criterion (SRVFAILT), K	Beta	α 2.72 β 6.79	811	1,143
SRV open area fraction (SRVOAFRAC)	Triangle	1	0.1	1
Main Steam line creep rupture area fraction (SLCRFRAC)	Discrete		5%	100%
Fuel failure criterion (FFC)	Discrete		BE -100 K, 0.5x dt	BE +100 K, 2x dt
Solid debris radial relocation time constant (RDSTC), s	Log triangle	360	180	720
Molten debris radial relocation time constant (RDMTC), s	Log triangle	60	30	120
Ex-Vessel Accident Progression Parameters				
Debris lateral relocation – cavity spillover and spreading rate (DHEADSOL), m	Triangle	0.5	0.1	1
Debris lateral relocation – cavity spillover and spreading rate (DHEADLIQ), m	Triangle	0.1524	0.05	0.25

Images courtesy of Sandia National Laboratories

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Peach Bottom Uncertain Parameters – Source Term (2)

Uncertain Parameter (21 total)	Distribution type	Mode/α,β	Lower bound	Upper bound
Containment Behavior Parameters				
Drywell liner failure flow area (FL904A), fraction of 1 m2	Log uniform		0.05	1
Hydrogen ignition criteria (H2IGNC), mole fraction	Triangle	0.1	0.04	0.2
Railroad door open fraction (RRIDRFAC, RRODRFAC)	Uniform		0.05	0.75
Drywell head flange leakage, K	Triangle	0.08	0.029	0.57
Drywell head flange leakage (E), Pa	Uniform		1.834E+11	2.027E+11
Drywell head flange leakage (δ), m	Uniform		6.604E-04	8.363E-04
Chemical Forms of Iodine and Cesium				
Iodine and Cesium fraction (CHEMFORM)	Discrete			
Aerosol Deposition				
Particle density (RHONOM), kg/m3	Triangle	1,000	870	4,037

Images courtesy of Sandia National Laboratories

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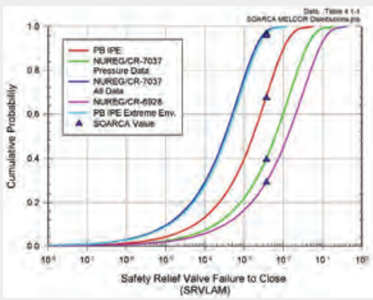
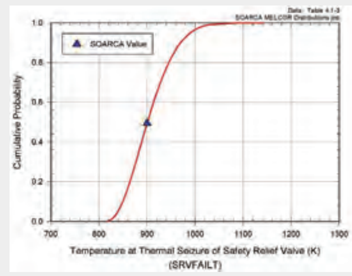
Important Peach Bottom Uncertainty - SRV Failure

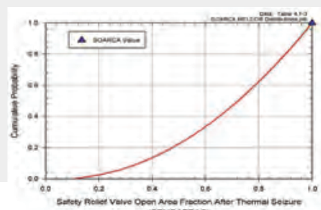
SRV stochastic failure to reclose (SRVLAM) - Beta distribution fit to mean value in Peach Bottom IPE (the SOARCA value) using the methodology in NUREG/CR-7037

SRV thermal seizure criterion (temperature) (SRVFAILT)

SRV open fraction given thermal seizure (SRVOAFRAC)

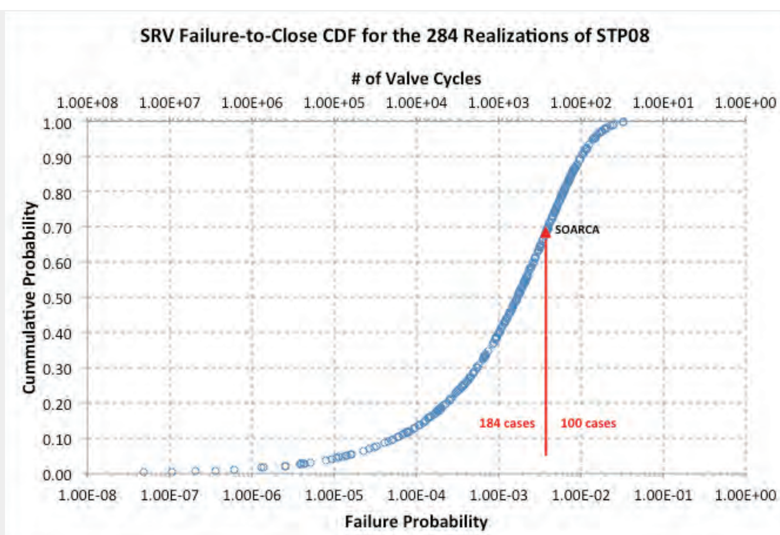






Images courtesy of Sandia National Laboratories

Sampled Peach Bottom SRV Failure-to-Close Probability



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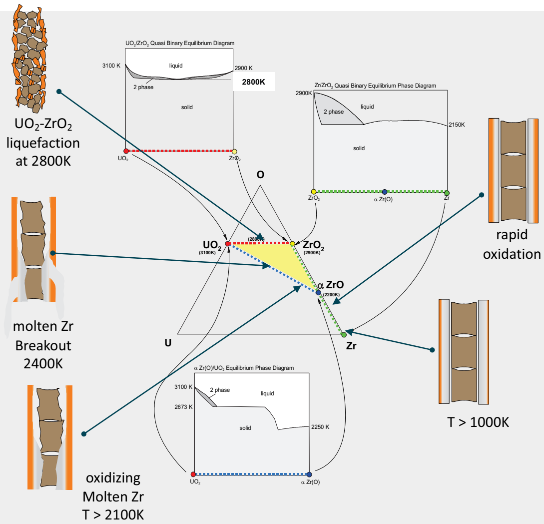
Important Peach Bottom Uncertainty - Fuel Damage Progression



Zr metal melting at 2,150 K

Uncertain parameters:

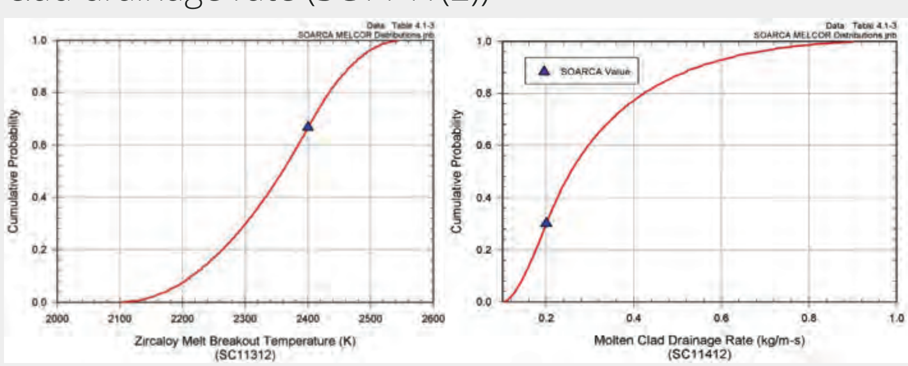
- UO2 dissolution in Zr metal ~20%
- Zr melt breakout ~2400K
- Loss of rod geometry ~2600K
- UO2-ZrO2 liquefaction ~2800K

Images courtesy of Sandia National Laboratories

Important Peach Bottom Uncertainty - In-Vessel Accident Progression Parameters

Zircaloy melt breakout temperature (SC1131(2))

Molten clad drainage rate (SC1141(2))



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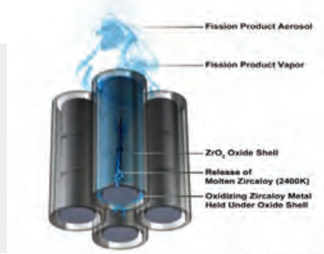
Uncertain In-Vessel Accident Progression Parameters (2)

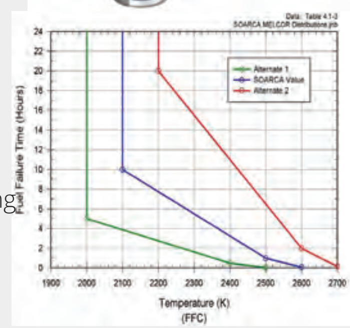
MELCOR lacks a deterministic model for evaluating fuel mechanical response to the effects of clad oxidation, material interactions (i.e., eutectic formation), zircaloy melting, fuel swelling and other processes that occur at very high temperatures

In lieu of detailed models in these areas, a temperature-based criterion is used to define the threshold beyond which original fuel rod geometry can no longer be maintained

Time-at-temperature criterion

- Fuel rod decreasing endurance with increasing temperature
- Assumptive of oxidized cladding
- Alternative 1 is derived from the best estimate by reducing temperatures by 100 K and halving time intervals
- Alternative 2 is derived from the best estimate
  - Temperatures increased by 100 K and time intervals doubled





