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U.S. Efforts in Support of Examinations at Fukushima Daiichi - November 2021 Meeting Notes with Updated Information Requests

Nuclear Engineering Division

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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 2022 (FY2022) 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 forensics meetings benefit operating, new, and advanced reactors. Significant safety insights are being 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. As discussed in this document, revised operator guidance was successfully used to improve operator response during a loss of off-site power event at the Duane Arnold Energy Center plant. Reduced uncertainties in modeling the events at Daiichi improve the realism of reactor safety evaluations that inform future D&D activities.

U.S. evaluations of information from Fukushima and input regarding future examinations are of interest to several organizations within Japan. Meeting presentations by Japan describe how comments and recommendations documented in prior U.S. forensics effort reports, including consensus information requests developed by forensics effort participants, are considered in future Fukushima D&D activities. As discussed in this report, TEPCO Holdings considered these information requests in their D&D planning activities. An updated list of consensus information requests is included in this FY2022 report.

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CONTENTS

FIGURES

TABLES

ACRONYMS AND ABBREVIATIONS

AC	Alternating Current or Air Conditioning
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 ANLArgonne National Laboratory
ANN	Artificial Neural Network
APD	Alpha Particle Detector
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
BSAF2	Phase 2 of BSAF
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CANDU	CANadian Deuterium Uranium
CBT	Computer Based Training
CCI	Core Concrete Interactions
CDE	Calculation System for Decontamination Effect
CDF	Core Damage Frequency
CLADS	Collaborative Laboratories for Advanced Decommissioning Science
CR	Control Rod
CRD	CR Drive
CS	Core Spray
CSARP	Cooperative Severe Accident Research Program
CSPE	ChloroSulfonated PolyDEthylene
CZT	Cd-Zn-Te
DAEC	Duane Arnold Energy Center
Daiichi	Fukushima Daiichi Nuclear Power Station
DBA	Design Basis Accident
DC	Direct Current or District of Columbia
D&D	Decontamination and Decommissioning
DFP	Diesel-driven Fire Pump
DOE	Department Of Energy
DOE-NE	Department of Energy Office of Nuclear Energy
dP	Differential Pressure
DPP	Decontamination Pilot Project

DW or D/W	Drywell
ELAP	Extended Loss of AC Power
EMUG	European MELCOR User Group
EOC	End of Cycle
EOP	Emergency Operating Procedure
EPA	Environmental Protection Agency
EPR	Ethylene-Propylene Rubber
EPC	Emergency Procedures Committee
EPG	Emergency Planning Guideline
EPRI	Electric Power Research Institute
ESTER	Experiments on Source Term for delayed Releases
EXOB	Expanded Operating Band
FACE	Fukushima Daiichi NPS Accident Information Collection and Evaluation
FDW	FeeDWater
FERMI-2	Fermi Unit 2
FE-SEM	Field Emission Scanning Electron Microscopy
FLEX	Diverse and Flexible Mitigation Capability (for accident mitigation)
FSAR	Final Safety Analysis Report
FY	Fiscal Year
GIS	Geographic Information System
GNSS	Global Navigation Satellite System
GOTHIC	Generation Of Thermal Hydraulic Information for Containments
GPS	Global Positioning System
GS	Turbine wheel with an 24 inch diameter and stainless steel surfaces in contact with steam
HALEU	High Assay Low Enriched Uranium
HBU	High Burnup
HEPA	High Efficiency Particulate Air
HPCI	High Pressure Coolant Injection
HYMERES	Hydrogen Mitigation Experiments for REactor Safety
HVAC	Heating, Ventilations, and Air Conditioning
IAE	Institute of Applied Energy
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
ICP	Inductively Coupled Plasma
INL	Idaho National Laboratory
INPO	Institute of Nuclear Power Operations
IP	Internet Protocol
IRID	International Research Institute for Nuclear Decommissioning

iRIS	Integrated Radiation Imaging System
IRIS	Industry Reporting and Information System
IRM	Intermediate Range Monitor
IRSN	Institut de Radioprotection et de Sûreté Nucléaire
JAEA	Japan Atomic Energy Agency
KIT	Karlsruhe Institute of Technology
LIBS	Laser Induced Breakdown Spectroscopy
LIDAR	LIght Detection And Ranging
LHV	Lower Heating Value
LPCI	Low Pressure Coolant Injection
LTM	Long-Term Management
LOOP	Loss of Offsite Power
LWR	Light Water Reactor
LWRS	Light Water Reactor Sustainability
MAAP	Modular Accident Analysis Program
MCCI	Molten Core Concrete Interactions
MELCOR	Methods for Estimation of Leakages and Consequences of Releases
METI	Ministry of Economy, Trade and Industry
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MUSA	Management of Uncertainties in Severe Accidents
NA	No information available or Not Applicable
NANTeL	National Academy for Nuclear Training e-Learning
NDF	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
NEA	Nuclear Energy Agency
NEET	Nuclear Energy Enabling Technologies
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
NTTF	Near Term Task Force
OECD	Organization for Economic Cooperation and Development
O&M	Operations and Maintenance
ORNL	Oak Ridge National Laboratory
PCV	Primary Containment Vessel
PLR	Primary Loop Recirculation
PM	Plant Maintenance

PPE	Personnel Protection Equipment
PreADES	Preparatory Studies for Fuel Debris Analysis
PRA	Probabilistic Risk Assessment
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
RESET	Restoration Support System for Environment
RMS	Remote Monitoring System
RN	RadioNuclide
ROSAU	Reduction Of Severe Accident Uncertainties
ROV	Remotely Operated Vehicle
RPV	Reactor Pressure Vessel
RTP	Rated Thermal Power
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)
SAFER	Strategic Alliance for FLEX Emergency Response
SBO	Station BlackOut
SC or S/C	Suppression Chamber
SFP	Spent Fuel Pool
SFPL	SFP Level
SGTS	Standby Gas Treatment System
SLAM	Simultaneous Localization Mapping
SLC	Standby Liquid Cooling
SME	Subject Matter Expert
SMR	Small Modular Reactor
SNL	Sandia National Laboratories
SOUL	Smart Open Universe Learning
SRM	Source Range Monitor
SRV	Safety Relief Valve
TAMU	Texas A&M University
T/H	Thermal Hydraulics
TBD	To Be Determined

TCV	Turbine Control Valve
TDAFW	Turbine Driven Auxiliary FeedWater
TDR	Time Domain Reflectrometry
TEM	Transmission Electron Microscopy
TEPCO Holdings	Tokyo Electric Power Company Holdings, Inc.
TIP	Traversing In-core Probe
TLO	Terminal Learning Objective
TMI-2	Three Mile Island Unit 2
T/B	Turbine Building
TTEXOB	Terry^{TM} Turbine Expanded Operating Band
TVA	Tennessee Valley Authority
U.S.	United States
UAV	Unmanned Aerial Vehicle
VIP	Vessel and Internals Program
VR	Virtual Reality
VVER	Voda-Vodyanoi Energetichesky Reaktor
WANO	World Association of Nuclear Operators
WW or W/W	Wetwell
X-#	Designation for PCV penetration number
XRD	X-Ray Diffraction
XRF	X-Ray Florescence
ZS	Turbine wheel with an 18 inch diameter and stainless steel surfaces in contact with steam
1F1	Fukushima Daiichi Unit 1
1F2	Fukushima Daiichi Unit 2
1F3	Fukushima Daiichi Unit 3
1F4	Fukushima Daiichi Unit 4
2D or 2-D	Two-Dimensional
3D or 3-D	Three-Dimensional
Φ	Equivalence ratio

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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). 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 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], Daiichi 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, vessel failure, containment failure, and core/concrete interactions after reactor pressure vessel (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 U.S. 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 TEPCO Holdings D&D plans for Daiichi.
- ***Objective 2:*** Evaluate obtained information for several reasons:
  - Gain a better understanding related to events that occurred in each unit at Daiichi;
  - Gain insights to reduce uncertainties in predicting phenomena and equipment performance during severe accidents;
  - Provide insights beneficial to 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. Such evaluations are useful because of U.S. experience in 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 Daiichi 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 U.S. 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 11] As documented in these publications, the U.S. has already gained significant safety benefit from the information obtained from the affected units at Daiichi to reduce uncertainties in BWR severe accident progression and implement safety enhancements for BWRs, pressurized water reactors (PWRs), and future nuclear power plant designs. As uncertainties in modeling the events at Daiichi are reduced, it not only improves guidance for accident mitigation but it informs future D&D activities by improving the capability to characterize potential hazards to workers involved with cleanup activities.

1.2. Approach

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

1.2.1. 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.[8] In this initial and subsequent annual reports, 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 of the information needs identified by the expert panel are related to Daiichi Units 1 through 4 (1F1, 1F2, 1F3, and 1F4).* Although details varied, U.S. experts generally identified needs required to answer fundamental questions related to how the accident progressed in each unit, to understand equipment and component survivability, and to benchmark severe accident progression and dose assessment

* 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.

codes. These needs are organized in tables per location [e.g., the reactor building (RB), the primary containment vessel (PCV), and the RPV]. These tables also identify applicable units for each need and other relevant factors (e.g., how information should be obtained, why it is needed, its expected use or benefits, and when it should be obtained). Each year, additional information becomes available. As discussed in Section 1.2.2, Objective 2 activities evaluate new information and make revisions to examination requests as appropriate.

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.[12] As indicated by the gray box, severe accident and plant operations experts from U.S. industry, universities, and national laboratories evaluate plant examination information obtained from Daiichi. Since its origin, the forensics effort has striven to include a broad spectrum of U.S. stakeholder input. Objective 2 activities are also informed by experts from the U.S. NRC, U.S. DOE, and Japanese organizations that participate in expert panel meetings.

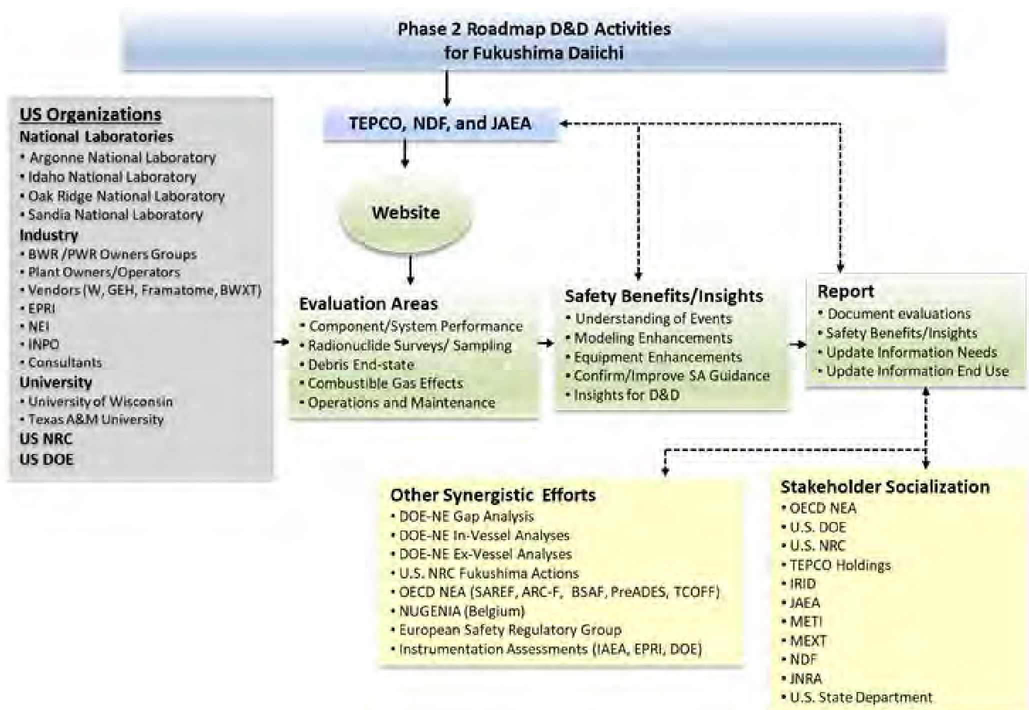

Figure 1-1. Objective 2 activities (organizations and programs defined in list of acronyms and abbreviations)

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

- Component/System Performance
- Radiological Sampling and Surveys
- Core Debris End-state

- Combustible Gas Effects[†]
- Operations and Maintenance[‡]

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

The primary source of information used in U.S. Forensics Effort evaluations is information provided by TEPCO Holdings[13] 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.[14 through 19] Websites created by organizations within Japan, such as the Institute of Applied Energy (IAE)[20], TEPCO Holdings[21], and JAEA [22], are important references for this effort. In addition, as discussed in Appendix B of Reference [4], 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 FY2022 report was completed, there were 796 references archived on this website (a significant increase over the 230 references archived at the time that Reference [4] was published).

Forensics evaluations have led to several types of safety benefits and insights. As shown in Figure 1-1, U.S. experts prepare a report documenting insights from these evaluations and updates related to the U.S. information requests for additional examinations. For the first five years of this effort, these reports were substantive in order to capture results associated with information coming from the affected units. For each area, prioritized questions of interest were identified; available information was reviewed; and insights gained from evaluating this information were documented. Where appropriate, information requests were revised based on new examination and evaluation results. Additional details, such as the benefits, use, and suggested methods for obtaining higher priority, near-term examination activities were also updated. In FY2020, it was decided that the program would gain more benefit from a more concise report that emphasizes new information and insights. This FY2022 report follows the FY2020 approach. However, a key aspect of the U.S. efforts, the updated list of information requests, is still included in this more concise report.

1.2.3. 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.

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

[‡] This fifth area was added in FY2018.

1.3. Report Objectives and Organization

As noted above, this FY 2022 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 FY2022 meeting. Section 3 highlights key findings and recommendations from these meetings. References for this report are listed in Section 4. Appendices to this document provide more detailed information. Specifically, Appendix A provides lists of attendees and agendas from the November 2021 U.S. Forensics Meeting. Appendix B provides updated tables with detailed information requests developed by U.S. experts and additional details for high priority, nearer term examination activities. Appendix C includes presentations from participants wishing to include them in this publication. Appendices D and E provide supporting information for findings and recommendations in Topics Areas 4 and 5, respectively.

2. FY2022 EXPERT PANEL MEETING HIGHLIGHTS

This section highlights presentations and discussions that occurred during the FY2022 Expert Panel meeting for the U.S. DOE sponsored Forensics Effort. The FY2022 meeting was conducted as a hybrid meeting (due to COVID19 restrictions). Appendix A includes an agenda and a list of meeting participants. The meeting consisted of three sessions. Representatives from Japanese organizations highlighted recent activities during Session 1; representatives from U.S. organizations provided presentations during Sessions 2 and 3.

Information discussed during the meeting is presented in four sections: new information obtained from Japanese organizations (Section 2.1), introductory U.S. presentations (Section 2.2), U.S. presentations related to systems analysis codes development and application activities (Section 2.3), and U.S. topic area evaluations (Section 2.4). As emphasized in Section 1, a key objective of the DOE Forensics Effort is to develop and update the list of consensus information requests. Section 2.5 highlights changes to these information requests developed during this meeting (updated information requests are found in Appendix B). Appendix C includes presentations from participants wishing to include them in this publication.

2.1. New Information from Japan

During Session 1, information related to Daiichi and plans for future activities were presented by four Japanese organizations: NDF, NRAJ, TEPCO, and JAEA. These detailed presentations provided significant new information and insights. In addition, several presentations (e.g., see Sections 2.1.3.4 and 2.1.4) demonstrate Japanese and U.S. organizations benefit from this collaboration.

2.1.1. NDF

In his presentation, Takatsune Ito provided an overview of the 2021 Strategic Plan for decommissioning of Fukushima, which was released in October 2021.[23] 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 International Research Institute for Nuclear Decommissioning (IRID). The 2021 plan focuses 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 the upcoming trial debris retrieval (to minimize COVID-19 impact)
- Issues related to selecting methods for further expansion of debris retrieval
- Strategies for discharging Advanced Liquid Processing System (ALPS)-treated water

The 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 recognizes and has taken 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 the local communities and governments (e.g., taking actions to increase their participation and to revitalize affected communities).

In discussions following this presentation (and in subsequent email correspondence [24]), several aspects about the revised strategic plan were clarified. For example, NDF indicated the most significant changes in the 2021 plan are related to improvements in 1F fuel debris analysis. NDF also indicated that they concurred with forensics activities as long as it does not significantly affect 1F decommissioning. In response to queries about the impact of forensics investigations on 1F D&D, NDF observed that they respect and agree with on-going TEPCO investigations (with NRAJ involvement) of 1F radiation levels because of their potential to optimize future decommissioning activities. In addition, the level of NRAJ involvement in the strategic plan was clarified. Namely, NRAJ remained cognizant of the strategic plan but did not find it appropriate for an independent regulator to endorse the strategic plan.

2.1.2. NRAJ

In his presentation (Appendix C.1.2), Masaya Yasui provided updates on several NRAJ activities highlighted in his presentation at the U. S. DOE November 2020 meeting (see Appendix C.2.1 of [2]) and a status report on a proposed new Japan-led Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) post-Fukushima project, Fukushima Daiichi NPS Accident Information Collection and Evaluation (FACE). Mr. Yasui's DOE Forensics November 2020 meeting presentation was included in the NRAJ annual report (issued March 2021)[25], and this November 2021 presentation will be included in the NRAJ annual report (expected to be issued in 2022). In their response to questions provided after the meeting, NRAJ indicated that they are planning to make an English translation of their 2022 report available.[26]

NRAJ investigations continue to provide new insights, which affect future D&D activities and have the potential to improve current understanding of the accident progression at each unit and global nuclear safety. Specific areas of investigation and preliminary findings include:

- Shield Plug Contamination - As indicated in the NRAJ November 2020 presentation, prior measurements indicated there was around 30 PBq of Cs 137 present under the upper 1F3 shield plug and more than 30 PBq possibly existing under the upper 1F2 upper shield plug. On-going investigations are focused on bore-hole measurements to gain insights about Cs contamination levels and movement through the 1F2 shield plugs. Using unique robots, bore-hole measurements at various depths and locations are being pursued. Preliminary results indicate higher resolution measurements can be obtained: lower depth measurements reduce floor surface and dust contamination effects, and multiple location measurements detect localized sources of contamination. Preliminary measurements indicate higher radiation levels (and associated deposits) are monitored at locations over the seams of the upper and middle shield plugs align (preliminary data suggest that increased depositions, due to higher flow rates, may be present at these locations). On-going evaluations are also considering what, if any, effects or gross movement and/or thermal deformation of the shield plug may have on cesium deposition. Additional discussions on this topic are found in Sections 2.1.3.3 and 2.4.2.
- 1F2 Reactor Well Radiation Field - February 2021 confirmatory measurements by TEPCO indicate radiation fields in the reactor well were much lower than values inferred from JAEA 1F2 accident progression calculation estimates for the fuel containing material still within the PCV (measurements indicate values less than 0.6 Sv/hr rather than 40 Sv/hr). On-going NRAJ efforts focus on resolving

this discrepancy. Preliminary investigations indicate that steam flow from the PCV may be the major factor affecting fission product transport and deposition. Investigations also suggest localized temperature increases associated with gas flow, rather than shield plug movement and/or deformation, may be an important factor. Additional information on this topic is found in Section 2.1.3.3.

- Standby Gas Treatment System (SGTS) Filter Train and Line Contamination - Evaluations of contamination levels in the 1F1 and 1F2 SGTS filter trains are hindered by the high radiation fields (over 5 Sv/hr), especially in 1F1 (estimated as over 5 Sv/hr). To facilitate upcoming SGTS line removals, detailed contamination measurements using gamma camera technologies are planned. Investigations suggest condensation may have been the major factor affecting cesium deposition with the SGTS filter trains. Some of the observed deposition, however, may have been due to events occurring in the years following the accident.
- Possible Sources of Organic Gas - To better understand phenomena, such as the orange flame color, observed in the high resolution images NRAJ obtained of the 1F3 explosion, NRAJ and TEPCO are exploring possible sources of organic combustible gases within the PCV. TEPCO and JAEA will be conducting tests to identify the type and amount of combustible organic gases released during heating of various types of cable sheath and insulation materials and painting materials present in the 1F3 PCV. The overall scope of the testing program and possible contribution by these additional sources of combustible gases (not considered in current severe accident analyses) was discussed further during several presentations (see Sections 2.1.3.2 and 2.4.4).

Finally, Mr. Yasui provided a brief update on the status of on-going Japan-led OECD/NEA projects, Analysis of Information from Reactor Buildings and Containment Vessels in Fukushima Daiichi Nuclear Power Station (ARC-F) and Preparatory Studies for Fuel Debris Analysis (PreADES), and the proposed new OECD/NEA FACE project. The primary objective of FACE is to inform participates about 1F examination and analysis results that provide insights about the accident progressions. It is anticipated that a single project coordinated by NRAJ, which will include accident analyses activities and results from a round robin hot cell examination of samples from an Argonne National Laboratory (ANL) Molten Core Concrete Interaction (MCCI) test, will prove more efficient than prior Japan-led OECD projects. Finally, Mr. Yasui emphasized the importance of considering feasibility and importance when information requests are proposed by researchers in OECD/NEA programs. In their response to questions provided after the meeting[26], Mr. Yasui indicated that NRAJ expects to have follow-on discussions about the information requests and supporting information developed by the U.S. DOE Forensics Effort (see Section 2.5 and Appendix B of this report).

2.1.3. TEPCO

Five presentations were given by representatives from TEPCO (see Appendix C.1.3): Shinya Mizokami discussed D&D knowledge management efforts; Kenji Owada reviewed visual information within 1F1, 1F2, and 1F3 related to MCCI, peak temperatures, and degradation of organic materials; Michal Cibula reviewed results from recent 1F2 reactor well investigations; Michal Cibula provided an update on planned 1F investigations; and Shinya Mizokami reviewed recent evaluation results of 1F3 Reactor Core Isolation Cooling (RCIC) operation.

2.1.3.1. Knowledge Management Mechanism

In their Fifth International Peer Review report,[27] the International Atomic Energy Agency (IAEA) commended TEPCO for their recently-initiated knowledge management effort to identify, document, and disseminate best practices in 1F D&D activities. In this presentation (see Appendix C.1.3.1), Dr. Mizokami elaborated on the processes used by this new TEPCO program to document and discuss best practices and lessons learned with staff during in-house training courses. To illustrate this process, a one-page example was provided that documented the limitations associated with the use of offsite mock-ups for 1F3 SFP removal.[**] Dr. Mizokami indicated TEPCO planned to include these lessons in training courses provided to workers performing decommissioning work at other plants in which no accident occurred. He observed, however, that many plants won't be restarted in Japan.

2.1.3.2. Insights from PCV Visual Information

To gain additional insights regarding the 1F explosions, visual information about the endstate of structures and components are used to identify sources of combustible gases within the PCVs. These images are used to assess the occurrence of MCCI, quantify peak containment temperatures, and identify other sources of flammable organic gases that may have contributed to these explosions. Because these other sources aren't currently considered in systems analysis codes, their contribution might impact accident progressions in existing plants, as well as proposed new advanced reactor designs. In this presentation (see Appendix C.1.3.2), Dr. Owada reviewed results from prior forensics investigations (e.g., degraded sheath material from control cables leaking from the 1F2 X-6 penetration, images of damaged cables and components within the PCV, damaged PCV surface paint, etc.). In some cases, such as the presence of previously molten organic material leaked from the 1F2 X-6 penetration and the presence of blueish fragments in relocated sediment on the 1F1 PCV floor, the images provide an indication of peak temperatures within the PCV (i.e., temperatures were sufficient to melt components). Conversely, Dr. Owada also provided images of undamaged components (e.g., a stainless steel cable tray in 1F2), suggesting MCCI may not have occurred in 1F2 because temperatures remained below the melting point of these components. Information provided by Dr. Owada is discussed further in Sections 2.4.3 and 2.4.4.

2.1.3.3. 1F2 Reactor Well Investigations

As a follow-on to topics discussed by NRAJ in Section 2.1.2, Dr Cibula's presentation (see Appendix C.1.3.3) provides results from TEPCO investigations to quantify contamination levels within the shield plugs and the reactor well. Results indicate peak 1F2 dose rates were measured at location 8 (slightly above the location above the PCV head bolts as shown in Slide 10 in Appendix C.1.3.3). Images obtained during this investigation show considerable surface corrosion. Other factors, such as the presence of curing sheet over the 1F2 shield plug, core spray injection during the accident, and aging over the 10 years since the accident, may affect observed contamination levels and obtained images. Information provided by Dr. Cibula is discussed further in Section 2.4.2.

** As indicated in NUREG/KM-0001[28], an important TMI-2 lesson was that the fabrication of defueling equipment on site was more effective because of the interactions of designers with onsite operators and engineers. Mockups (offsite and then onsite) were used extensively for project planning, procuring, designing, testing, drafting procedures, shipping, training, assembling, and operations. Mockups proved effective in improving work efficiency, minimizing radiation exposure, and eliminating contaminated waste from equipment that did not work as expected

2.1.3.4. Mid-and-Long-Term Plan for 1F Investigations

In this presentation (Appendix 2.1.3.4), Dr. Cibula outlined the mid-and-long-term plan for future 1F forensics investigations. Future investigations are planned for each unit based on implementation time (immediate: within approximately the next 6 months; short term: within the next 3 years; mid term: within the next 10 years; and unknown); location (the reactor building or R/B, the PCV, the RPV, and the turbine building [T/B]); D&D (equipment removal), stakeholder needs, and external commitments; safety importance (irrespective of D&D relevance); and D&D progress. Some investigation information is also obtained through D&D work not listed in this plan. Of special interest is that Dr. Cibula's presentations explicitly consider U.S. DOE Forensics information requests (with the identifiers) listed in Appendix B. Other sources, such as the NRAJ, also provide information requests that TEPCO plans to address. Information provided in this presentation is discussed further in Section 2.5.

2.1.3.5. Long-Term Cooling Insights

In this presentation (Appendix C.1.3.5), Dr. Mizokami described a lesson learned, the need to consider unexpected conditions (e.g., long-term cooling) when selecting and using instrumentation. In this example, he observed that the 1F1 operating procedure required verification that water injection rate never fell below 55 m^3/hr (which was appropriate for expected decay heat levels of 30 MWt). TEPCO had installed a flowmeter with operating ranges from 20 to 200 m^3/hr (if the flowrate fell below the lower threshold of 20 m^3/hr, the flowmeter reading was zero). On March 20, 2011 (when alternate source of water, such as a fire protection pump, were used for 1F1 water injection), it is estimated the decay heat ranged between 2 and 4 MW. For such low decay heats, the installed flowmeter was unable to provide accurate readings; operators reported values less than 2 m^3/hr (well below the lower threshold value). In the subsequent discussion, U.S. experts observed that inadequate consideration of instrumentation ranges have led to errors by operators in several accidents. BWR Owners Group (BWROG) representatives indicated special emphasis was placed on avoiding this problem in Diverse and Flexible Mitigation Capability (FLEX) equipment selection and revised guidance development.

2.1.4. JAEA

In his opening remarks, Tadahiro Washiya introduced the three JAEA presentations topics (see Appendix C.1.4): an update on fuel debris analysis activities by Shin-ichi Koyama; an update on 1F sample analysis by Hirotomo Ikeuchi; and an introduction to the capabilities of iRIS (integrated Radiation Imaging System by Yuki Sato.

Dr. Koyama first provided an update on the new JAEA Collaborative Laboratories for Advanced Decommissioning Science (CLADS) facilities and plans for conducting examinations at these facilities. JAEA is updating the draft report, "Analysis of Debris Samples of Tokyo Electric Power Company Holdings Fukushima Daiichi Nuclear Power Station,"[29] which outlines JAEA plans for future examinations of fuel containing debris. During the last year, several U.S. DOE Forensics panel members provided comments on the initial version of this document.[30] Dr. Koyama indicate that these comments are being considered in their update. U.S. participants expressed their appreciation that the suggestions provided to JAEA were being considered and encouraged JAEA to provide questions if some of their comments required clarification. This topic is discussed further in Section 2.4.3.