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Important Peach Bottom Uncertainty - Chemical Forms of I and Cs

Based on Phebus tests, SOARCA treated:

- I as CsI
- Cs as a particular combination of CsI and Cs_2MoO_4

Cs_2MoO_4 considerably less volatile than CsOH or CsI

Peach Bottom uncertainty assessment explored alternative balances of speciation considering the chemical forms I_2, CsOH, CsI and Cs_2MoO_4

- With gaseous I (fraction of initial core inventory) defined between 0 and 0.03 for the 5 combinations, enough Cs was defined as CsI to involve all of the I not defined as gaseous (i.e., most all of the I).
- Remaining Cs was defined in the different combinations to be either in the form of all CsOH, all Cs_2MoO_4, or half CsOH and half Cs_2MoO_4

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Peach Bottom Uncertainty Assessment Results

Basic Statistics of the Peach Bottom UA MELCOR Simulations

- The lowest set-point SRV failed to reclose at some point in every UA calculation
- Stochastic (over-cycle) SRV failure occurred ~43% of the time
- Thermal (overheating) SRV failure occurred ~57% of the time
- An MSL rupture occurred ~18% of the time
- A melting-through of the DW liner by core debris occurred in every UA calculation
- The reactor building railroad access doors blew open over 80% of the time
- An overpressure rupture of the WW happened ~6% of the time

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Peach Bottom Influential Phenomena

Whether the sticking open of the SRV (the lowest setpoint SRV) occurs before or after the onset of core damage

Whether an MSL “creep” rupture occurs

The elapsed time between the onset of core damage and MSL rupture (if an MSL rupture occurs)

The amount of Cs chemisorbed from CsOH into the stainless steel of RPV internals

Whether core debris relocates from the RPV to the reactor cavity all at once or over an extended period of time

The degree of oxidation, primarily Zircaloy oxidation, occurring in-vessel (identified by the amount of in-vessel hydrogen production)

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Peach Bottom Influential Phenomena (2)

Late re-vaporization - most all of the Cs that releases to the environment by 48 hours does so by the following sequential steps:

- Releasing from the dismantling core as CsOH, CsI, or Cs_2MoO_4 vapor
- Condensing into aerosols
- Gravitationally settling onto reactor internals
- Re-vaporizing after RPV lower head failure steadily over approximately the next day
- Re-condensing into aerosols that are carried out a breach in the DW liner resulting from core debris contacting the liner and melting through it

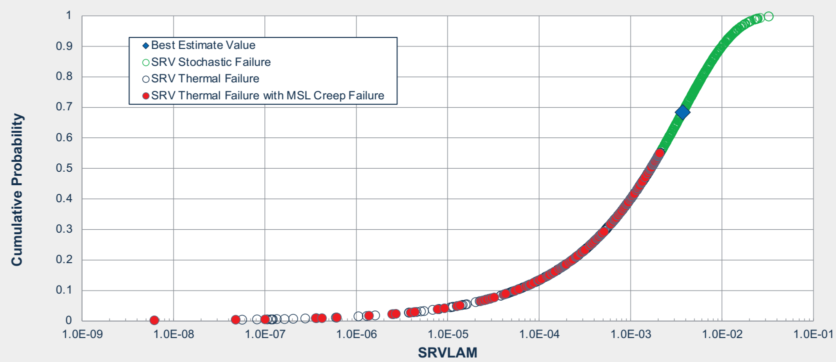
Peach Bottom Influential Phenomena (3)

Whether a surge of water from the wetwell (WW) up onto the drywell (DW) floor occurs at DW liner melt-thru

Whether an overpressure rupture of the WW occurs

Whether the reactor building railroad doors are blown open by a hydrogen deflagration

Influence of # of Cycles to SRV Failure on SRV Failure Mode and on MSL Rupture



- Larger failure probabilities (smaller # of cycles to failure) lead to SRV failures due to excessive cycling or “stochastic” failures
- Smaller failure probabilities (larger # of cycles to failure) lead to SRV failures due to overheating or “thermal” failures
- Thermal SRV failures potentially lead to MSL ruptures

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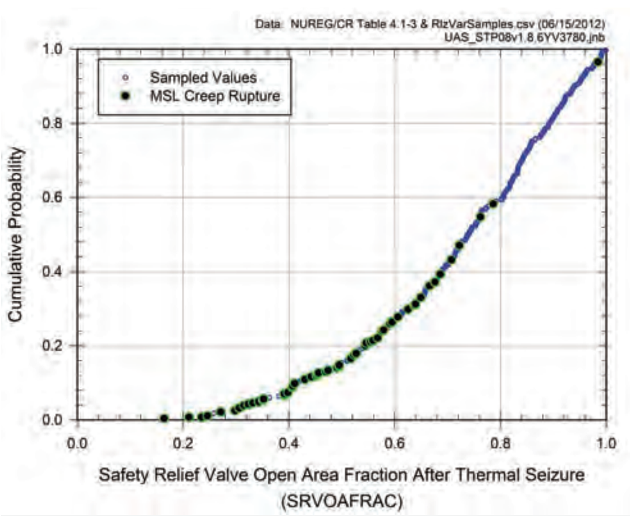
MSL Rupture Dependence on SRV Seized Position



- SRV thermal failure showed to be a necessary condition for MSL rupture
- Very strong dependence of MSL rupture on SRV position at seizure
  - A mostly closed SRV likely resulting in a MSL rupture
  - A mostly open SRV likely ***not*** resulting in a MSL rupture

Images courtesy of Sandia National Laboratories

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Consequences of MSL Rupture

MSL rupture results in:

- The venting of fission products to the DW rather than to the WW
- Lost benefit of pool scrubbing in the WW
- A containment pressure spike unmuted by the WW that unseats the DW head releasing fission products and hydrogen to the reactor building
- WW overpressure rupture in a few cases (which actually showed to have beneficial mitigative influence)

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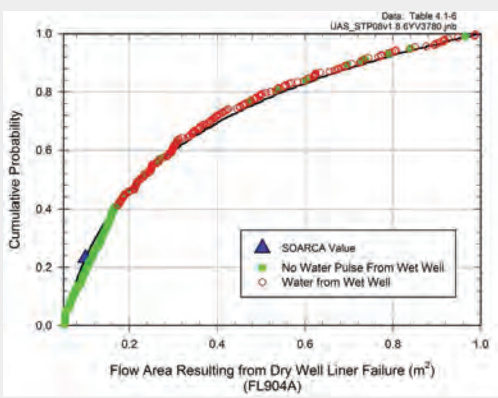
Influence of Drywell Liner Melt-Through Area



Spreading core-concrete debris contacts the liner and melts a hole in it breaching containment

Ensuing containment depressurization results in WW pressure being higher than DW pressure

Areas above a threshold size most often led to the vacuum breakers being overwhelmed which led in turn to RN-laden water being expelled up from the WW out onto the DW floor

Most of water displaced from the WW flowed out the DW liner breach but the DW floor was left flooded to the bottom of the breach (16" above the floor)

Heat from core debris evaporated the DW pool freeing the RNs in the pool to transport further

This expulsion of water up from the WW out onto the DW floor did not happen in the SOARCA calculation

Images courtesy of Sandia National Laboratories

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Summary of Peach Bottom Uncertainty Assessment Insights

In-vessel degradation of critical importance - in particular, early failure of assemblies that could be influenced by material interactions

Vessel depressurization mechanisms have key influence on source term

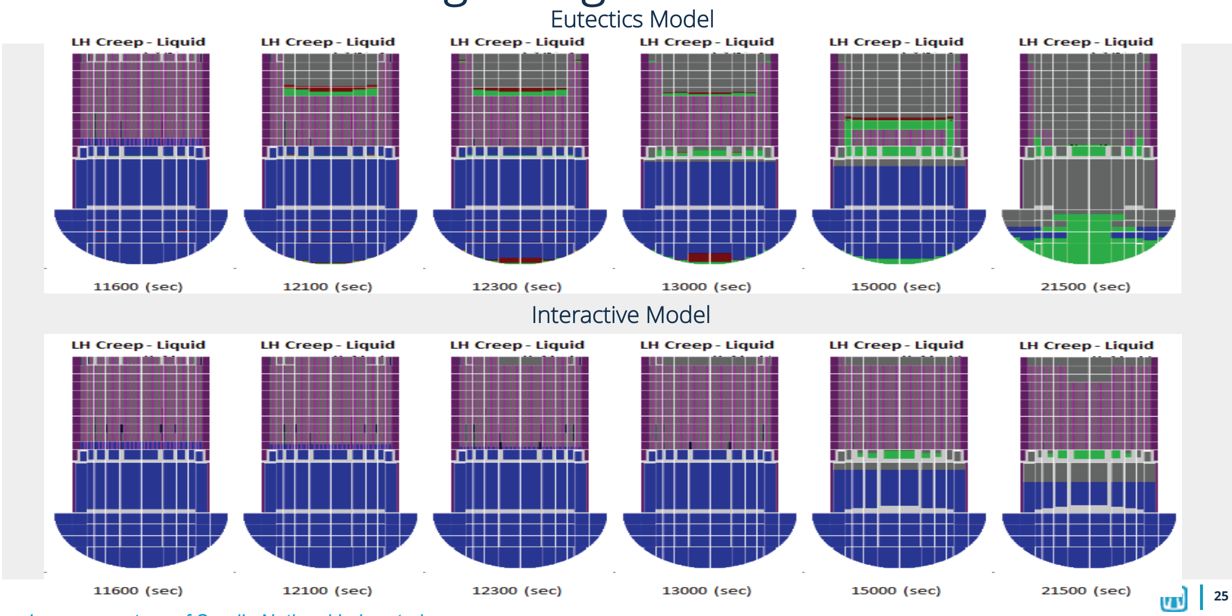
Drywell liner failure important contributor to accident progression and fission product release