Dr. Ikeuchi reviewed results from analyses obtained from selected locations within 1F1, 1F2, and 1F3. Although there remains some uncertainty, examination results provide insights regarding the peak temperatures and durations of these high temperatures that samples experienced as well as the sample chemical and isotopic content. Information provided by Dr. Ikeuchi is discussed further in Section 2.4.3.

Dr. Koyama next described a collaborative effort, involving several Japanese organizations, to examine simulated fuel debris. Examinations focused on four characteristics: morphology, nuclide/element amounts, phases, and density (porosity). By comparing and discussing results from this activity, insights were gained in developing standardized techniques and methods for analyzing materials expected to be retrieved from Daiichi and characterizing the uncertainties associated with these measurements. It was observed that this activity complements an international round robin activity, which will be completed in the OECD FACE project. This information is also discussed further in Section 2.4.3

Finally, Dr. Sato described the capabilities of Integrated Radiation Imaging System (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 with a small backpack. In his presentation, Dr. Sato included videos illustrating how workers can easily detect hot spots and minimize their exposures. There was considerable enthusiasm about this effort by U.S. participants. Additional discussion on this topic may be found in Section 2.4.5.

2.2. Introductory Presentations

Session 2 started with four introductory presentations: a welcome by John Butler, who represented the Nuclear Energy Institute, the host for Sessions 2 and 3 of this meeting; an overview of the DOE-sponsored efforts to support forensics examinations by Damian Peko, the U.S. DOE program manager; 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; and a summary of findings and recommendations from the U.S. DOE FY21 report and Fukushima Lessons Learned documents released by international organizations by Joy Rempe, Rempe and Associates, LLC, and Technical Lead for the DOE-sponsored forensics efforts.

In his presentation (Appendix C.2.1), Mr. Peko reviewed the objectives of the DOE Forensic Effort, emphasizing that the focus is to obtain as much information as possible from the examination activities without adversely affecting Japan's efforts to expeditiously complete Fukushima reactor D&D and site cleanup. He also reviewed the approach used by this effort and how results from this collaboration continue to benefit participants from Japan and the U.S.

Dr. Esmaili then provided an overview of NRC-sponsored severe accident and source term research activities. As emphasized in his presentation (see Appendix C.2.2), these research activities provide important input to several agency regulatory activities. Dr. Esmaili reviewed the status of the NRC-sponsored systems analysis code, Methods for Estimation of Leakages and Consequences of Releases (MELCOR),[31] development and application activities (noting that a new version of the code is scheduled for release in December 2021). 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, ARC-F, PreAdES, etc.), 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.

In her presentation (see Appendix C.2.3), Dr. Rempe summarized findings and recommendations documented in the FY2021 Forensics effort report.[2] Due to continuing COVID19 restrictions, she observed the FY2022 meeting was conducted as a hybrid meeting, allowing in-person and virtual participation. As indicated in Appendix A.2, nearly 70 experts in reactor safety and/or plant operations participated in the FY2022 meeting (an increase of approximately 25% from FY2021 and more than twice the number of participants in FY2019). Dr. Rempe highlighted FY2021 recommendations and actions taken to implement several recommendations. For example, an expert panel completed a more detailed review of NRAJ activities related to combustible gas generation and ignition; in addition, a draft document was prepared evaluating new technologies being deployed for 1F D&D for routine plant maintenance activities. FY2021 reviews did not identify any new information requests. However, U.S. experts clarified text for several information requests and concluded several existing requests had been completed.

During the last year, several workshops and reports were completed focusing on lessons learned since the March 11, 2011 accidents at Daiichi. Dr. Rempe's presentation included several slides highlighting short-term and long-term lessons identified in these activities and initial U.S. thoughts about these lessons. Dr. Rempe concluded her presentation with an overview of the meeting agenda and a link where participants could access presentation material.

2.2.1. Summary

These introductory discussions led to one insight and several recommendations:

Insight: Available Fukushima-related information from Japan and discussions of this information at DOE Forensics Expert Panel meetings continue to benefit the U.S. operating fleet as well as new Light Water Reactor (LWR) and non-LWR design efforts.

During the last decade, 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. Efforts continue to improve our understanding of the accident progressions in each unit and the performance of structures, systems, and components during these accidents. This includes participation in the DOE-sponsored Forensics effort as well as international collaborative efforts. As new information becomes available, the U.S. continues to benchmark (and revise, as warranted) models in systems analysis codes and containment fission product transport codes. Likewise, severe accident knowledge management activities continue to ensure that information is archived and transferred to the global nuclear community.

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

2.3. Systems Analysis Code Development and Application Activities

Knowledge gained from Daiichi examinations is used to reduce uncertainties in systems analysis codes, such as the EPRI-sponsored Modular Accident Analysis Program (MAAP) code[32] and the NRC-sponsored MELCOR code.[31] Representatives from Electric Power Research Institute (EPRI) and Sandia National Laboratories (SNL) provided an update on these and other related activities.

2.3.1. Related EPRI-Sponsored Activities

Matt Nudi provided an overview of related EPRI severe accident research activities. His presentation (see Appendix C.3.1) focused on three topics: EPRI activities addressing recommendations in the FY2021 DOE forensic report; recent activities to enhance and apply the EPRI-sponsored MAAP code; and recent activities to enhance and apply the EPRI-sponsored Generation Of Thermal Hydraulic Information for Containments (GOTHIC) code.

The FY2021 DOE forensics report recommended activities to compare risk-important parameter uncertainty distributions developed for systems analysis codes calculations. EPRI extended its MAAP plant and scenario uncertainty analysis approach, which was initially developed for MAAP Fukushima assessments, to other plant designs (PWRs and BWR Mark II and III designs) and continued its participation in the international Management of Uncertainties of Severe Accidents (MUSA) projects. Significant insights have been gained from uncertainty evaluations.[33] For example, results indicate that accident scenario input uncertainties have the potential to outweigh phenomenological uncertainties. These insights are being used to optimize BWROG and PWR Owners Group (PWROG) severe accident guidance implementation efforts, identifying actions and decisions that should be emphasized in training and drills. Efforts to benchmark the MAAP RCIC model using Tennessee Valley Authority (TVA) operational data will be completed in 2022, and implementation of the BWR water level instrumentation model is being tracked for consideration.

During the last year, EPRI released MAAP Version 5.06 with several code enhancements obtained from Fukushima analyses (improved nodalization and improvements in models for predicting ex-vessel relocation, corium jet fragmentation heat transfer models, suppression pool quenching, core fission product scrubbing, and gamma-induced water radiolysis). EPRI is also continuing MAAP code modernization efforts. This activity will cumulate in the release of MAAP6, a single source code (in C++) capable of analysis for a large range of reactor types, including PWRs, BWRs, Canadian Deuterium Uranium (CANDUs) reactors, and Voda-Vodyanoi Energetichesky Reaktors (VVERs).

To improve its applicability for Fukushima D&D activities, EPRI has also completed several efforts to enhance the capabilities of the GOTHIC code. These efforts have primarily focused on GOTHIC aerosol transport capabilities. As confirmatory SGTS and shield plug contamination measurement data become available, EPRI plans to apply GOTHIC to gain insights regarding fission product transport that occurred during the 1F accidents in the proposed OECD/NEA FACE project (see Section 2.1.2).

Subsequent discussions explored MAAP modeling limitations and conclusions regarding scenario input uncertainties versus phenomenological uncertainties. Mr. Nudi agreed that conclusions about input uncertainties dominating were specific to the reactor design, the scenario, and the calculation end use. Discussions emphasized several important insights and recommendations:

Insight: The Forensics effort has led to several improvements in MAAP models, and uncertainty analyses with the improved code have been used to optimize Severe Accident Guidance (SAG) training[††].

Recommendation: EPRI should complete efforts to benchmark MAAP RCIC models against Tennessee Valley Authority data in which the RCIC system ran on April 27, 2011 after a tornado.

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

As emphasized in the FY21 report, several industry experts emphasized the importance of the water level sensor model to provide an indication of instrumentation readings available to plant staff (and compare it with actual predicted water levels within the RPV and suppression chamber [SC]). It was observed that in response to Generic Letter 84-23[34], the BWRs located at one U.S. plant site (and possibly others) implemented significant physical changes in the routing of the water level sensor reference leg, decreasing the portion within the PCV from approximately 6 to 0.6 m (20 to 2 feet). Calculations with this new model could provide important insights for operator guidance and training, but it will be important to obtain accurate plant-specific geometry information.

2.3.2. MELCOR Update and Related NRC-Sponsored Activities

Four presentations on MELCOR development and application activities were given by David Luxat. The fourth presentation focused on Light Water Reactor (LWR) activities; whereas the first three presentations focused on non-LWR activities.

In his first three presentations, Dr. Luxat highlighted enhancements and application activities that allow MELCOR to simulate several types of advanced reactors, including high temperature gas reactors, molten salt reactors, sodium fast reactors, and heat pipe reactors. As new models are incorporated into the code, capabilities are demonstrated for typical transients in representative designs. Results from these calculations are providing important insights about the response of non-LWRs and gaps where model validation data are needed.

In his fourth presentation, Dr. Luxat reviewed SNL efforts to improve MELCOR RCIC modeling.[35] As indicated in his presentation (see Appendix C.3.2), this effort focused on simulating the self-regulating modes inferred from 1F2 and 1F3 data, supplemented by data obtained from recent Texas A&M University (TAMU) testing (see Section 2.4.5). Analysis results indicate RCIC systems can exhibit three self-regulating modes of operation: stable, degraded with constant turbine speed and stable water injection; unstable with oscillations in turbine speed and RPV injection; and semi-stable, degraded with significant turbine speed/injection fluctuations. Nozzle modeling choices, such as the number, orientation, and elevation of nozzles placed about the rotor, were found to influence the predicted RCIC mode of operation.

Subsequent discussions focused on questions related to RCIC model implementation and the potential for alternate conditions to affect the conclusion that nozzle modeling choices affected results (additional discussions related to RCIC operation are found in Section 2.4.5). Additional testing for alternate conditions may lead to additional insights. These discussions led to one insight and recommendation:

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

Insight: Improvements in RCIC modeling, supplemented by TAMU data, have led to significant improvements in our understanding of RCIC performance and operator guidance.

Recommendation: As remaining TAMU test results become available, the MELCOR RCIC modeling and application effort should be updated.

2.4. 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 2.1), other information released by organizations from Japan during the last year, and progress on related U.S. activities.

2.4.1. Topic Area 1 - Component/System Performance

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 the following inspections and evaluations:1F1 reactor well vent line operation, 1F2 drywell (DW) and PCV integrity, shield plug placement, and cavity differential pressure adjustment line samples; and 1F3 main steam isolation valve (MSIV) bellows and 3rd floor damage surveys.

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]

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]	Partially flooded [58]	Partially flooded [58]
	Rusted handrails/equipment [36]	Non-rusted handrails/ equipment [36,59]	Non-rusted handrails/ equipment [36,60]
	NA	Some room penetrations tested, no leakage observed [61]	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]

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 Standby Liquid Cooling (SLC) pump injection line in the DW. This penetration is designated as X-27 in 1F1.
d. X-53 is vacant for 1F2 and 1F3, allowing these penetrations to be used for DW investigations.
e. X-53 is the HPCI steam supply penetration, and X-54 is the HPCI steam instrument pipe penetration for 1F1. X-11 is the HPCI steam supply penetration for 1F2 and 1F3.

Topic Area 1 leads discussed recent observations regarding changes in 1F1 and 1F3 PCV water levels after the February 2021 seismic event. As discussed in [55], the 1F3 PCV water level dropped and 1F1 PCV water level decreased. Although new leaks may have occurred, it is believed to be more probable that the seismic events caused changes in existing leakages (the 1F1 vacuum rupture line bellows and 1F3 main steam pipe penetration). Additional water injection tests (and increased monitoring) are being used to provide insights regarding the situation. This topic is discussed further in Section 2.4.3.

In addition, leads highlighted recent measurements obtained from 1F1/1F2 vent line investigations.[71] High radiation measurement data (indicative of significant fission product deposition) provide insights regarding backflow from 1F1 to 1F2 (which wasn't vented). Topic Area 1 leads concluded their presentation by re-emphasizing two of their FY21 recommendations. First, the leads continued to recommend that additional investigations be performed to improve understanding of safety relief valve (SRV) performance. Although risk assessments typically consider that SRVs reclose at elevated containment pressure, additional efforts are needed to gain insights about actions not considered in Probabilistic Risk Assessments (PRAs), such as the potential for reduced accumulator pressure to lead to a partial opening of a SRV. Second, leads recommended the need for continued focus on instrumentation readings that could be impacted by plant conditions.

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

***Insight*:** The influence of reduced accumulator pressure on intermediate opening of SRVs with possible area reductions is not typically addressed in PRAs.

Recommendation: Given the importance of SRVs in Emergency Operating Procedure (EOPs) and SAMGs, additional analyses (and possibly testing) should be performed to evaluate the risk impact of SRV behavior due to low N_2 accumulator pressure.

The performance of SRVs, which may vary with vendor, design, and environmental conditions, should be better understood and communicated to plant operators. Analyses should consider the effects of PCV pressure and temperature, operator actions, core decay heat, RCIC steam requirements, and RCIC cold water injection. Additional radiation measurement data from plant piping, if available, may provide insights about valve operation.

***Insight*:** Plant conditions may impact the plant instrumentation readings available to operators.

Recommendation: Further evaluations should be performed to understand the impact of plant instrumentation affected by plant conditions. In addition to implementing the water level model into the MAAP code, analyses should be conducted to gain insights regarding the impact of plant conditions on instrumentation readings.

Some clarifications were made to existing examination requests (see Appendix B), but area leads did not propose any new items.