Recent MELCOR Model Refinements

Representation of material interactions has been noted as quite important to core damage progression

- Post-Fukushima considerations around enhancements to MELCOR modeling have been under consideration
- Initial assessments to understand how simulation estimates will change have been performed

Overall Core Damage Progression



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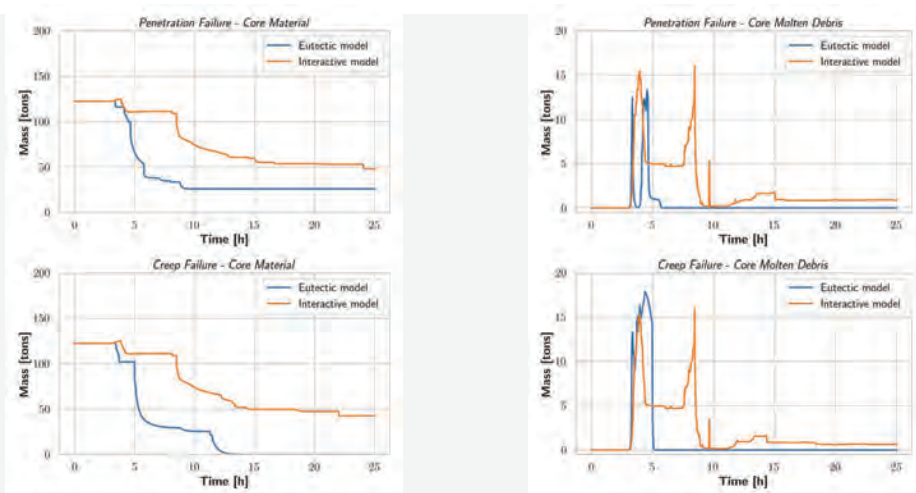
Eutectic Modeling – Molten Debris Formation in Core



Earlier and large slumping to lower plenum with eutectic modeling due to molten material formation

Images courtesy of Sandia National Laboratories

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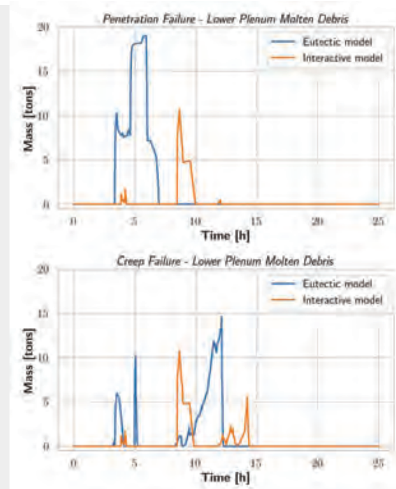
Eutectic Modeling – Lower Plenum Molten Debris

- Eutectic interactions normally lead to earlier melt formation
- Effective interactive material modeling in MELCOR captures as melting point adjustment
- Debris in lower plenum a more complicated mixture of materials
- Assess the impact of eutectic modeling on debris melting in lower plenum

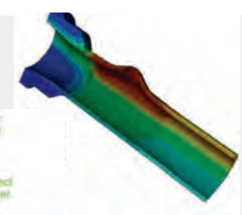


Greater amount of molten material will be available at time of vessel breach for MELCOR eutectic modeling

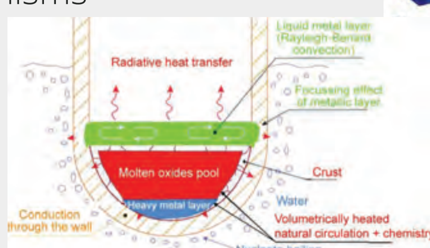
Images courtesy of Sandia National Laboratories

Particular Forensic Items of Interest

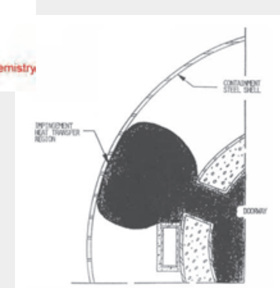
RPV depressurization mechanisms



Vessel integrity and breach



Melt spreading



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APPENDIX D. An Assessment of Crust Bridging Behavior during MCCI in 1F1

Mitch Farmer

D.1. Introduction

A recent evaluation has been carried out by NRA (see Section 2.2.4 and Appendix C.2.4) of information gathered by TEPCO and IRID (see Section 2.2.3 and Appendix C.2.3.1) during ex-vessel examinations in 1F1 (see Figures D-1 and D-2). As the figures indicate, MCCI led to exposure of rebar in the pedestal wall that remained standing after the surrounding concrete decomposed. In addition, there is evidence that crust shelves or ledges have anchored to structures at relatively high elevations (~ 1 meter) above the underlying debris. There is further evidence that this anchored crust material may have broken at some point, leading to a collection of pieces that fell on top of the underlying debris.

The previously mentioned evaluation by NRAJ raised several questions regarding the ex-vessel debris distribution and morphology, as well as the possible physical mechanisms at play that could have led to the final debris state. These questions can be summarized as follows:

- Why did the debris released from the RPV not spread?
- Why was the pedestal wall concrete damaged but not the rebar?
- How was the "suspended bridge crust" material formed?
- What is the source of the white powder that appears to cover some surfaces?

Figure D-1. Debris state in drywell near pedestal doorway; upper surface of the crust material is at ~ 1 meter elevation (Courtesy of TEPCO Holdings, [112])

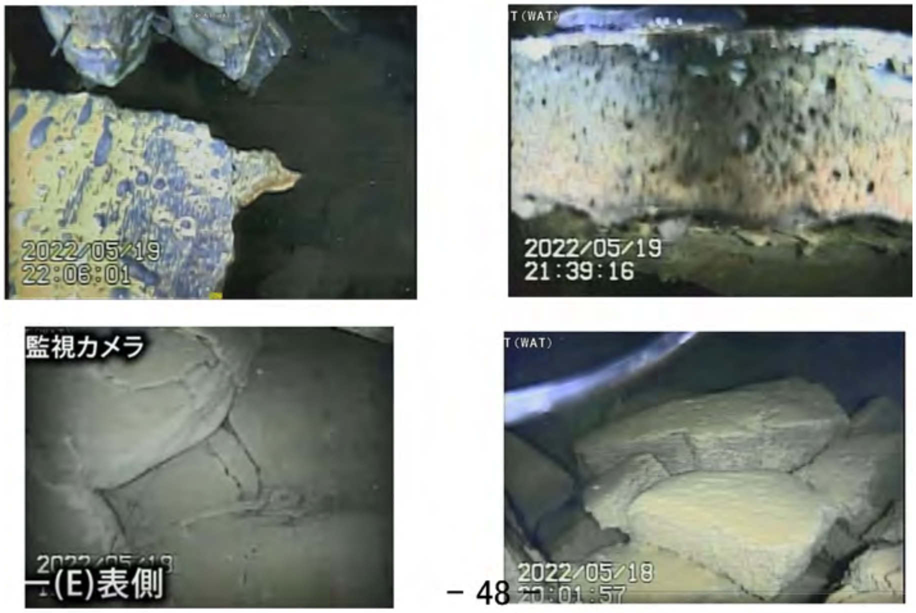

Figure D-2. Details of broken crust material sediment (Courtesy of TEPCO Holdings; [112])

The purpose of this Appendix is to provide potential answers to these questions based on physical observations from past MACE and OECD/MCCI core-concrete interaction and coolability experiments,[113] as well as insights from the crust anchoring model that can be implemented as a user option in the CORQUENCH core debris cooling code.[114]

To this end, a summary review of past MACE and OECD/MCCI tests is first presented, with a focus on using experimental observations to provide insights into questions posed by NRA and others (during the discussion after the presentations by NRAJ and TEPCO on results from 1F1 PCV investigations). CORQUENCH scoping calculations of crust anchoring behavior in 1F1 are then provided to further address these questions from a modeling viewpoint. The Appendix concludes with a discussion of insights gained as part of this study.

D.2. Insights from MACE and OECD/MCCI Tests Relevant to 1F1 Observations

The MACE tests[113] examined debris coolability under early cavity flooding conditions (e.g., minutes after MCCI initiation). All tests were one-dimensional (1-D) except the MACE Scoping Test (M0), which utilized a three-dimensional (3-D) rectilinear cavity. The OECD/MCCI (or 'CCI') tests provided additional data on two-dimensional (2-D) MCCI behavior as well as debris coolability. All tests in this series were flooded 'late' (e.g., hours after MCCI was initiated), with the exception of CCI-6 that featured early cavity flooding.

MACE tests (which used both limestone-common sand [LCS] and siliceous [SIL] concrete types) all exhibited behavior in which the upper crust, formed by water cooling, would 'anchor' to the test section sidewalls. The 'anchored' crust would eventually separate from the melt due to: i) reduced gas sparging as the test progressed, causing the voided melt height to decrease, and ii) concrete densification upon melting (i.e., 'slumping'). In all tests, this led to suspended crust material anchored to test section sidewalls that eventually separated from the underlying melt leaving an intervening gap (illustrations to follow).

Regarding the degree of debris slumping during ablation, the volume reduction due to concrete densification upon melting is given by the equation:

$$V_f / V_0 = (1 - \chi_{gas})(\rho_{con} / \rho_{slag}) \tag{D-1}$$

In this equation, χ_{gas} is the mass fraction of decomposition gases in concrete (H_2O and CO_2), ρ_{con} is the original concrete density, and ρ_{slag} is the density of the slag produced by melting (i.e., the gases volatilize, and $Ca(OH)_2$, $CaCO_3$, and $MgCa(CO_2)_3$ are decomposed into the simple oxides, CaO and MgO).

In terms of applying this equation, note that for CORCON default 'Basalt' concrete, χ_{gas} = 0.0708; and based on CORQUENCH thermo-physical property subroutines, ρ_{con}=2431 kg/m^3 and ρ_{slag}= 2542 kg/m^3. With this information, Equation (D-1) yields a volume contraction fraction of V_f / V_0 =0.889 for Basalt concrete. This result implies that for every 10 cm of concrete erosion, 1.1 cm of surface elevation reduction will occur during 1-D ablation cases. Note that more slumping will occur under 2-D cavity erosion cases, but the extent of elevation reduction is more complicated to predict as it depends on the extent of lateral versus axial ablation.

D.2.1. MACE Scoping Test

This experiment,[113] which utilized LCS concrete, was carried out in a rectilinear concrete test section that was originally 30 cm x 30 cm in cross section. The corium mass was 130 kg. This was a high-power density experiment, with input power ranging from 700 to 1400 W/kg UO_2. The post-test debris distribution, melt-water heat flux, and input power (expressed as a heat flux) are shown in Figure D-3.

In this 3-D experiment, the top crust anchored to the sidewalls and remained mechanically stable for the rest of the test. Due to the high power density, a relatively thin, conduction-limited crust formed. Periods of high melt void fraction occurred in which the melt re-contacted the crust, leading to melt eruptions and particle bed formation. This early experiment illustrated the propensity for crust material to anchor to stable physical structures (in this case, the test section sidewalls) and to separate from the underlying melt due to concrete densification upon melting as well as melt eruptions that reduced the remaining melt volume.

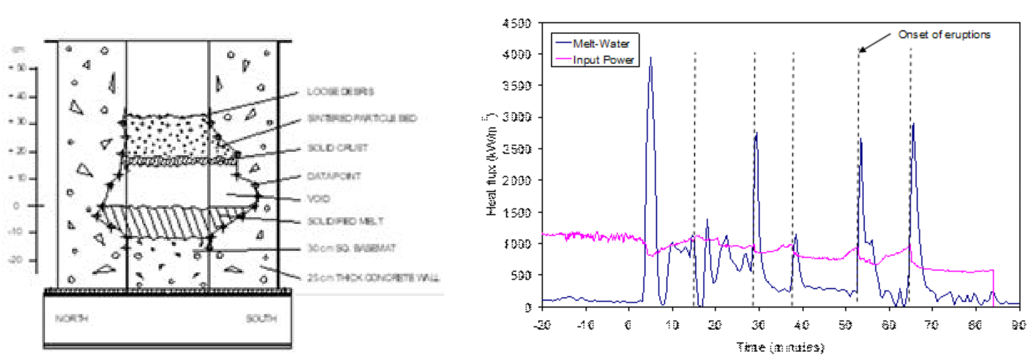

Figure D-3. MACE M0 post-test debris distribution (left) and power levels (right) [113]

The anchored crust formed in this experiment was viewed as ‘non-prototypic’ by experts at the time. Thus, in subsequent tests, the test section design was changed to use refractory MgO sidewalls, and the basemat cross sectional area was increased in an effort to achieve a ‘floating crust’ boundary condition.

D.2.2. MACE Test M3b

This test, which was also conducted with LCS concrete, used a much larger 120 cm x 120 cm square test section. As noted above, this was a 1-D experiment that utilized refractory MgO sidewalls. The initial corium mass was 2000 kg; the test was carried out at a power density typical of a PWR ~2 hours after scram.

Supporting crust mechanical strength analyses [115,116] indicated that a test section size of ~2 meters or greater would be needed to achieve a ‘floating crust’ boundary condition. However, practical considerations (i.e., costs associated with the need to increase the power supply used for direct electrical heating of the melt) limited the maximum test section size to 1.2 m x 1.2 m.

A schematic of the M3b post-test debris is shown in Figure D-4, while photographs of the debris upper surface at different stages during disassembly are shown in Figure D-5. This test showed similar phenomenological behavior as M0 (i.e., formation of an anchored crust, along with periodic melt eruptions). However, there was also evidence of crust mechanical failure (Figure D-5), indicating that an increase in lateral scale acts to mechanically destabilize anchored crust material formed during core-concrete interaction.

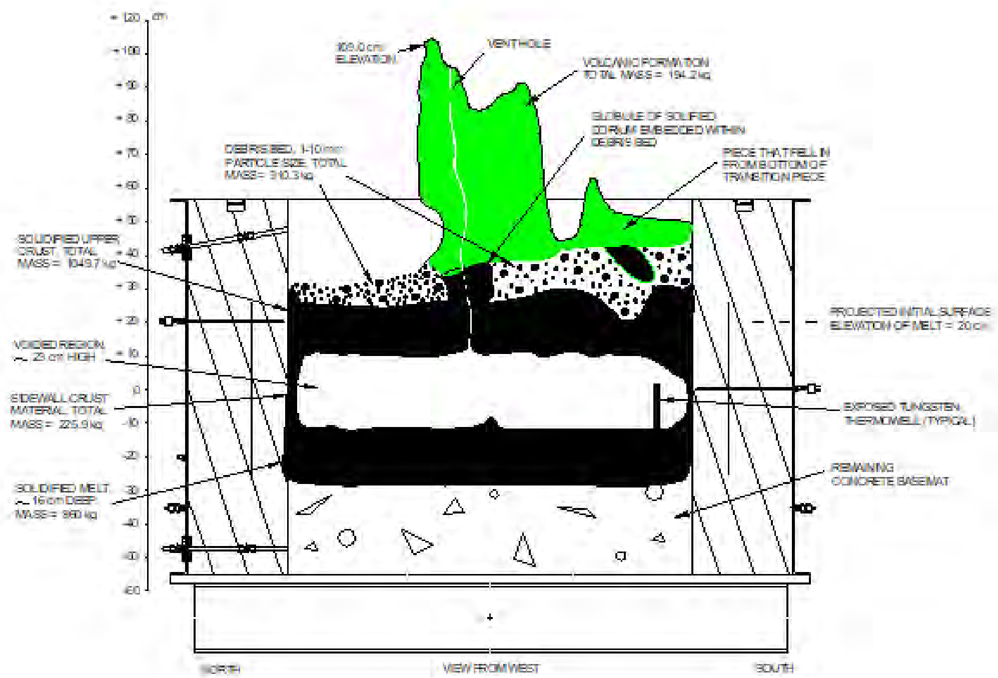

Figure D-4. Schematic of MACE M3b post-test debris configuration [113]

Figure D-5. Photographs of M3b post-test debris [113]

D.2.3. OECD/MCCI Test CCI-2

The CCI-2 experiment also used LCS concrete. However, in contrast to the latter MACE tests, a 2-D test section geometry was adopted; the initial concrete basemat size was 50 x 50 cm. After test initiation, the MCCI was allowed to progress for 6 hours under dry cavity conditions, followed by cavity flooding to gather data on debris coolability after an extended period of dry cavity ablation. A schematic and photographs of the post-test debris distribution are shown in Figure D-6.