2.4.2. Topic Area 2 - Radionuclide Surveys and Sampling

Topic Area 2, which is led by Lucas Albright and David Luxat, SNL, focuses on insights from recent radionuclide survey and sampling information from the affected units at 1F. In their presentation (see Appendix C.4.2), the leads focused upon insights gained by considering results from NRAJ and TEPCO radiation surveys (see Sections 2.1.2 and 2.1.3.3 and Reference [25]) in conjunction with predictions from MELCOR sensitivity studies. In particular, their presentation focused upon radiation surveys (1F1, 1F2, and 1F3 shield plugs and the 1F2 reactor well) and variations in severe accident progression phenomena and source term predictions (considering uncertainties related to injection and containment leakage). Results provide insights regarding the impact of water injection timing and magnitude (relative to vessel failure timing) on fission product transport and release for various radionuclide groups (and the associated source term) as well as in-vessel hydrogen generation.

These initial MELCOR sensitivity calculations only consider decay heat sources. Subsequent discussion focused on the need to consider other heat sources, such as exothermic chemical reactions from in-vessel oxidation. Nevertheless, these initial calculation results provide insights into what factors caused 1F2 and 1F3 shield plug contamination levels and dose rates to be significantly higher than values measured in 1F1.

Insight: Containment failure is delayed when water injection begins prior to significant core degradation.

Insight: Early water injection can reduce containment and environmental source terms.

Insight: Injection reduces source terms for key radionuclide groups.

The Topic Area 2 discussions led to the following recommendation:

Recommendation: Given the importance of water addition in the EOPs and SAMGs, additional analyses should be performed by SNL to refine their findings from MELCOR sensitivity studies.

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

2.4.3. Topic Area 3 - Debris Endstate

Topic Area 3 included two presentations: a presentation by Mitch Farmer, ANL, on recent core debris location evaluations; and a presentation by Marty Plys, Fauske and Associates, LLC, on data needs to support development of a passive interim storage container design.

In his presentation (see Appendix C.4.3.1), Dr. Farmer provided comments and suggestions related to 1F2 X-6 penetration examination information and JAEA 1F sample evaluations. He also suggested a new measurement to provide insights regarding debris location and permeability when TEPCO conducts tests varying PCV water levels.

- Dr. Farmer first focused on X-6 penetration investigations (see Section 2.1.3.2 and Reference [72]). He observed the sediment distribution indicates aerosols were deposited as gas flowed out of the PCV (the sediment height is highest near the penetration hatch on the PCV exterior). This suggests that MCCI contributed to these deposits. Although observations reported in Section 2.1.3.2 were interesting, he emphasized that aerosol composition evaluations may also provide important insights regarding the timing when PCV leakage occurred and suggested future investigations should include chemical analysis to determine whether the sediment originated from paint, cabling, or MCCI. Several existing information requests, such as PC-3b, PC-17, PC-18, PC-20, and PC-22, suggest future sample analysis should include options to detect concrete oxides. As emphasized in detailed Appendix B.2 tables supporting these requests, this information is important for future D&D operations (i.e., the presence of concrete oxide slag significantly affects debris mechanical properties) and for our understanding of severe accident progression.
- Dr. Farmer then commented on JAEA examination results (see Section 2.1.4). As indicated in Slides 9 through 18 of his presentation, JAEA chemical analysis results can be used to provide insights regarding the conditions in which these sample particulates were formed, the location from which particulates originated, and the cooling rates these particulates experienced. Although caution must be exercised in extrapolating results from a limited number of samples, such examinations can be used to gain accident progression insights, such as peak temperatures within the reactor core and the presence of MCCI. He suggested JAEA consider several additional evaluations that could provide MCCI insights (see Slides 17 and 18 in Appendix C.4.3.1). It was also observed that current plans to include an international round robin to benchmark and characterize uncertainties in hot cell examination results will provide confidence in results obtained from the new CLAD facility.
- Finally, Dr. Farmer proposed a possible action to gain insights regarding debris permeability as TEPCO conducts additional coolant suspension tests to support their plans to reduce 1F1 and 1F3 PCV water levels.[55] Although sufficient water must be available to ensure the coolability of relocated debris, TEPCO plans to lower PCV water levels to reduce additional PCV damage during future seismic events (see Section 2.4.1). Coolant suspension tests have confirmed debris coolability when water

injection rates are reduced. When injection rates are varied, Dr. Farmer proposed available instrumentation and visual images could be used to gain insights regarding debris permeability and dryout limits (as well as insights regarding prototypic debris accumulation height distribution). Namely, when water levels are lowered to a location near the pedestal doorway, the injection rate could be temporarily increased until videos indicate water begins to spill over the doorway. Subsequent discussion on this suggestion noted that accurate injection flows (from the core spray and feedwater lines) must be measured. It was also observed that there was the potential for some of the injected water to not flow through the RPV into the pedestal region (adversely impacting measurement accuracy).

In his presentation (see Appendix C.4.3.2), Dr. Plys highlighted Fauske & Associates efforts to develop a cask design for passive vented interim storage of 1F debris. Using experience from Hanford and Sellafield, the design has been developed to meet requirements for radiological containment as well as hydrogen, decay heat, and water removal. The current design assumes a pessimistic bounding source for hydrogen removal; the design could be optimized (with considerable cost savings) if additional data were obtained to better characterize water removal. Specific debris properties of interest include: debris macroscopic and internal porosity (permeability), particle size distribution, and thermal conductivity (diffusivity, density, and specific heat capacity). In many cases, Dr. Plys noted that evaluations of surrogate material (from prior MCCI tests) might be acceptable to the regulator.

Topic Area 3 discussions resulted in several insights and recommendations:

Insight: Coolant injection suspension test data will be used to provide a technical basis for reducing water injection and lowering PCV water levels.

Recommendation: Additional consideration should be given to the proposal of temporarily increasing coolant injection flowrate during future coolant suspension tests to gain insights about debris permeabilities, dryout limits, and accumulation distributions.

Following the meeting, Dr. Farmer provided additional information regarding his suggestion to temporarily increase water injection to gain insights about debris insights about debris permeabilities, dryout limits, and accumulation distributions.[73] Although several representatives from TEPCO expressed interest in such an evaluation, practical limitations (associated with the current D&D schedule, radiation-induced degradation on video devices and other investigation equipment, and uncertainty in obtained data) precluded such a test in the near term.[74] Hence, the U.S. did not recommend any new information request related to this activity at this time (although it is recognized that Japan may consider it in longer term investigations).

Insight: Sample examinations provide important information required for future D&D activities (offering the potential to reduce costs) and useful for increasing our knowledge about the accident progressions in each unit.

Recommendation: Future sample examination efforts should consider information requests to allow detection of MCCI components (i.e., detection of Ca, Si, Mg, and Al in concrete oxides).

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

As noted in Section 2.1.4, U.S. experts were pleased that their comments on proposed examinations are being considered by JAEA as they update Reference [29]. In addition, U.S. experts observed that the planned international round-robin exercise, which will use debris containing prototypic materials from a prior MCCI test, [see Section 2.1.2] will add confidence in data obtained from these new facilities and could provide the properties requested by Dr. Plys to support the passive vented interim storage task.

Insight: The Fauske & Associates efforts to develop a cask design for passive vented interim storage has the potential to significantly reduce 1F fuel debris storage costs.

Insight: Debris properties, such as porosity, morphology and particle size distribution, are critical for several D&D activities: 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.

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).

In summary, no new information requests were recommended for Topic 3. However, several prior requests related to examinations from debris samples found in Appendix B were clarified (including the need for material properties and detection of concrete oxide components).

2.4.4. Topic Area 4 - Combustible Gas Effects

Reviews of higher resolution videos have led NRAJ and TEPCO to explore the possibility that other sources of combustible gases (e.g., thermal degradation of cabling materials or coatings within the PCV) contributed to the 1F3 and 1F4 explosions (see Sections 2.1.2 and 2.1.3). In Topic Area 4, presentations by Wison Luangdilok, H2 Technology LLC, and Mike Salay, U.S. NRC, provided additional insights on this subject.

In his presentation (see Appendix C.4.4.1), Dr. Luangdilok compared his 2020 ballpark estimate of combustible gases in terms of hydrogen equivalent (the amount of combustible gas required to release the energy associated with hydrogen combustion) based on a global mass balance (see Section 6 of [4]) with amounts of combustible gas estimated by various organizations for the 1F3 event in the 2020 Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant Phase 2 (BSAF2) program.[75]. His 2020 ballpark estimate suggests the observed large fireball-shaped explosion in 1F3 and the migration of combustible gases from 1F3 to cause the 1F4 explosion would require between 2100 and 2400 kg of hydrogen equivalent. From a global mass balance perspective, he observed combustible gases must be rapidly generated such that there is a sufficient amount of combustible gases produced at the right time to support the observed explosions at the recorded times. In order for this to be predicted by current systems analysis codes, Dr. Luangdilok indicated vessel lower head failure would need to be predicted early so that MCCI could generate sufficient amounts of combustible gases. NRAJ investigations of unaccounted sources of combustible gases, from thermal decomposition of hydrocarbon materials in cable insulators and paint, may provide insights about the missing sources of combustible gases. If the magnitude of these sources of hydrocarbon gases is shown to be substantial, the required amounts of combustible gas generation from MCCI to produce the observed explosions could be reduced.

Dr. Luangdilok then proposed a complete explosion mechanism (Slides 26 and 27 in Appendix C.4.4.1) for the 1F3 explosion based on the NRAJ insights obtained from the high resolution video and the additional insights gained from previous hydrogen explosion experiments (unrelated to 1F research) with the equivalence ratio, Φ, ranging from a value less than one to a value of six (corresponding to the entire range of flammable hydrogen-air mixtures, from a lean to very rich mixture).

- This proposed mechanism suggests the explosion on the 4th floor of 1F3 is an explosion of $\Phi<1$ (lean fuel), while the 5th floor mushroom fireball explosion is a $\Phi >1$ (rich fuel).

- Experiments discussed in his presentation indicate a mushroom-shaped long-stem fireball requires a gas mixture that is very rich in fuel ($\Phi >1$). Dr. Luangdilok's review led him to conclude that the fuel in the 1F3 event was much richer than the stoichiometric ratio so that, following an initial ignition, there is excess unburned fuel to drive the fireball formation.

- A possible range of 1F3 5th floor pre-ignition concentrations of hydrogen and steam were presented on the Shapiro diagram (slide 30 in Appendix C.4.4.1). These conditions are the boundary between the flammable and the non-flammable mixtures on the fuel-rich side of the diagram. Although he had not calculated the amount of hydrogen equivalents, Dr. Luangdilok estimated that values will be lower than his 2020 estimate if the presence of steam is considered.

Finally, Dr. Luangdilok commented that the NRAJ proposed combustion experiments are limited to studying the chemistry aspect of the combustion problem. The fluid dynamics aspect of the combustion problem do not appear to be addressed by the current proposed scope. He suggested NRAJ consider additional experiments with the following attributes:

- Designs similar to ones discussed in his presentation (but with the combustion vessel designed to have the same geometric and failure characteristics as the 4th and 5th floors of the1F3 reactor building)

- The hydrogen concentration should range from $\Phi <1$ (to provide basic understanding) to $\Phi >1$ (to provide insights regarding the 1F3 explosion)

The presentation (see Appendix C.4.4.2) by Dr. Mike Salay summarized insights gained from prior efforts to quantify sources of organic materials within a PCV. He listed several references that can be used to quantify the mass of organic cable material within a containment (as well as on-going efforts investigating this topic). In addition, he characterized degradation properties for typical materials (e.g., Hypalon®‡‡, Polyvinyl Chloride, etc.) found in cables. He noted combustible gas generation can be initiated at temperatures above 100 °C. Of particular interest were data indicating thermal degradation (and combustible gas generation) significantly increased with radiation exposure.

Discussion time was limited during the FY2022 Forensics Panel meeting. Before expanding the scope of the NRAJ program, U.S. experts expressed interest in first quantifying the potential for cable thermal degradation to be a significant unrecognized source of non-condensible and flammable gas generation in late phase severe accident progression. It was recommended that a 'back-of-the-envelop' calculation be completed to evaluate whether it was possible for cable degradation to contribute significantly to the combustible gas generation. Appendix D was developed after the meeting to address this recommendation. As discussed in this appendix, initial scoping calculations indicate thermal degradation of cable materials will

‡‡ Hypalon® is a registered trademark of DuPont for chlorosulfonated polyethylene rubber (CSPE).

produce less than 2% of the combustible gas generated from other in-vessel sources of hydrogen generation during the 1F3 accident progression.

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

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

Insight: There are uncertainties in predicting combustible gas generation, mixing, and transport. Proposed NRAJ and TEPCO testing could produce additional insights regarding an additional source of combustible gas generation; this source could affect existing, new, and advanced reactors. Prior testing indicates that radiation significantly increases the amounts of combustible gas generation associated with cable thermal degradation (and combustible gas generation).

Recommendation: Topic Area 4 investigations should remain focused on identifying and reducing, where possible, uncertainties that impact accident management strategies. To better understand this potential source of combustible gas in existing LWRs as well as new plant designs (LWR and non-LWRs), U.S. experts should develop an initial estimate of the potential impact of cable thermal degradation during a severe accident. As additional information on cable testing from Japan becomes available, this panel may wish to develop a list of questions and comments.

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

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

For Topic Area 5, information emphasizing the importance of information coming from Forensics Examinations was provided by two representatives from the BWROG: Phillip Ellison, GE Hitachi and Randy Bunt, Southern Company. The information included three presentations (supporting information related to these presentations is provided in Appendix E):

- ***BWROG Computer Based Training (CBT) for SAMGs Update.*** In this presentation (see Appendix C.4.5.1), Dr. Ellison provided an overview of BWROG efforts to implement Fukushima lessons learned into severe accident guidance and training. In addition to revising guidance, the BWROG has developed a CBT program (with 32 unique courses or modules) that emphasizes Fukushima case studies. This training, which is part of the U.S. utilities commitment related to severe accident management to the U.S. NRC, is provided to licensed and non-licensed plant operators along with decision makers and implementors of accident management programs. At the time of this meeting, more than 3000 individuals have been certified through this CBT program.

  Initial training certifications require 10 to 12 hours, but it is hoped that training times can be reduced (without impacting technical content) to around 8 to 9 hours. The CBT is available on the National Academy for Nuclear Training e-Learning or NANTeL (U.S. Utility members) and Smart Open Universe Learning or SOUL (NRC and International BWROG members) platforms. This BWROG-developed Severe Accident Interactive Learning (SAIL) training relieves plant owners/operators from having to develop and provide their own training, resulting in cost savings.

  This effort is part of a larger industry effort to develop and maintain technical CBT modules for targeted areas of engineering expertise. As described in Appendix E.1, important safety lessons were learned from Fukushima; the BWROG is committed to perform the required maintenance and updates of the SAMG and the FLEX CBT modules. The BWROG is ensuring DOE Forensics insights are being utilized to improve plant operations, reduce total operations cost and improve safety in the U.S. and worldwide. The BWROG is committed to perform the required maintenance and updates of the SAMG CBT modules.

This BWROG initiative, along with a broader industry effort implemented through the Exelon Powerlabs Advanced Engineering Training (AET) program, has developed technical CBT modules for targeted areas of engineering and operations expertise. As described in Appendix E.1, important safety lessons were learned from Fukushima. The efforts of the BWROG and Exelon Powerlabs, LLC, is ensuring forensics insights are being utilized to improve plant operations, reduce total operations costs, and improve safety in the U.S. and worldwide.

- ***Terry™ Turbopump Testing 2021 Update.*** In this presentation (see Appendices C.4.5.2), Mr. Bunt provided an update on the Terry™ Turbine Expanded Operating Band (TTEXOB) project, an international collaborative effort between the BWROG, IAE, DOE [funding participation by Idaho National Laboratory (INL), SNL, and TAMU], and EPRI. The goal of this project, which was initiated to investigate the long duration RCIC performance observed at 1F2 and 1F3, is to expand and define the actual operating limitations (margins) of Terry™ Turbine systems [i.e., RCIC/Turbine Driven Auxiliary FeedWater (TDAFW)] used in the nuclear industry.

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

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

  Results from the TTEXOB project [35,76,77,78] and 1F forensics investigations have led to revised generic operating guidance for RCIC system operation in the BWR fleet. When this 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. TTEXOB testing provided a basis for determining the number of manual valve actuator turns for minimum and maximum RCIC flow, simplifying the manual adjustment of flow during a station blackout (SBO) event. In addition, test results led to the addition of a caution to applicable site procedures or guidance associated with RCIC operation at elevated oil temperatures. These changes have led to the recommendation that site PRAs should be reviewed and updated to better reflect RCIC system operation for beyond design basis events. An early insight from 1F forensics investigations was that continued RCIC operation during some events is not strongly influenced by water ingestion. This led to revised guidance and procedures that the RCIC trip on high water can be removed, which simplifies plant operation during SBOs. This insight was trans-

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

mitted early to the fleet and 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 (see Appendix E.2).

Available insights were already incorporated into operator guidance. The ongoing action for the BWROG Procedures Committee is to add references to the testing program in operator guidance and to include a definitive statement that if the Terry^{TM} Turbine driven pump is the only feed to the RPV, then there is industry experience and scientific experimental data to show the machine can continue to pump water even in conditions outside of the normal design or operating ranges. The actions taken addressed bypassing many RCIC trip functions to provide capability to cope with extreme external events. These actions proved helpful in the mitigation of the DAEC event.

It was observed that additional TAMU RCIC testing, beyond the scope of the currently-funded program, is needed to define some of the one-off scenarios postulated in the mitigation actions for the suite of emergency procedures. These one-off scenarios occur because the suite of emergency procedures is tasked with addressing all mechanistically possible conditions. Current models and 1F insights are used to identify these one-off situations.[35] Identified scenarios include situations in which water is present in lower portions of the turbine casing and additional situations in which the turbine is operating in a steam self-regulation mode. Water may enter the turbine casing due to spillover from the RPV, water ingress during periods when the turbine is sitting idle, or other causes. This water can provide a drag on the fluid flowing through the turbine and on the turbine wheel itself if the accumulated water level reaches the bottom of the turbine wheel. With steam as the driving force of the turbine, the steam has some degree of superheat (or moderate quality) after the reactor has re-pressurized due to loss of coolant injection (because the turbine has stalled). Then, the excess heat in the steam will act to heat the water in the turbine and add a boil-off mechanism to what would otherwise be an entrainment-limited sweep-out phenomenon.

Tests with the ZS-1 can be performed to clarify the effect of water on turbine operation. Additional small-scale steam-water self-regulation tests could evaluate the effect of energy exchange between the steam and water. Air-water with the GS-2 did not observe a mode of self-regulation in which the turbine power oscillated between high and low power (one hypothesized model of performance at 1F2). If this mode of no self-regulation is observed with steam in small scale ZS-1 tests, then it would be consistent with the full-scale air-water self-regulation data. This additional insight would be valuable because self-regulation in the 1F units has been postulated to occur. If this mode of self-regulation with steam on the small-scale is observed, the discrepancy with the full-scale air-water data would need to be explained, and this process may shed further insights into the full-scale air-water data.

- ***Radiation Best Practices – Lessons Learned from Fukushima Daiichi D&D.*** In this presentation (see Appendix C.4.5.3), Mr. Bunt provided an update on DOE/BWROG efforts to develop a draft letter report reviewing new technologies being deployed at Daiichi to facilitate D&D. Developed in response to a FY2021 recommendation, the draft report provides: (i) an overview of new D&D technologies and measures, (ii) information to characterize their effectiveness, and (iii) areas where future research would be beneficial to apply these new technologies to routine O&M activities. The presentation included more detailed information for two example technologies: (i) small compact gamma cameras combined with software for real-time monitoring using two-dimensional (2D) or three-dimensional (3D) visualizations showing photos, radiation levels, and temperatures; and (ii) plastic scintillation fibers (PSFs) for real-time radiation detection and monitoring of contaminated water. Mr. Bunt concluded his presentation suggesting the draft letter report be included as an appendix in this DOE FY2022 report. He also suggested additional efforts be devoted to facilitate deployment of these technologies for routine O&M activities. Candidate funding sources, such as the DOE Nuclear Energy

Enabling Technology (NEET), Nuclear Energy University Programs (NEUP), and Light Water Reactor Sustainability (LWRS) programs, should encourage bi-lateral cooperation to facilitate deployment of these technologies.

Topic Area 5 discussions emphasized several insights and recommendations:

Insight: Expert panel members expressed support for the new CBT efforts that incorporate lessons learned from Daiichi examinations into severe accident training for operators and other decisions-makers.

Insight: Expert panel members expressed support for the Terry^{TM} Turbopump Test program, which has already led to important reactor safety insights and reduced plant operating costs. Insights from this effort were successfully used by operators to withstand the DAEC LOOP event.

Recommendation: Funding should be provided to support additional testing that will help validate RCIC extended operational regimes. Because such work can be done in an academic setting, we suggest a work scope be developed under the DOE NEUP or the LWRS research programs.

Insight: Expert panel members are interested in lessons obtained in the area of radiation protection (as described by Dr. Mizokami in Section 2.1.3.1) and in new technologies being deployed for D&D at Fukushima (as described here and in Section 2.1.4).

Recommendation: The draft information highlighting new D&D should be included as an appendix to this FY2022 report. [This information now appears as Appendix E.3]

Recommendation: Additional efforts should be devoted to facilitate deployment of new D&D technologies from 1F for routine O&M activities. It is suggested that workscope be funded under the DOE LWRS programs.

The draft report highlighting new D&D technologies was updated and included as Appendix E.3 of this document. Leads did not propose any changes to the U.S. list of examination requests in Appendix C of Reference [3]).

2.5. Updates to U.S. Information Requests

As described in Section 1.1, a primary objective of the forensics effort is to provide consensus U.S. input for high priority time-sequenced examination tasks and supporting research activities that can be completed with minimal disruption of D&D plans for Daiichi. Initial information requests were developed in FY2015. Every year, these information requests are reviewed and as appropriate, updated. Since FY2015, several new information requests were added and the status of several U.S. information requests was modified. Appendix B presents the current version of these information requests.

An important aspect of request documentation is to include a description of why the information is needed and how obtained information will be used. U.S. participants factored in experience from TMI-2 examinations, prioritizing information that would be beneficial for defueling efforts and for operations and safety. In addition, participants considered information provided by representatives from TEPCO Holdings and other Japanese organizations (e.g., JAEA, NDF, and NRAJ). As noted in Section 2.1.3.4 (see Appendix C.1.3.4 slides), TEPCO plans for future investigations consider U.S. information requests. In addition, as Japan moves forward with the FACE project, it is important that the benefit of information requests be documented (see Section 2.1.2 and Appendix C.1.2 slides). NRAJ plans to have follow-on discussions about the information requests and supporting information developed by the U.S. DOE Forensics Effort.

During this FY2022 meeting, experts developed no new information requests. In addition to noting that several requests have been addressed, experts modified several existing requests to reflect insights from recent investigations, and provided additional documentation to clarify how information could benefit D&D as well as operation of the existing fleet, new, and advanced reactors. Notable changes identified during the review of information requests in Appendix B (summary information requests found in Appendix B.1 as well as detailed descriptions for selected information requests found in Appendix B.2) include the following:

- RB-8 was updated to reflect that its importance may have increased due to the February 2021 seismic events and that new technologies may facilitate obtaining the requested information.
- RB-9 was updated to reflect new information provided by NRAJ and planned shield plug investigations.
- RB-11 was updated to reflect new information provided by TEPCO and NRAJ.
- RB-15 was modified, based on input provided by TEPCO, to add a request that evaluations of water samples from the Reactor Building Closed Cooling Water System (RCW) include detection of metals (Ca, Si, Mg, and Al) found in oxides from MCCI
- Several requests (PC-3b, PC-17, PC-18, PC-20) were updated to note that JAEA is publishing additional examination results and more results are being prepared for publication.
- Several request (PC-7, PC-9, PC-21) were modified to reflect on-going interest in the potential for thermal degradation of cable insulation and sheaths and PCV coatings to be a source of combustible gas generation.

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. FY2022 meeting presentations and discussions again emphasize the importance of this effort and the benefit being obtained by the nuclear enterprise. In addition, several presentations from Japan emphasized that input from the U.S. forensics effort is being considered in future D&D efforts. Key findings and associated recommendations from this meeting are summarized in Section 3.

3. KEY FINDINGS AND ASSOCIATED RECOMMENDATIONS

This section summarizes recommendations developed from the FY2022 U.S. DOE Forensics Effort meeting. In this section, the recommendations listed in Section 2 are grouped into three findings.

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 U.S. presentations, the nuclear enterprise has and continues to use Fukushima insights to enhance the safety of the operating fleet. In addition to updating assessments of the potential hazards associated with external events, industry has taken actions to increase the equipment available to respond to beyond design basis events and to improve operator guidance and support procedures for severe accident progression and mitigation. New BWROG CBT ensures lessons learned from forensics examination are incorporated into severe accident training for operators and other decisions-makers.

NRAJ evaluations of the explosions observed at 1F1 and 1F3 are providing new insights with respect to accident progression at these units. Proposed NRAJ and TEPCO testing could produce additional insights regarding an additional source of combustible gas generation.

The broad participation in the FY2022 meeting demonstrates the importance of information exchanged during these forums. Although in-person meetings offer more direct communication between experts in reactor safety and operations, the FY2022 hybrid meeting (with virtual and in-person attendance) allowed broad domestic and international participation (an increase of approximately 25% from FY2021 and more than twice the number of participants in FY2019).

Recommendation: U.S. organizations should continue to monitor and evaluate information obtained from 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: Topic Area 4 investigations should remain focused on identifying and reducing, where possible, uncertainties that impact accident management strategies. To better understand this potential source of combustible gas in existing LWRs as well as new plant designs (LWR and non-LWRs), U.S. experts should develop an initial estimate of the potential impact of cable thermal degradation during a severe accident. As additional information on cable testing from Japan becomes available, this panel may wish to develop a list of questions and comments.

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

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.

Several presentations by Japanese representatives emphasize that future activities, such as plans for 1F debris hot cell examinations, an international round robin in the proposed new FACE project, future 1F RB and PCV examinations, and PCV water level reduction activities, consider information documented in the FY2021 forensics effort report.

Recommendation: Additional consideration should be given to the proposal of temporarily increasing coolant injection flowrate during future coolant suspension tests to gain insights about debris permeabilities, dryout limits, and accumulation distributions.

Recommendation: Future sample examination efforts should consider information requests to allow detection of MCCI components (i.e., detection of Ca, Si, Mg, and Al in concrete oxides).

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).

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.

Data from 1F2 and 1F3, supplemented by the Terry™ Turbopump Test program data, have led to significant improvements in our understanding of RCIC performance and 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. Significant insights also continue to be gained regarding the importance of severe accident phenomenological and scenario uncertainties. These insights help focus improvements in operator guidance and training. Discussions during the FY2022 forensics meetings proposed additional investigations, such as consideration in PRAs of the influence of reduced accumulator pressure on intermediate opening of SRVs with possible area reduction and the impact of plant conditions on plant instrumentation readings, having the potential to offer additional benefits to the operating fleet. Expert panel members are also interested in additional lessons that can be obtained in the area of radiation protection and if new technologies being deployed to facilitate 1F D&D could be used for routine plants operations and maintenance.

Recommendation: EPRI should complete efforts to benchmark RCIC MAAP models against Tennessee Valley Authority data in which the RCIC system ran on April 27, 2011 after a tornado.

Recommendation: Further evaluations should be performed to understand the impact of plant instrumentation affected by plant conditions. In addition to implementing the water level model into the MAAP code, analyses should be conducted to gain insights regarding the impact of plant conditions on instrumentation readings.

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

Recommendation: Given the importance of water addition in the EOPs and SAMGs, additional analyses should be performed by SNL to refine their findings from MELCOR sensitivity studies.

Recommendation: As remaining TAMU test results become available, the MELCOR RCIC modeling and application effort should be updated.

Recommendation: Funding should be provided to support additional testing that will help validate RCIC extended operational regimes. Because such work can be done in an academic setting, we suggest a work scope be developed under the DOE NEUP or the LWRS research programs.

Recommendation: Given the importance of SRVs in EOPs and SAMGs, additional analyses (and possibly testing) should be performed to evaluate the risk impact of SRV behavior due to low N_2 accumulator pressure.