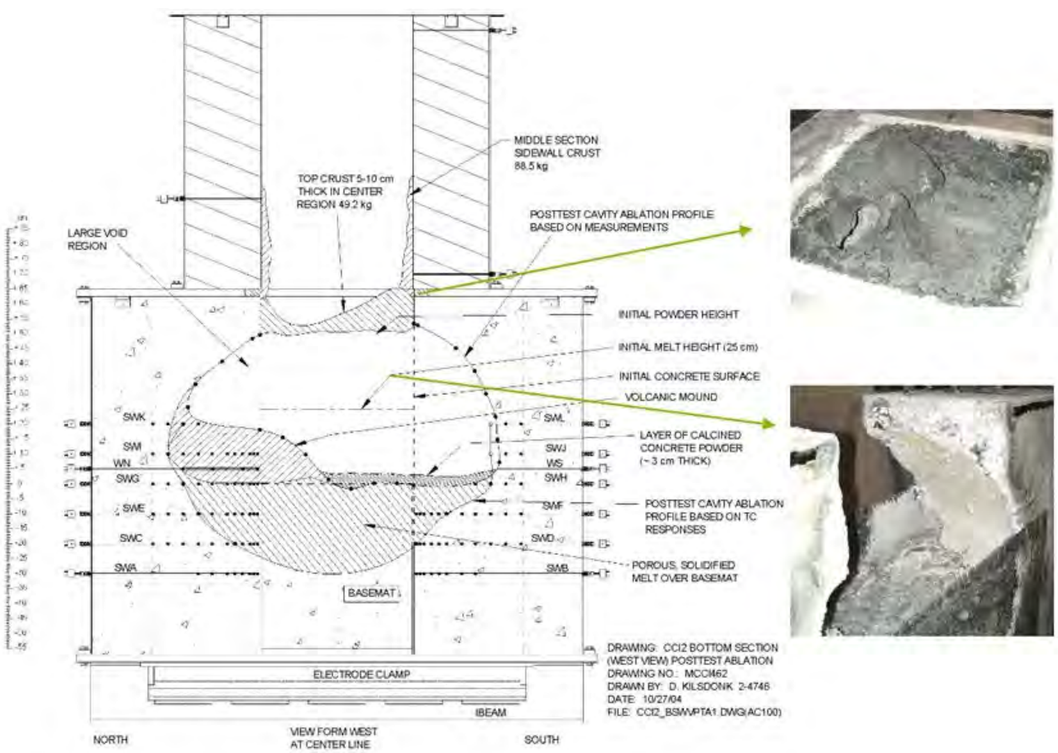

Figure D-6. Schematic and photographs of the CCI-2 post-test debris distribution [117]

Crust anchoring did not occur in this test (since it was operated under predominately dry conditions), but a large accumulation of solidified core debris formed in the upper portion of the test section. This was due to deposition from a highly swelled melt pool. This test also experienced significant sidewall ablation above the upper surface of the solidified debris. This was likely due to radiation heat transfer from the upper surface of the melt to the concrete sidewalls during the test. A relatively thick (~3 cm) layer of calcined concrete was also found on top of the solidified debris. This supports the idea that the overlying calcined concrete layer was formed by radiation heat transfer from the upper surface of the melt during the test, and not by ablation via direct contact with the melt. In particular, if the calcined concrete had been formed by direct contact with the melt, it is very likely that it would have been incorporated into the melt itself, as opposed to being deposited on top of it.

D.3. Crust Anchoring in Relation to Plant Conditions

Crust anchoring in the MACE tests raised concerns that if this phenomenon occurred at plant scale (Figure D-7), it may limit or prevent debris cooling by forming an insulating gap between the suspended crust and remaining melt below. On this basis, a series of crust mechanical strength tests was carried out on slabs of corium sectioned from SSWICS water ingression cooling test specimens in order to characterize the tensile strength of core debris quenched by water.[118] A photograph of one of the slab specimens is shown in Figure D-8, along with a 3-D rendering of the loading machine that was developed in order to carry out the loading experiments. In addition to these room temperature tests, data on high temperature crust mechanical strength were also obtained in-situ during core-concrete interaction tests using an insertable lance equipped with a load cell.[118]

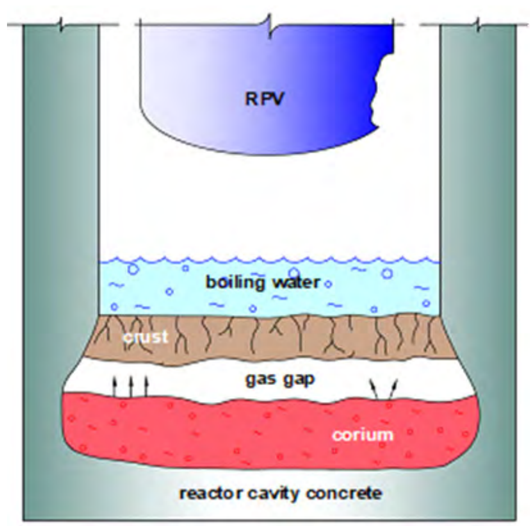

Figure D-7. Illustration of possible crust anchoring scenario during an MCCI under plant accident conditions [118]

Figure D-8. Corium slab (30 cm outside diameter, 5 cm thick) sectioned from SSWICS post-test corium ingot (left); loading device for measurement of crust tensile strength (right)

The results of these mechanical strength tests indicate that the crust material formed during quench by overlying water is quite weak (see Figure D-9). For instance, the measured section strength is considerably lower (i.e., by ~ two orders of magnitude) than that of sintered UO_2, ~150 MPa[119] and comparable to that of conventional concrete, ~2-5 MPa.[120]

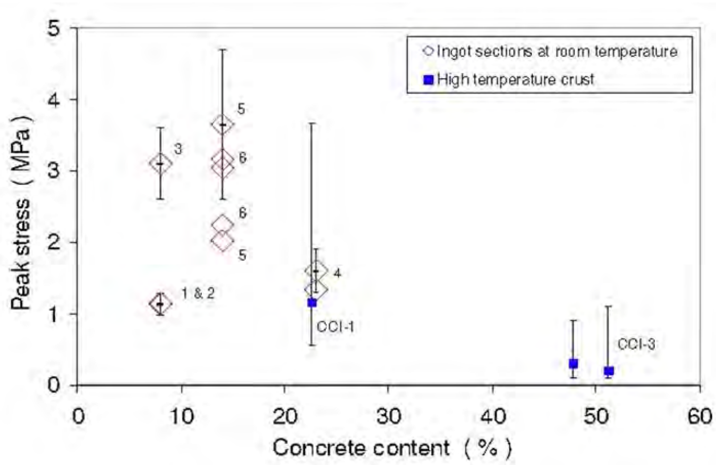

Figure D-9. Maximum centerline stress before fracture for ingot sections at room temperature and for high temperature crusts loaded in-situ during two core-concrete interaction tests [118]

This information was used as a basis for a simple crust mechanical loading calculation to examine the potential for crust anchoring to occur at plant scale.[118] An idealized circular reactor cavity is assumed with an inner diameter of 6 m (typical of many plants in the region below the reactor). Figure D-10 includes curves representing the peak centerline stress within a 6 m diameter, self-supported crust. The lower curve corresponds to a dry crust and the upper curve to a crust covered by a 50 cm deep water pool. The crust is presumed to be anchored at the perimeter and subjected to a distributed load equal to the weight of the crust itself (and water in the second case). Crust density was assumed to be 7000 kg/m^3. The peak stress can be compared with the strength data for an indication of the thickness required for a mechanically stable, self-supporting crust. The plot shows that the required thickness is 20-30 cm under dry cavity conditions, while a crust with an overlying water layer must be >30 cm thick to be mechanically stable.