Recommendation: The draft information highlighting new D&D should be included as an appendix to this FY2022 report. [This information now appears as Appendix E.3]

Recommendation: Additional efforts should be devoted to facilitate deployment of new D&D technologies from 1F for routine O&M activities. It is suggested that workscope be funded under the DOE LWRS programs.

4. REFERENCES

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

2. J. Rempe (editor), L. Albright, P. Amway, N. Andrews, S. Basu, F. Bolger, R. Bunt, M. Cibula, M. Corradini, P. Ellison, M. Farmer, J. Gabor, R. Gauntt, H. Hoshi, S. Ito, K. Iwanaga, K. Kirkland, K. Klass, W. Kikuchi, T. Kobayashi, R. Kojo, S. Koyama, S. Kraft, M. Kurata, R. Linthicum, W. Luangdilok, D. Luxat, R. Martin, P. McMinn, S. Mizokami, J. Nakano, A. Nakayoshi, M. Nudi, T. Okamoto, C. Paik, M. Plys, J. Rempe, K. Robb, M. Taira, H. Tanoue, K. Voelsing, R. Wachowiak, T. Washiya, P. Whiteman, B. Williamson, M. Yasui, *U.S. Effort in Support of Examinations at Fukushima - November 2020 Meeting Notes with Updated Information Requests*, ANL-20/79, March 2021, doi: 10.2172/1773089, https://www.osti.gov/biblio/1773089-efforts-support-examinations-fukushima-daiichi-november-meeting-notes-updated-information-requests

3. J. Rempe (editor) N Andrews, S Basu, R Bunt, M Corradini, P Ellison, M Farmer, J Gabor, R Gauntt, C Henry, T Honda, T Kindred, K Klass, T Kobayashi, S Kraft, W Luangdilok, R Lutz, D Luxat, S Mizokami, J Nakano, A Nakayoshi, J Rempe, M Plys, K Robb, K Shearer, H Wabakabayashi, P Whiteman, B Williamson, *U.S. Efforts in Support of Examinations at Fukushima-November 2019 Meeting Notes with Updated Information Requests,* ANL-19/48, January 31, 2020, doi:10.2172/1602189. https://www.osti.gov/biblio/1602189.

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

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

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

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

8. J. L. Rempe (editor), P. Amway, R. Bunt, M. Corradini, P. Ellison, M. Farmer, M. Francis, J. Gabor, R. Gauntt, C. Henry, D. Kalinich, S. Kraft, R. Linthicum, W. Luangdilok, R. Lutz, D. Luxat, C. Paik, M. Plys, C. Rabiti, J. Rempe, K. Robb, R. Wachowiak, and B. Williamson, *US Efforts in Support of Examinations at Fukushima Daiichi,* ANL/LWRS-15/2*,* August 2015.

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

10. J. Rempe, et al., "Nuclear Safety Enhancements based on Daiichi Forensics Information", *19th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-19)*, Brussels, Belgium, March 6-11, 2022.

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

12. METI, "Mid-and-Long-Term Roadmap towards the Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station Units 1-4", https://www.meti.go.jp/english/earthquake/nuclear/decommissioning/, last accessed December 2021.[As an example, consider Progress Report dated, September 30, 2020, https://www.meti.go.jp/english/earthquake/nuclear/decommissioning/pdf/mp202109.pdf, last accessed December 28, 2021]

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

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

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

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

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

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

19. TEPCO Holdings, *The 5th Progress Report on the Investigation and Examination of Unconfirmed and Unresolved Issues on the Development Mechanism of the Fukushima Daiichi Nuclear Accident*, December 25, 2017, last accessed May 2019, https://www7.tepco.co.jp/newsroom/press/archives/2017/1485273_10469.html, Includes: 171225e0201.pdf [Slides], 171225e0201.pdf [Main Body], "Evaluation of the situation of cores and containment vessels of Fukushima Daiichi Nuclear Power Station Units 1to 3and examination into unsolved issues in the accident progression - Progress Report No. 5"; 171225e0217.pdf - [Attachment 1-10] "Analysis of the hydrogen explosion at Unit-1 reactor building"; 171225e0101.pdf; 171225e0218.pdf - [Attachment 1-11] "Estimation of accident progression at Unit-1 based on the air dose rate monitoring data"; [Attachment 2-14] "Estimation of reactor water levels at the time when core damage and core melt progressed at Unit-2";171225e0241.pdf [Attachment 3-9] "Estimation of reactor water levels at the time when core damage and core melt progressed at Unit-3"; and 171225e0242.pdf - [Attachment 3-10] Evaluation of the fraction of Unit-3 vent gas that flowed into Unit-4 reactor building".

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

21. TEPCO, websites related to Fukushima activities (e.g., see: https://photo.tepco.co.jp/en/date/2019-e/201912-e/191203-01e.html, https://www.tepco.co.jp/en/insidefukushimadaiichi/index-e.html#/guide12, and http://www.tepco.co.jp/en/decommissiontraject/index-e.html), last accessed December 15, 2021.

22. JAEA, websites related to Fukushima activities (e.g, see: https://tenkai.jaea.go.jp/english/library/accident.html, https://f-archive.jaea.go.jp/index.php?locale=eng, and https://clads.jaea.go.jp/en/rd/map/map.html), last accessed December 15, 2021.

23. NDF, "Technical Strategic Plan 2021 for Decommissioning of the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company Holdings, Inc.', includes Full Report, Overview Report, and Summary Presentation, October 29, 2021, https://www.dd.ndf.go.jp/english/strategic-plan/index2021.html, last accessed December 15, 2021.

24. NDF, email from J. Nakano, to J. Rempe, Rempe and Associates, LLC, "FY22 Meeting Followup", dated December 28, 2021.

25. NRAJ, TEPCO Fukushima Daiichi Nuclear Power Station Interim report on accident investigation and analysis -Examination from September 2019 to March 2021", March 5, 2021, https://www.nsr.go.jp/data/000345595.pdf, last accessed February 15, 2022.

26. K. Konishi, NRAJ, "Response to FY22 Meeting NRAJ Presentation Chat Questions", email to J. Rempe, Rempe and Associates, LLC, December 9, 2021.

27. IAEA report. "IAEA International Peer Review of Mid-and-Long-Term Roadmap towards the Decommissioning of TEPCO'S Fukushima Daiichi Nuclear Power Station", August 2021, https://www.iaea.org/sites/default/files/21/08/review-report-270821.pdf, last accessed December 12, 2021.

28. USNRC report, "Three Mile Island Accident of 1979 Knowledge Management Digest---The Cleanup Experience: A Literature Review", NUREG/KM-0001, Supplement 2, December 2020, https://www.nrc.gov/reading-rm/doc-collections/nuregs/knowledge/km0001/index.html, last accessed January 31, 2022

29. JAEA, *Analysis of Debris Samples of Tokyo Electric Power Company Holdings Fukushima Daiichi Nuclear Power Station (Translated Document)*, JAEA-Review 2020-055, December 2020.

30. J. Rempe, Rempe and Associates, LLC, Email to S. Koyama, JAEA, email attachment, "US Comments on JAEA Report 2020-55", February 13, 2021.

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

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

33. EPRI, *Severe Accident Uncertainty Quantification and Analysis Using the Modular Accident Analysis Program (MAAP),* https://www.epri.com/research/products/000000003002020762, September 27, 2021.

34. U.S. NRC, "Reactor Vessel Water Level Instrumentation BWRS (Generic Letter No. 84-23)", October 1984.

35. L. Gilkey, M. Solom, B. Beeny, and D. Luxat, *Terry Turbopump Expanded Operating Band Modeling and Simulation Efforts in Fiscal Year 2021- Final Report,* SAND2021-7815, June 2021.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

55. TEPCO Holdings, "Handling of the PCV Water Level Drops at Units 1 and 3", presentation by Tokyo Electric Power Company Holdings, Inc., https://www.tepco.co.jp/en/hd/decommission/information/newsrelease/reference/pdf/2021/reference_20210325_01-e.pdf, March 25, 2021, last accessed February 2022.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

70. M. Yasui, NRAJ, "Investigation and analysis of the TEPCO's Fukushima Daiichi NPS Accidents", presentation at U.S. Expert Panel Forensics Meeting, November 2020.

71. NRAJ, "Summary of Investigation and Analyses Items Related to 1F Accident Analyses", https://www.nsr.go.jp/data/000352410.pdf, May 18, 2021, last accessed February 9, 2021.

72. TEPCO Holdings, https://www.tepco.co.jp/decommission/information/committee/roadmap_progress/pdf/2020/d201126_08-j.pdf, November 26, 2020, last accessed February 2022.

73. Farmer, M., ANL, Email to S. Mizokami, TEPCO, dated February 17, 2022.

74. Mizokami, S., TEPCO, Email to M. Farmer, ANL, dated February 25, 2022.

75. W. Luangdilok, "The explosion at Fukushima Daiichi Unit 3 and 4 and implications on the evaluation of 1F3 accident", *Nuclear Engineering and Design*, **362** (2020) 110536.

76. A. Patil, S. Sundar, Y. Wang, M. Solom, K. Vierow Kirkland, and G. Morrison, "Characterization of Steam Impulse Turbine for Two-Phase Flow", *International Journal of Heat and Fluid Flow,* Vol. 79, paper 108439, 2019.

77. A. Patil, Y. Wang, M. Solom, A. Alfandi, S. Sundara, K. Vierow Kirkland, and G. Morrison "Two-phase Operation of a Terry Steam Turbine using Air and Water Mixtures as Working Fluids", *Applied Thermal Engineering*, Vol. 165, paper 114567, 2020.

78. J. Vandervort, G. Lukasik, B. Ayyildiz, M. Solom, A. Delgado, K. Vierow Kirkland, and A. Patil, "Performance Evaluation of a Terry GS-2 Steam Impulse Turbine with Air-Water Mixtures", *Applied Thermal Engineering*, Vol. 191, 116636, 2021.

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

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

81. E. C. Beahm, R. A. Lorenz, and C. F. Weber, "Iodine Evolution, and pH Control", US Nuclear Regulatory Commission, NUREG/CR-5950, 1992.

82. TEPCO, "Sources of Flammable Gases including Organic Compounds", TEPCO presentation at Nuclear Regulator Authority Meeting on 1F Accident Analysis, September 14, 2021.

83. "Fuels - Higher and Lower Calorific Values", The Engineering Toolbox, https://www.engineeringtoolbox.com/fuels-higher-calorific-values-d_169.html, last accessed January 26, 2022.

84. "Heat of Combustion", https://en.wikipedia.org/wiki/Heat_of_combustion, last accessed January 26, 2022.

85. Cameo Chemicals, "Polyvinyl Chloride, Chemical Data Sheet", June 1999, https://cameochemicals.noaa.gov, last accessed January 26, 2022.

86. NEI, *Diverse and Flexible Coping Strategies (FLEX) Implementation Guide,* NEI 12-06, Revision 4, Nuclear Energy Institute, December 2016.

87. U.S. NRC, "Flexible Mitigation Strategies for Beyond-Design-Basis Events", Regulatory Guide 1.226, Rev. 0 (2019) https://www.nrc.gov/docs/ML19058A006.html

88. NEI, *Industry Guidance for Compliance with NRC Order EA-12-051, To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation,* NEI 12-02, Rev 1 (2012).

89. U.S. NRC, "Wide-Range Spent Fuel Pool Level Instrumentation", Regulatory Guide 1.227, Rev 0 (2019). https://www.nrc.gov/docs/ML1905/ML19058A013.pdf

90. NEI, *Industry Guidance for Compliance with Order EA-13-109*, NEI 13-02, Revision 1 (2015).

91. U.S. NRC, "Issuance of Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions", Order EA-13-109 (2013).

92. U.S. NRC, 10 CFR 50.155, "Mitigation of Beyond-Design-Basis-Events", https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0155.html, last accessed December 2021.

93. U.S. NRC, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events", EA-12-049 (2012).

94. U.S. NRC, "Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation", EA-12-051 (2012).

95. Institute of Nuclear Power Operations (INPO), INPO Event Report (IER) L1- 11-2 Revision 2, "Fukushima Daiichi Nuclear Station Spent Fuel Pool Loss of Cooling and Makeup", dated October 28, 2014.

96. Institute of Nuclear Power Operations (INPO), INPO Event Report (IER) L1- 13-10, "Nuclear Accident at the Fukushima Daiichi Nuclear Power Station", dated March 28, 2013

97. NEI, *Guideline for Assessing Beyond-Design-Bases Accident Response Staffing and Communication Capabilities,* NEI 12-01, Revision 0, dated May 2012 (ML12125A412/ML12131A043).

98. U.S. NRC, "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3 and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident", 2012 (ML12053A340); see https://www.nrc.gov/docs/ML1205/ML12053A340.pdf

99. EPRI Nuclear Maintenance Applications Center Program (NMAC), "FLEX Equipment collaboration Site" see https://flex.epri.com; requires EPRI account to access.

100. NEI "Efficiency Bulletin 16-17 Revision 1, Optimizing FLEX Equipment Preventive Maintenance Strategies and Sharing of Operational Experience", dated January 2021.

101. B. Williamson, R. Harter, T. Brandt, K. Steimer, D. Curtand, and P. Ellison, "Duane Arnold Energy Center (DAEC) Loss of Offsite Power (LOOP) Due to Derecho", BWROG - Emergency Procedures Committee (EPC), June 2021.

102. D. Curtlant, Duane Arnold Energy Center, "Notice of Unusual Event and Unit Trip Due to Loss of Offsite Power Due to High Winds", Licensee Event Report 2020-001-01, September 30, 2020, https://www.nrc.gov, Adams Accession No. ML20283A373.

103. B. C. Dickson, US NRC, letter to D. Curtlant, DAEC, "Duane Arnold Energy Center Integrated Inspection Report 05000331/2020003 and 07200032/2020001", November 6, 2020, https://www.nrc.gov, Adams Accession No. ML20314A150.

104. MHLW, "Good Practices in Radiation Exposure Dose Reduction Measures", Commissioned by the Ministry of Health, Labour and Welfare, FY2018 Project to Enhance the Radiation Exposure Dose Reduction Measures for works Relating to the Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Plant", 2019, https://www.mhlw.go.jp/english/topics/2011eq/workers/index.html, last accessed February 2022.

105. MHLW, "FY2019 Project to Enhance the Radiation Exposure Dose Reduction Measures for Works Relating to the Decommissioning - Practices in Radiation Exposure Dose Reduction", Commissioned by the Ministry of Health, Labour and Welfare, August 2020, https://www.mhlw.go.jp/english/topics/2011eq/workers/index.html, last accessed February 2022.

106. MHLW, "Good Practices in Radiation Exposure Dose Reduction Measures", Commissioned by the Ministry of Health, Labour and Welfare, March 2021, https://www.mhlw.go.jp/english/topics/2011eq/workers/index.html, last accessed February 2022.

107. OECD, "Occupational Radiation Protection in Severe Accident Management", OECD Report NEA/CRPPH/R(2014)5, January 2015.

108. OECD, "Long-Term Management and Actions for a Severe Accident in a Nuclear Power Plant- Status Report", OECD/NEA Status Report, Nuclear Safety 2021.

109. IAEA, "Final Report of the International Mission on Remediation of Large Contaminated Areas Off-site the FDNPP", 7-15 Oct 2011, Report NE/NEFW/2011, November 11, 2011.

110. IAEA, "Final Report: The Follow-up IAEA International Mission on Remediation of Large Contaminated Areas Off-Site the Fukushima Daiichi Nuclear Power Plant Tokyo and Fukushima Prefecture", Japan 14 - 21 October 2013, IAEA Report NE/NEFW/2013, January 23, 2014.

111. JAEA, Fukushima Environmental Safety Center, "Remediation of Contaminated Areas in the Aftermath of the Accident at the Fukushima Daiichi Nuclear Power Station: Overview, Analysis and Lessons Learned Part 1: A Report on the "Decontamination Pilot Project", JAEA-Review 2014-051, March 2015.

112. JAEA, Fukushima Environmental Safety Center, "Remediation of Contaminated Areas in the Aftermath of the Accident at the Fukushima Daiichi Nuclear Power Station: Overview, Analysis and Lessons Learned Part 2: Recent Developments, Supporting R&D and International Discussions", JAEA-Review 2014-052, March 2015.

113. K. Miyahara, I. McKinley, K. Saito, S. Hardie and K. Iijima, "Use of Knowledge and Experience Gained from the Fukushima Daiichi Nuclear Power Station Accident to Establish the Technical Basis for Strategic Off-site Response", Fukushima Environmental Safety Center, JAEA-Review 2015-001, March 2015.

114. EPA, *Plastic Scintillation Fibers for Radiological Contamination Surveys*, EPA/600/R-17/370, September 2017.

115. JAEA, "Real-time Monitoring Technology of β-ray emitted radionuclide (90Sr-90Y) in Water has been Successfully Developed", January 30, 2020 https://www.jaea.go.jp/english/news/press/2020/013001/, last accessed July 28, 2021.

116. T. Washiya, M. Kurata, S. Koyama, and A. Nakayoshi, "JAEA Update and Discussion", presentation at 2020 Reactor Safety Technology Expert Panel Meeting, November 10-12, 2020 [Appendix C of Reference [2].

117. MoE, “FY2014 Decontamination Report - A compilation of experiences to date on decontamination for the living environment conducted by the Ministry of the Environment - (Tentative Translation)”, March 2015, http://josen.env.go.jp/en/, last accessed February 2022.

118. MoE, “Decontamination Projects for Radioactive Contamination Discharged by Tokyo Electric Power Company Fukushima Daiichi Nuclear Power Station Accident”, March 2018, (Tentative Translation), February 2019, http://josen.env.go.jp/en/, last accessed February 2022.

119. Y. Sato, et al., “Remote Detection of Radioactive Hotspot using a Compton Camera Mounted on a Moving Multi-copter drone above a Contaminated Area in Fukusihma”, J*ournal of Nuclear Science and Technology*, 57:6 734-744, http://doi.org/10.1080/00223131.2020.1720845 (2020)

120. Y. Sato, Y. Terasaka, W. Utsugi, H. Kikuchi, H. Kiyooka, and T Torii, “Radiation imaging using a compact Compton camera mounted on a crawler robot inside reactor buildings of Fukushima Daiichi Nuclear Power Station”, *Journal of Nuclear Science and Technology*, 56:9-10, 801-808, (2019) DOI: 10.1080/00223131.2019.1581111, https://doi.org/10.1080/00223131.2019.1581111

121. NRA, “Investigations inside the Unit 2 Turbine Building and the Unit 4 Reactor Building”, *NRA Meeting on 1F Accident Analysis*, https://www.nsr.go.jp/data/000358696.pdf, July 8, 2021.

122. N. Ishii, et al., “I&C Equipment Actions in Decommissioning Work after Fukushima Daiichi Nuclear Accident”, Invited Plenary Presentation, SAMMI-2020, December 7, 2020.

123. MoE, “Decontamination Guidelines 2nd Ed., 2013 (Tentative Translation) Part 1 - Guidelines on the Methods for Investigation and Measurement of the Status of Environmental Pollution in Intensive Contamination Survey Areas”, 2nd Ed., 2013, http://josen.env.go.jp/en/, last accessed February 2022.

124. RAND, “Technological Lessons from the Fukushima Dai-Ichi Accident”, https://www.rand.org/pubs/research_reports/RR857.html, 2016.

125. M. Sasaki, Y. Sanada, E. W. Katengeza, and A. Yamamoto, “New Method for Visualizing the dose rate distribution around the Fukushima Daiichi Nuclear Plant using Artificial Neural Networks”, https://doi.org/10.1038/s41598-021-81546-4, *Nature Research*, November 2021.

APPENDIX A. FY2022 Meeting Agenda and Attendee List

A.1. November 28-30, 2021 Meeting Agenda

Reactor Safety Technology Expert Panel Forensics Meeting

Meeting Agenda
November 28-30, 2021

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

Session 1 - November 28, 2021 Washington DC/November 29, 2021 Tokyo, Japan

Virtual Meeting (WebEx: Must Respond to Registration Email)
Time relative to Start: 6:00 pm EST/8:00 am JST

00:00	Welcome and FY22 Meeting Agenda	J. Rempe, Rempe and Associates, LLC
00:05	Strategic Plan 2021	T. Ito, NDF
00:20	NRA – Japan Activities -	M. Yasui, JNRA
01:35	*Break*	*All*
01:45	TEPCO Update and Discussion - Future Plans presented at IAEA Review (Decommissioning KM) - Summary of Visual information inside PCV related to MCCI, Peak Temperature, Degradation of Organic Cables, etc. - 1F2 Reactor Well Investigations and Plan for 1F Investigations - Long Term Cooling Insights – Flowmeter Importance	TEPCO Holdings S. Mizokami, K. Owada M. Cibula S. Mizokami
03:00	JAEA Update and Discussion - Updates to JAEA report (JAEA-Review-2020-055) - Update on 1F sample analysis - integrated Radiation Imaging System (iRIS) based on Compton Camera	JAEA T. Washiya, S. Koyama H. Ikeuchi Y. Sato
04:00	*Adjourn*	*All*

Reactor Safety Technology Expert Panel Forensics Meeting

Meeting Agenda
November 28-30, 2021

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

Session 2 - November 29, 2021 Washington DC/November 30, 2021 Tokyo, Japan

Hybrid Meeting (TEAMS: Must Respond to Registration Email)
Time relative to Start: 10:30 am EST/12:30 am JST

00:00	NEI Welcome and Administrative Comments	J. Butler, NEI
00:10	Welcome and Overview of DOE Activities - Overview of US DOE-sponsored Forensics Activities (US and International) including Report on New Remediation Technologies	D. Peko, DOE-NE
00:25	Welcome and Update on Related NRC Activities	H. Esmaili, US NRC
00:40	FY21 Findings and Recommendations; - FY22 Meeting Agenda - Thoughts about OECD and IAEA Lessons Learned Reports	J. Rempe, Rempe and Associates, LLC
01:00	Related EPRI Activities	M. Nudi, EPRI
01:45	*Break (In-Person Attendees Obtain Food for Working Luncheon)*	*All*
02:15	Recent MELCOR Development Activities	D. Luxat, SNL
03:00	*Break*	*All*
03:15	Topic 1 – Component and System Performance	J. Gabor, Jensen-Hughes/ K. Robb, ORNL
04:00	Topic 2 – Radiation Surveys and Sampling	D. Luxat SNL
04:45	Topic 3 – Core Debris Location Evaluations Recent Insights regarding Debris Endstate and Coolability Passive Interim Storage	M. Farmer ANL/ M. Plys, FAI
05:30	*Adjourn*	*All*

Reactor Safety Technology Expert Panel Forensics Meeting

Meeting Agenda
November 28-30, 2021

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

<u>Session 3 - November 30, 2021 Washington DC/November 30, 2021 Tokyo, Japan</u>

Hybrid Meeting (TEAMS: Must Respond to Registration Email)
Time relative to Start: 8:00 am EST/10:00 pm JST

00:00	Topic 4 – Combustible Gas Effects -Comments on recent NRAJ Combustible Gas Investigations -Comments on Proposed and Prior Experimental Investigations	W. Luangdilok, H2 Technology, LLC
00:45	Topic 5 – Operations and Maintenance - BWR EPG SAG Overview - Terry Turbopump Testing 2021 Update - Radiation Best Practices – Lesson Learned from 1F D&D	R. Bunt and P. Ellison, BWROG
01:30	*Break*	*All*
01:45	Update to Consensus Information Requests	*All*
03:30	Next Steps ▪ Proposed letter report(s) ▪ Action items and schedule	J. Rempe, Rempe and Associates, LLC
04:00	*Adjourn*	*All*

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

Name	Organization
Lucas Albright	Sandia National Laboratories
Don Algama	U.S. Department of Energy
Phillip Amway	Exelon, BWR Owners Group
Lake Barrett	L. Barrett Consulting LLC
Sudhamay Basu	McGill Engineering Associates
Frances Bolger	Electric Power Research Institute
Randolph Bunt	Southern Nuclear Company, BWR Owners Group
John Butler	Nuclear Energy Institute (NEI)
Shawn Campbell	U.S. Nuclear Regulatory Commission
Alice Chung	U.S. Nuclear Regulatory Commission
Michal Cibula	TEPCO Holdings
Michael L. Corradini	University of Wisconsin-Madison
Paul Dickman	Argonne National Laboratory
Phillip G. Ellison	GE-Hitachi, BWR Owners Group
Hossein Esmaili	U.S. Nuclear Regulatory Commission
Mitchell T. Farmer	Argonne National Laboratory
Terri V. Farthing	GE Hitachi
Jeffrey R. Gabor	Jensen Hughes
Randall O. Gauntt	Gauntt Technical Safety Associates, LLC
Kazuki Hida	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Taro Hokugo	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Takeshi Honda	TEPCO Holdings
Harutaka Hoshi	Japan Nuclear Regulation Authority
Hirotomo Ikeuchi	Japan Atomic Energy Agency
Shungo Ito	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Takatsune Ito	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Kohei Iwanaga	Japan Nuclear Regulation Authority
Karen Kirkland	Texas A&M University
Wataru Kikuchi	Japan Nuclear Regulation Authority
Kenneth Klass	BWR Owners Group (Consultant)
Tatsuro Kobayashi	TEPCO Holdings
Retsu Kojo	Japan Nuclear Regulation Authority
Shinichi Koyama	Japan Atomic Energy Agency
Steven Kraft	Kraft-Contente, LLC
Masaki Kurata	Japan Atomic Energy Agency
Richard Lee	U.S. Nuclear Regulatory Commission (retired)
Roy Linthicum	PWR Owners Group

Name	Organization
Wison Luangdilok	H2 Technology, LLC
David Luxat	Sandia National Laboratories
Donald Marksberry	U.S. Nuclear Regulatory Commission
Robert Martin	BWX Technologies
Paul McMinn	Fauske and Associates, LLC
Masato Mizokami	TEPCO Holdings
Shinya Mizokami	TEPCO Holdings JAEA (CLADS)
Junichi Nakano	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Mariko Nishizawa	Member, TEPOCO Holdings Nuclear Reform Committee
Kenji Noshita	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Matt Nudi	Electric Power Research Institute
Shuichi Ohashi	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Tatsuo Okamuro	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Yoshimi Ota	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Kenji Owada	TEPCO Holdings
Chan Paik	Fauske and Associates, LLC
Damian Peko	U.S. Department of Energy
Marty Plys	Fauske and Associates, LLC
Joy Rempe	Rempe and Associates, LLC
Kevin Robb	Oak Ridge National Laboratory
Yuki Sato	Japan Atomic Energy Agency
Mike Salay	U.S. Nuclear Regulatory Commission
David Skeen	U.S. Nuclear Regulatory Commission
Yoshitaka Suzuki	Chubu Electric Power Company, Inc.
Masami Taira	Japan Nuclear Regulation Authority
Hirokazu Tanoue	Japan Nuclear Regulation Authority
Nozomu Toyoda	Chubu Electric Power Company, Inc.
Haruka Usuda	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
Richard Wachowiak	Jensen Hughes
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
Masaya Yasui	Japan Nuclear Regulation Authority

APPENDIX B. Information Requests

As described in Section 1.1, a primary objective of the U.S. forensics effort is to provide consensus U.S. input for high priority time-sequenced examination tasks and supporting research activities that can be completed with minimal disruption of D&D plans for Daiichi. Initial information requests were developed in FY2015. Every year, these information requests are reviewed and as appropriate, updated. As discussed in Section 2.1.3.4 (see Appendix C.1.3.4 slides), TEPCO plans for future investigations consider U.S. information requests. In addition, as Japan moves forward with the FACE project, it is important that the benefit of information requests be documented (see Section 2.1.2 and Appendix C.1.2 slides).