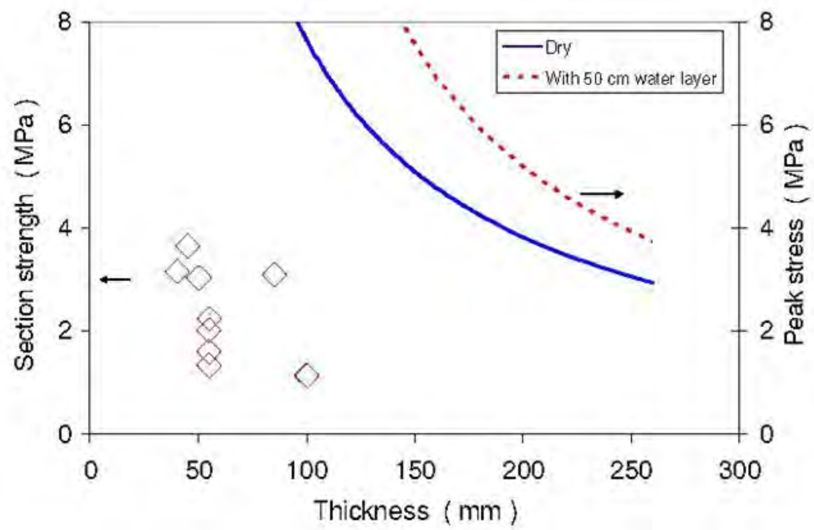

Figure D-10. Comparison between measured section strength and calculated peak stress in a 6 m diameter, self-supported crust[118]

Overall, these results indicate that sustained crust anchoring within the reactor cavity is unlikely and that 'crust breach' would likely occur, leading to renewed pathway(s) for water to re-contact and ingress into the core debris. The data (and MACE/CCI test results) further indicate that crust anchoring during MCCI is a possibility in smaller reactor cavity locations with a lateral span of ~ 2 m or less. Test M3b, at a lateral scale of 1.2 m, indicated that crust breach can in this lateral span range (see Figure D-5).

D.4. Main Insights from MACE/CCI Test Results Related to 1F1 Debris Distribution

In MACE tests featuring early cavity flooding, crust anchoring led to stable bridge crusts in the relatively small scale (compared to plant conditions) test sections used in these experiments. Crust anchoring was exacerbated by the fact that tests were flooded early when pool swell due to gas sparging was the highest. However, in MACE M3b, there was evidence of crust mechanical failure at some point in the experiment; see Figure D-5. This test featured a relatively large (at experiment scale) lateral span of 1.2 m. The occurrence of a breach event at this scale, along with the crust strength analyses carried out at the University of Wisconsin [115,116], indicates that the threshold lateral span for onset of stable bridge crust formation is on the order of 1-2 m. This size range is further noted to be similar to the annular gap width in the drywell of a Mark I reactor, factoring in equipment/piping/jet deflectors within the drywell that can act to mechanically support suspended crust material.

These observations seem to be consistent with 1F1 findings of suspended crust material and occurrence of crust shelves attached to structure in the doorway opening and in the drywell annulus.

Another question raised by NRA (see Section D.1) is, ‘why did the debris not spread?’ This question is based on the high surface elevations of solidified debris in the doorway/annulus region of 1F1 of ~ 1 m. At this debris depth, the majority of the core debris would have been retained in the pedestal region if the debris morphologies were fairly dense through the axial extent of the material.* One finding from the CCI tests [113] is that large melt void fractions (>50%) periodically occurred under dry cavity conditions, leading to deposition of crust material at high elevations in the test section (e.g., see Figure D-6). Aside from the CCI tests, note that periods of large melt void fractions also occurred during the 1-D dry cavity core-concrete interaction Tests L1 and L5 that were carried out as part of the ACE/MCCI program [121]. Periods of large melt void fractions coupled with crust anchoring/bridging might explain the elevated debris heights and bridge crust ledges observed in 1F1, as opposed to a lack of spreading. Researchers attributed the large melt void fractions observed during dry cavity MCCI tests to melt ‘foaming’, and models for this process have been developed and published in the literature.[122]

The dry cavity tests also showed evidence of sidewall ablation above the collapsed melt height, leaving calcined powder buildup on top of the crust material (see [117] and Figure D-6). The vertical extent of this ablation was limited by a suspended bridge crust attached to structure above the melt. The bridge crust apparently acted as an insulator preventing sidewall ablation above the bridge crust. This finding is consistent with observations of pedestal wall ablation below crust material attached to the pedestal walls in 1F1, but a lack of wall ablation above the bridge crust itself.

The buildup of a calcined concrete layer over the solidified debris in CCI-2 is also consistent with the occurrence of white powder layers covering material in various locations in 1F1.[Appendix C.2.3.1] The fact that the powder was not incorporated into the melt, but rather accumulated on top of the upper crust, indicates that this material was probably formed by radiation heat transfer to the sidewall during dry cavity ablation, as opposed to direct contact with the core debris.

D.5. CORQUENCH Scoping Calculations of Crust Anchoring Behavior in 1F1

A simple crust anchoring model was integrated into CORQUENCH in the mid 1990's and has been validated against MACE test results.[114] The motivation for the development of this model was a need for a tool to better understand crust anchoring behavior observed in MACE tests.[113] The model has been retained in the current code version, but has been rarely used except for code validation exercises against MACE test results.[114] A schematic of the floating crust boundary condition as modeled in CORQUENCH is shown in Figure D-11, while schematics of two different anchored crust modeling scenarios are shown in Figure D-12.

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* The melt pour mass predicted by severe accident codes, such as MELCOR, MAAP, and SAMPSON, for 1F1 is ~140 MT. For an assumed melt density of 7000 kg/m^3, this corresponds to a melt volume of ~ 20 m^3. The pedestal surface area in 1F1 is ~ 20 m^2. Thus, if all the core debris was retained in the pedestal after vessel failure, then the debris depth would be ~ 1 m (if fully dense or higher, assuming porosity exists in the solidified debris).

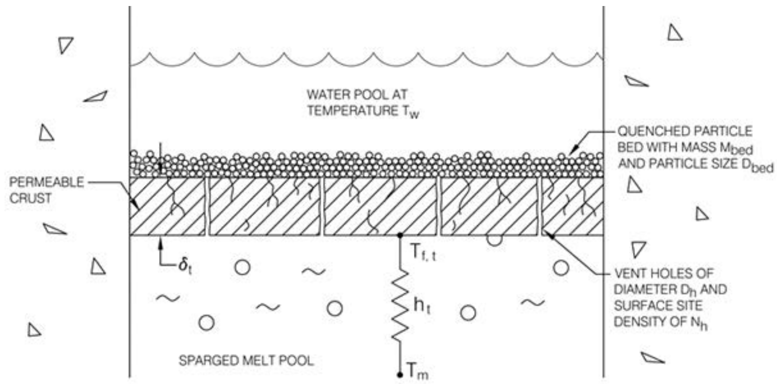

Figure D-11. Illustration of floating crust boundary condition as modeled in CORQUENCH [114]

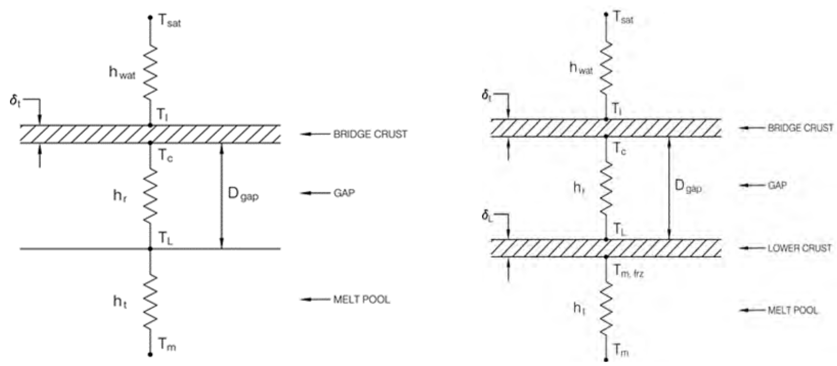

Figure D-12. Illustration of anchored crust boundary conditions modeled in CORQUENCH [114]

D.5.1. CORQUENCH Crust Anchoring Model Description

From a mechanical modeling viewpoint, the upper crust over the melt is treated as a flat plate with a known (user-supplied) tensile strength. The basis assumption is that the crust will continue to 'float' over melt as long as the mechanical strength of the crust is less than that which can support the combined weights of: i) the crust itself; ii) an overlying particle bed (if one exists); and iii) the weight of overlying water layer (as applicable). If the crust grows to a thickness where it can support those loads, then it is assumed to anchor at its current position to the sidewalls of the cavity.

After crust anchoring occurs, the ability of the melt to remain in contact with the lower surface of the crust depends on the fixed crust elevation versus the time-dependent voided melt height. If the melt swells to maintain or re-contact the anchored crust, then normal cooling mechanisms (Figure D-11) are calculated. If the voided height is calculated to be less than the bottom of the fixed crust, then a radiation heat transfer resistance is introduced between the melt and crust, as shown in Figure D-12.

As time progresses, the crust strength is continuously checked against the applied loads; and if the loads exceed its strength, the crust is assumed to 'fail' and is instantaneously placed back on top the melt as a 'floating' crust. Expressed as an equation, the crust anchoring criterion is as follows:

$$\underbrace{g(m_{bed} + \rho_{t,c}A_b\delta_{t,min} + m)}_{\text{applied load on crust}} \leq \underbrace{C_{geom}\sigma_{t,f}\delta^2_{t,min}}_{\text{crust mechanical strength}} \quad \text{(D-2)}$$

In this equation, m denotes mass; subscripts, '*bed'* and '*wat',* denote the particle bed and water layer regions, respectively; g is gravitational acceleration; $\rho_{t,c}$ is the crust density; $\delta_{t,min}$ is the minimum crust thickness for mechanical stability; $\sigma_{t,f}$ is the crust ultimate tensile strength; and C_{geom} is a constant depending upon the cavity geometry.

Once anchoring occurs, Equation (D-2) is also used to determine if the crust subsequently fails; then, the crust is placed back atop the melt pool. Mechanisms that can lead to failure after anchoring are: i) increased crust lateral dimension (by radial ablation); ii) reduced crust thickness due to crust remelting (via radiation heat transfer from underlying melt); and iii) changes in loading conditions on top of the crust (e.g., water addition).

The ability of a gap to form/exist at any time between the lower surface of the anchored crust and the upper surface of the melt pool is determined by simply tracking the voided melt height relative to the anchored crust bottom surface position; i.e.,

$$D_{gap} = \langle EL_{anchor} + (\delta_t t) - \delta_{t,anchor}) - EL_{m,v}(t), 0)\rangle \quad \text{(D-3)}$$

where EL_{anchor} is the elevation of the lower surface of the crust when it anchors, $\delta_{t,anchor}$ is the crust thickness when it anchors, $EL_{m,v}(t)$ is the current elevation of the upper surface of the melt pool (including a floating crust if it exists, see Figure D-12), and $\delta_t(t)$ is the current crust thickness.

D.5.2. Summary of Simulated Cases

Two cases were carried out in order to investigate the potential for crust anchoring to occur in the doorway region of 1F1 from a modeling perspective. This information is intended to supplement observations and insights from the experiments that were described in Section D.4.

One case examines conditions within the pedestal doorway, while the second examines conditions in the drywell annulus outside the doorway. The assumed geometries for the two cases are as follows:

- *Pedestal Doorway Opening*: 0.851 m wide door with 1.28 m pedestal wall thickness; ablation into pedestal walls adjacent to the doorway; adiabatic on other two sides.
- *Drywell Annulus*: 2.55 m radial slice between exterior of pedestal wall and drywell liner; width of slice assumed to be 2 m. Ablation into pedestal wall and PVC liner modeled; other two sides treated as adiabatic.

The depth of the spread melt after vessel failure in the pedestal/drywell regions was assumed to be uniform at 30 cm. For a 140 MT pour mass, this is equivalent to filling the pedestal sumps with corium and spreading material out the doorway to cover 112 degrees of the drywell area. For the purposes of this study, the initial melt composition and temperature are based on MELCOR simulations of the 1F1 accident sequence; see [123] for details. The concrete composition was assumed to be the same as the CORCON Basalt composition.

Based on the previously described crust strength measurements made as part of the OECD/MCCI program [118], a tensile strength of 3 MPa is assumed for the crust material (see Figure D-9). The melt void fraction due to sparging concrete decomposition gases was calculated using the Brockmann correlation [124]; melt foaming is not modeled as part of this work.

The calculation was run out to 14 days (336 hours) of real time simulation, which includes 11.25 days (270 hours) of dry cavity conditions, followed by cavity flooding to a uniform depth of 2 m (current condition). CORQUENCH does not include a containment response model. Thus, containment pressure needs to be provided as part of the code input. This is also an important parameter for this particular application since the pressure influences the melt volumetric gas sparging rate (via the ideal gas law) and, as a result, melt void fraction. The assumed containment pressure variation is shown in Figure D-13, as well as the data used as a basis for the pressure variation (i.e., TEPCO measurements supplemented by MELCOR simulation results [125]).

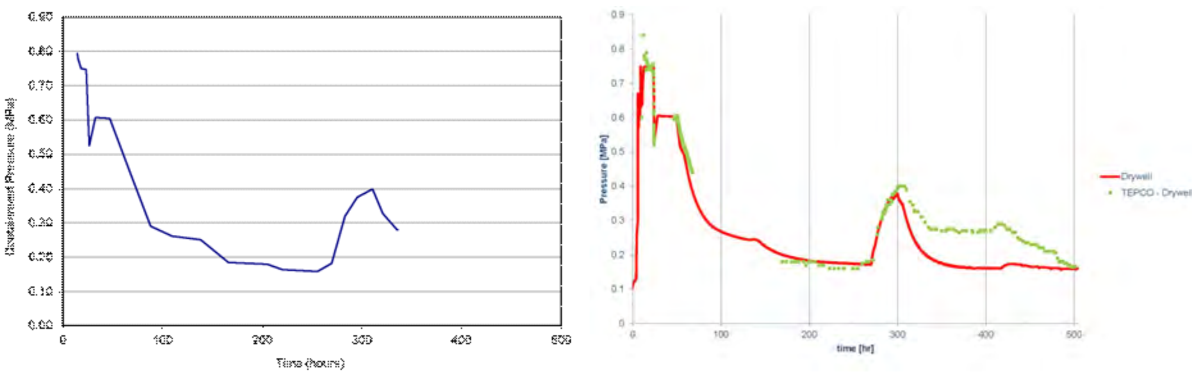

Figure D-13. Assumed 1F1 containment pressure variation (left); measured containment pressure compared to MELCOR simulation results (right) [125]

D.5.3. Doorway Opening Simulation Results

The principal results of the simulation in the current context are the evolutions of debris surface elevations and temperatures in the doorway opening; see Figures D-14 and D-15, respectively. As is evident from Figure D-14, three crust anchoring and two crust failure events are predicted to occur over the calculated time interval. Crusts are predicted to be anchored over a major fraction of the time. The first crust failure event, occurring at ~ 90 hours under dry cavity conditions, is predicted to occur as a result of lateral ablation in the doorway; the increased lateral span increases stress in the crust until it fails. Once back in contact with the melt, the crust gradually grows to a thickness of 4.2 cm, at which point it gains sufficient mechanical strength to anchor again. Melt separation occurs soon thereafter. This second anchored crust remains anchored with an intervening gap until the cavity is flooded at 270 hours. The large time intervals during the dry phase in which the crusts were anchored with intervening gaps would allow lateral sidewall ablation by radiation heat transfer to the exposed concrete surfaces. Although not explicitly modeled in CORQUENCH, the crust failure events would likely leave crust shelves attached to structures in the pedestal/regions. This is consistent with recent findings in 1F1 (see Appendix C.2.3.1).

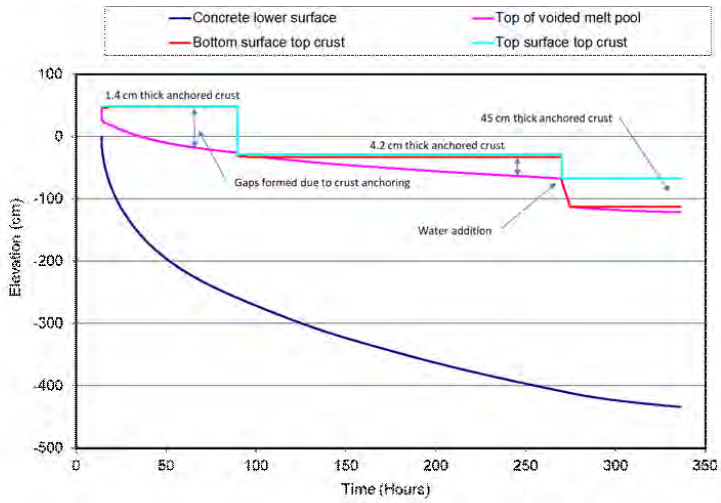

Figure D-14. Predicted surface elevation evolutions in the doorway opening for 1F1 over the first 14 days of the accident

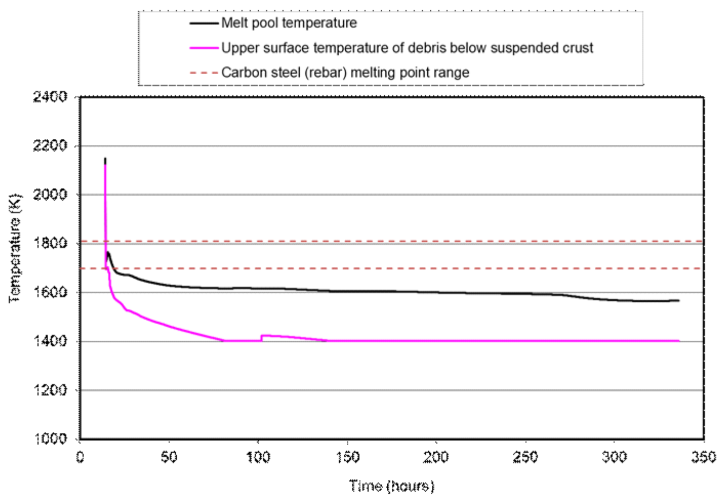

Figure D-15. Predicted melt and debris upper surface temperature evolutions in the doorway opening for 1F1 over the first 14 days of the accident

The introduction of water weight over the anchored crust causes it to fail. With water now present, the upper crust which is now floating on top of the core debris steadily grows by water ingression cooling to a thickness of 45 cm at ~ 275 hours. At this point, the crust re-anchors and remains in that state until the end of the simulation. An intervening gap forms soon after re-anchoring occurs, terminating cooling.

The current modeling assumption in CORQUENCH is that once the crust separates, water is not able to flood below the anchored crust to cool the underlying debris. This would lead to an uncoolable situation (by assumption) in which MCCI would continue (in theory) indefinitely. However, 1F1 observations (i.e., gas sample measurements in the containment atmosphere indicating a lack of H_2/CO production) suggest that the accident had been terminated; and MCCI was arrested in the August timeframe following event initiation. This observation supports the idea that, at reactor scale, water is able to flood below anchored crusts and continue debris cooling even if crust anchoring occurs.

The predicted debris temperature evolutions shown in Figure D-15 indicates that, due to concrete erosion and extensive slag ingress into the melt, the melt freezing temperature falls rapidly. The solidus temperature for the core debris rapidly approaches the solidus temperature of siliceous concrete, $T_{con,sol}$ (~1400 K).[126] For periods in which the anchored crust is separated from the underlying debris, the upper surface temperature of the debris (T_L in the schematics shown in Figure D-12) is predicted to hover near the concrete solidus temperature. Thus, radiation heat transfer from the upper surface of the debris to the adjacent concrete sidewalls occurs at a temperature of nominally $T_{con,sol}$. Figure D-15 also shows the melting point temperature range for carbon steel, which is the material used to fabricate concrete reinforcing bar. As is evident, the surface temperature of the debris remains substantially below the melting point for rebar. Thus, based on this analysis, the concrete rebar in the pedestal sidewalls of 1F1 would likely not have melted during the extended period of dry cavity erosion. This is a result of the large amount of concrete erosion predicted during this accident, which rapidly drives the core debris solidus (or freezing) temperature down below the melting point of steel.

D.5.4. Drywell Annulus Simulation Results

The principal results of the simulation for the drywell annulus case (i.e., the evolutions of debris surface elevations and temperatures) are shown in Figures D-16 and D-17. Overall, the trends and observations are similar to the doorway results. However, there are a few differences. Specifically, four anchoring events are predicted, as opposed to three for the doorway case. This is due to the fact that the crust is less stable in the broader lateral expanse of the drywell annulus in comparison to the doorway. Following the first crust failure event, there was an extended period (~ 2 days) in which the upper surface was initially crust free, and then was covered with a floating crust.

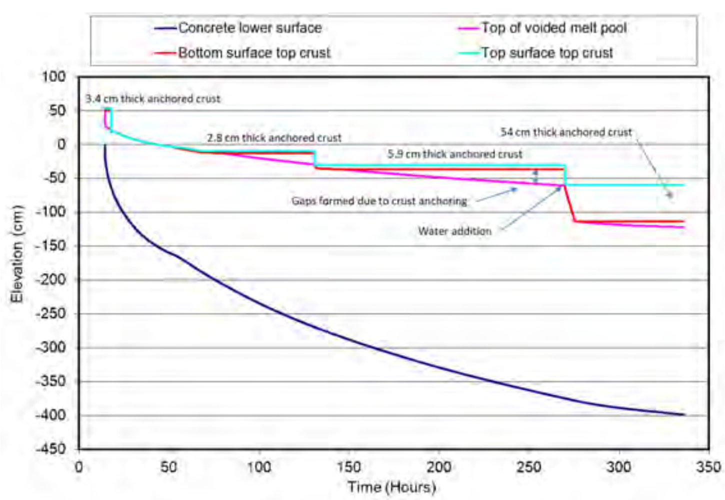

Figure D-16. Predicted surface elevation evolutions in the drywell annulus for 1f1 over the first 14 days of the accident.

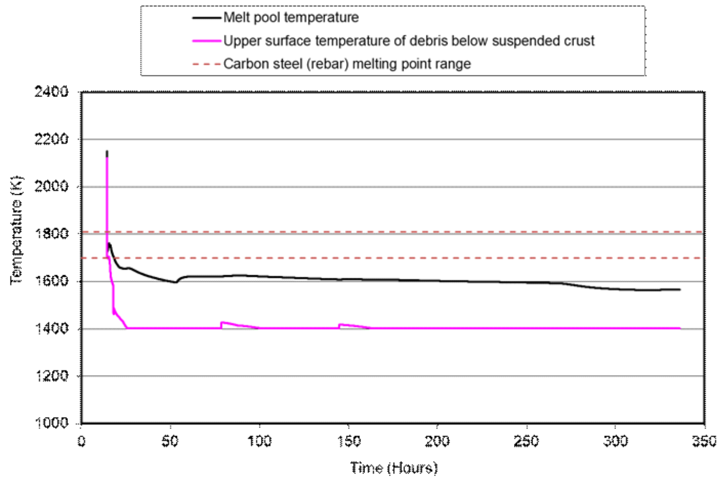

Figure D-17. Predicted melt and debris upper surface temperature evolutions in the drywell annulus for 1F1 over the first 14 days of the accident

Examination of Figure D-17 indicates temperature evolutions that are similar to the doorway case. However, it is noteworthy that during the extended period in which a floating crust is calculated to occur that the melt temperature rapidly falls below the carbon steel melting temperature. Thus, even though melt was in contact with rebar, the model predicts that rebar melting would not have occurred *from a thermal viewpoint*. There may be chemical interactions that could lead to rebar dissolution under these circumstances, but CORQUENCH currently does not model this type of interaction.

D.6. Summary of Insights Related to 1F1 Core Debris Distribution/Morphology

This work has focused on reviewing results from previous MCCI experiments, as well as conducting supporting analyses, in order to provide potential answers to some of the questions posed by NRAJ in view of recent findings by TEPCO regarding the ex-vessel core debris distribution in 1F1. The results of CORQUENCH calculations with the crust anchoring model activated indicate that several crust anchoring/failure events would likely occur over the first 11 days in which the cavity remained dry. These crust failure events would leave crust ledges in the range of 1-5 cm attached to sidewall materials. This is consistent with 1F1 video data.

Predictions of the occurrence of anchored crusts as well as the temperature evolutions in the gap between the anchored crust and melt were consistent with the idea that concrete around the rebar would have been ablated, but not the rebar itself. This is also consistent with recent 1F1 observations.

It is noteworthy that various reactor material experiments carried out at relatively small scale compared to the reactor case (i.e., up to 1.2 m in the test section lateral span), and discussed in detail in Section D.4, also indicate that crust anchoring to cavity sidewalls can occur. At the largest scale tested (i.e., 1.2 m lateral span in Test M3b), there was also evidence of crust breach/failure occurring.

D.7. Modeling Shortcomings Identified as Part of This Work

It is important to note that the CORQUENCH crust anchoring calculations presented herein are scoping in nature; this is first time these models have been used for a long-duration real plant accident scenario including extended dry and wet phases. There were a few instances (in time) for both cases where the code was not able to meet specified convergence criteria.

This work also revealed model shortcomings, as follows:

- The crust strength calculation is based on a simple flat plate (i.e., MCCI surface area) model. For places like the annulus, a beam strength model would be more appropriate because the behavior is essentially 1-D in the radial direction.
- The model does not leave crust ledges when the crust fails, as observed in 1F1.
- The current crust anchoring model does not include decay heat in suspended crust material which would change thickness relative to that calculated herein. In particular, thinner crust thicknesses would be predicted with decay heat included, making the crust more prone to mechanical failure.

As noted in the discussion, the code also pessimistically assumes that once the crust anchors, water is not able to flood below the crust to continue cooling. In essence, ‘crust breach’ is not modeled. Much work was done in the MACE and OECD/MCCI programs addressing the issue of crust anchoring and whether or not this type of behavior would be applicable to plant sequences. Specifically, there were concerns on whether anchored crust(s) would inhibit debris coolability. The results (both analytical and experimental) indicated that for tight cavity regions (a few meters) this may occur. However, it was argued that even if the crusts did anchor, they would not be completely stable. In particular, ‘crust breach’ would occur, allowing water to flood below the anchored crust and thereby maintain debris cooling. The results from 1F1 seem to support this vision of crust anchoring and breach behavior, thereby allowing the debris to cool. This is a very beneficial confirmatory observation from the viewpoint of reactor safety.

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