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

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

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

- **RB-9b:** Photos / videos of damaged walls and structures (1F3).
- **RB-10:** Photos / videos and dose surveys of 1F1 (vacuum breaker), 1F1, 1F2, and 1F3 PCV leakage points (bellows, penetrations).
- **RB-15:** Examinations (water level and additional dose surveys) of 1F1 RCW surge tank and evaluations of RCW water samples
- **PC-1:** Photos/ videos of drywell head, head seals, and sealing surfaces (1F1, 1F2, and 1F3). Procedures used to tension and torque the bolts used to close the drywell head bolts.
- **PC-3a:** Photos/ videos of relocated debris and crust, debris and crust extraction, hot cell exams, and possible subsequent testing (1F1 - 1F3)
- **PC-3b:** PCV liner examinations of debris (photos/videos and metallurgical exams; 1F1-1F3)
- **PC-3c:** Photos/ video, RN surveys, and sampling of debris and water samples near the pedestal wall and floor (1F1-1F3).
- **PC-3d:** Concrete erosion profile; photos/videos and sample removal and examination (1F1-1F3)
- **PC-3e:** Photos / videos of RPV lower head and of structures and penetrations beneath the vessel to determine damage and corium hang-up (1F1-1F3).
- **PC-5:** Photos/videos of 1F1, 1F2, and 1F3 main steam lines and Automatic Depressurization System (ADS) lines to end of SRV tailpipes, including instrument lines.

- **PC-6:** Visual inspections of 1F1, 1F2, and 1F3 SRVs and Main Steam Lines (MSLs) including standpipes (interior valve mechanisms).
- **PC-17:*** Chemical and isotopic analysis of the upper layer of sediment on drywell floor at the X-100B penetration location in 1F1. Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes.
- **PC-18:*** Evaluate nature of material below the sediment at the 1F1 X-100B penetration location to determine if fuel debris is present. Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes
- **PC-19:***Chemical analysis (using X-Ray Florescence or XRF) of black material discovered on CRD exchange rail in 1F2 at X-6 penetration location. This item has been completed, so it is now shaded gray rather than purple.
- **PC-20:*** Chemical analysis of black material on 'existing structure' in 1F1 images at location 'D3'.
- **PC-21:** Images from examinations in 1F3 X-53 penetration
- **PC-22:*** Chemical analysis of debris from locations at different axial and radial positions (bores, if possible). Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes (1F1-1F3)
- **RPV-1b:** Photos/videos, probe inspections of 1F1, 1F2, and 1F3 MSLs; interior examinations of MSLs at external locations. If significant distortion observed, then metallurgical exams of samples would be of interest for D&D
- **RPV-4:**† Remote mapping of 1F1, 1F2, and 1F3 core through shroud wall from annular gap region (muon tomography and other methods, as needed).
- **RPV-5:**† Mapping of end state of core and structural material (visual, sampling, hot cell exams, etc.).

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

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

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

B.1. Summary Information Requests

Table B-1. Information requests for the reactor building

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

Table B-1. Information requests for the reactor building

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

Table B-1. Information requests for the reactor building

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

Table B-1. Information requests for the reactor building

Item	What/How Obtained	Why	Benefit /Use	When	Status
RB-7	Isotopic evaluations of obtained concrete samples (1F2)	• Code assessments. • Possible model improvements for building retention assumptions.	Understanding what happened; assist D&D efforts. Improved BWR modeling and emergency planning; cross check of RN surveys. Potential PWR impacts (e.g., modeling, AM strategies, etc.).	Completed.	JAEA has obtained surface RN concentrations and RN distribution from boring concrete samples. Surface radionuclide concentrations and distribution of radionuclides of boring core samples were obtained. If additional samples or surveys are obtained, isotopic composition is of interest (but the U.S. recognizes that additional information may not be obtained).
RB-8	Photos/videos and inspection of seismic susceptible or radiation degraded components and structures (e.g., bellows, penetrations, welds, structures, supports, etc. in 1F1, 1F2, 1F3, and 1F4)	• To confirm with data that there were no seismic-induced failures • To determine with data if there are any radiation-degraded components and concrete structures; • To develop lessons learned regarding their performance under high radiation conditions	Understanding what happened; assist D&D efforts. Improved plant robustness; observed differences between 1F1 and 1F3. Potential PWR impacts (e.g., similar penetrations, structures, and components). Additional seismic data for large magnitude earthquakes that is specific to nuclear related components and systems is of interest for operating and new reactors. It may be possible to use results to discern differences between challenges from H_2 explosions and seismic events.	Now and later (as debris is removed); Note that debris currently precludes data from being obtained. The February 2021 seismic event may increase priority of such examinations. New remote technologies may facilitate such inspections.	Images obtained by TEPCO Holdings have been archived per request of NRAJ for Unit 4 (see NRAJ website). TEPCO has published report on Units 5 and 6 and on Daini. TEPCO Holdings will review and provide additional images of interest. **1F1:** The IC main unit, major pipes, and major valves visually investigated to confirm whether there was any damage that could cause reactor to lose coolant. Since inside area of PCV inaccessible, IC, pipes, and valves outside PCV checked. **1F2:** No large abnormality was found in the robot camera's visual inspection. Visual inspection inside PCV performed in 1F1, 1F2, and 1F3 but inspection range limited. If additional information is obtained as part of planned D&D activities, please provide (but the U.S. recognizes that additional information may not be obtained).

Table B-1. Information requests for the reactor building

Item	What/How Obtained	Why	Benefit /Use	When	Status
RB-9	a) DW Concrete Shield 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.	Now and later (as debris is removed).	Additional RN surveys obtained by TEPCO Holdings and NRAJ being shared. NRAJ conducting additional surveys between upper and middle layers of the shield plug to quantify remaining radionuclides (e.g., Cs-137) within the shield plugs.
	b) Photos/ videos and dose surveys around mechanical seals and hatches and electrical penetration seals (as a means to classify if joints in compression or tension)	• Potential leakage paths for RN and hydrogen release.[b] • To develop lessons learned regarding seal performance under high 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.	Now and later (as debris is removed).	Images and RN surveys obtained by TEPCO Holdings have been archived per request of NRAJ. If photos are obtained as part of NRAJ investigations or other planned D&D activities, please provide (but the U.S. recognizes that additional information may not be obtained).
RB-10	Photos/videos and dose surveys of 1F1 (vacuum breaker), 1F1, 1F2, and 1F3 PCV leakage points (bellows, penetrations)	• Potential leakage paths for RN and hydrogen release. • To develop lessons learned regarding penetration performance under high radiation/high temperature conditions	Improved AM strategies (Plant improvements for more robustness, training, education); applicable to BWRs and PWRs (which have similar penetration designs). Improved codes. Improved understanding of events; assist D&D efforts.	Now and later.	Images and RN surveys obtained by TEPCO Holdings have been archived per request of NRAJ and TEPCO Holdings provided additional information on 1F1. As additional testing is completed, the US would appreciate it.[c] Now, restoring works for PCV to stop water leakage are higher priority, and there is no plan to scrutinize the damaged area or degree of PCV. If additional photos or information is obtained, please provide (but the U.S. recognizes that additional information may not be obtained).

Table B-1. Information requests for the reactor building

Item	What/How Obtained	Why	Benefit /Use	When	Status
RB-11	Photos/ videos and dose information on 1F1, 1F2, 1F3, and 1F4 containment hardpipe venting pathway, SGTS and associated reactor building ventilation system	• To assess performance of SGTS under high temperature and radiation conditions.[d] • To develop lessons learned regarding their performance under high radiation/high temperature conditions • Accident progression benchmarks.	Improved AM strategies (Plant improvements). Improved understanding of events, assist D&D efforts.	Initial request completed; additional information obtained as part of D&D.	**1F1:** Dose rate of venting pathway and the point in front of SGTS room. Because of high dose rate, access to SGTS room is difficult. TEPCO conducted a robotic investigation of 1F1/1F2 SGTS room during 2021. 1F2 and 1F3: Photos and dose rate of SGTS trains and venting pathway available. NRAJ has obtained additional information using new technologies (e.g., gamma cameras). 1F1 and 1F2 samples will be stored for possible future analysis.
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.	Initial request completed; additional information may be available due to new technology.	This item has been addressed. No additional activities currently considered by TEPCO Holdings. If additional photos are obtained as part of planned D&D activities, please provide (but the U.S. recognize that additional information may not be obtained). NRAJ has and is sharing additional information using new technologies (e.g., gamma cameras).

Table B-1. Information requests for the reactor building

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

Table B-1. Information requests for the reactor building

Item	What/How Obtained	Why	Benefit /Use	When	Status
RB-15	Examinations (water level and additional dose 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.	Now.	RN surveys obtained by TEPCO Holdings have been archived per request of NRAJ, and are being shared by NRAJ. TEPCO Holdings will be performing investigations as part of decommissioning work in CY2022.

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

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

c. **1F1:** Water leaks from a sand cushion drain pipe and an expansion joint (bellows) for vacuum breaker tube observed. The water leak from a sand cushion drain pipe was confirmed since the vinyl chloride pipe (connecting the sand cushion drain tube and drain funnel with an insertion-type joint) had been displaced. Water leaks could not be confirmed at other seven drain pipes, since the drain tubes had not been displaced. However, concrete seams (joints) below sand cushion drain piping were observed to be wet all around on the concrete wall, which indicates that leaked water is filled in the sand cushion area outside of PCV wall. The water leak from bellows of vacuum breaker tube is located in the direction of access opening of pedestal wall in the PCV floor where molten corium might spread out first.
1F2: It was confirmed SC water level changes together with torus room water level. This indicates water is leaking from the lower position of SC including suction piping. No water leakage from sand cushion drain pipes or vent pipe was observed. As of now, water leakage is not specified.
1F3: Water leak from near the expansion joint (bellows) of main steam line D in MSIV room was confirmed. The water level in the PCV is estimated at about 2 m above the reactor building first floor by converting the S/C pressure obtained by the existing pressure indicators to water head. This elevation is on the level of PCV penetrations for main steam lines, thus indicating the possibility of water leaks from the PCV penetration of MSL.
1F3: Water seeping from equipment hatch is inferred from the following observations.
- Rust was observed along with the hatch interface lower than DW water level (in November 2015). Upper part of the interface does not have the rust.
-The increasing dose rate on the floor towards the equipment hatch was observed (in November 2015), which indicates contaminated water had flown from DW side
- Equipment hatch rail was dry in December 2015. Current DW water level is lowest since 2011. The DW water level in 2011 was higher and water seeping from DW through equipment hatch seal would be higher.
- The observed high dose rate at the rail in front of shield plug for equipment hatch (in September 2011) would be attributed to water leak through equipment hatch seal.
- Water dripping due to rain fall observed (in November 2015, rainy day), which might be intruding from refueling floor.
No specific observation regarding gas phase leakage other than dose rate distribution on refueling floor and steam discharging from refueling floor.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

B.2. Additional details for Information Requests

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

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

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

• **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
RB-15: 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, Reactor Building):**
PC-1: Photos/ videos of drywell head, head seals, and sealing surfaces (1F1, 1F2, and 1F3). Procedures used to tension and torque the bolts used to close the drywell head bolts. This information is of interest both prior to event and during debris removal. • Visual - signs of asymmetric lift or leakage paths. Look for thermal deformation due to high temperatures/high radiation conditions over time. • RN Swabbing • Visual inspection of seal • Visual inspection of the head. Look for evidence of permanent strain in the head flange or bulging of the head hemisphere and for evidence of bending/bowing of the bolts along their length that could result from head flange strain and result in permanent leakage location even after PCV decompression. • Inspect shield plug - visual inspection of cracks. 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, Reactor Building):**
PC-3a: Photos/videos of debris and crust, debris and crust extraction, possible hot cell exams, and possible subsequent testing (1F1, 1F2, and/or 1F3) High-resolution images (photos/videos) of debris and crust both in the as-found state and during extraction, and chemical analysis to determine composition and oxidation state. Imaging should be sufficient to provide insights into material characteristics (i.e., particle bed versus crust material, and if crust material, the morphology and extent of cracking if possible). A sufficient number of samples should be selected to estimate the spatial variations in composition. Elemental analysis of samples should look for fuel, structural, and concrete components. Evaluations should determine the approximate proportions of Uranium/Zirconium/Stainless Steel/Boron from the drive tubes below the vessel head and the corium samples retrieved from the cavity region. In addition, samples from the cavity region should be analyzed for the presence of Al/Ca/Si/Mg that would provide evidence of MCCI.
• **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
Structural characteristics of the material are important for supporting tooling design for removal; chemical analysis important for criticality evaluations. These same data are important for improving reactor safety analysis models and accident management.
• **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
Benchmark and reduce uncertainty in models for predicting molten core concrete interaction (MCCI) phenomena. MCCI phenomena are important for assessing combustible gas generation during late phase accident progression, as well as the extent of attack on containment structures. It is important to reduce uncertainty in this phenomenon because it affects strategies for venting and water addition. Improved knowledge will be used to enhance accident management strategies.
• **Methods/Tools Needed to Collect Information or Data:**
• Irradiation resistant high-resolution imaging system • Hot cell elemental analysis system, and/or in-situ elemental analysis using Laser Induced Breakdown Spectroscopy (LIBS) and/or X-ray Florescence • Ultimately, D&D cutting and removal tools able to extract materials
• **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later (Sample removal possible within next 2 years).
• **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
Obtaining /using this information may require additional material property and coolability testing (Young's modulus, linear expansion, ultimate strength, hardness, tensile strength, etc.) for cutting tool development and for model development. Evaluation of this information may require composition information for concrete (to distinguish between sand and concrete).

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

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

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

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

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

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

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

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

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

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

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

• **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
PC-6: Visual inspections of 1F1, 1F2, and 1F3 SRVs including standpipes in the torus and drywell (interior valve mechanisms)
• **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
BWR AM Strategies (maintenance practices, etc.), SRV functioning in test facility data, and better simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.).
• **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
To determine if there was any failure of SRVs and associated piping.
• **Methods/Tools Needed to Collect Information or Data:**
• Irradiation resistant high-resolution imaging system (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, Reactor Building):**
PC-17: Chemical analysis of upper layer of sediment on drywell floor at the X-100B penetration location in 1F1. The upper surface of the sediment is ~ 30 cm above drywell floor. Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes. **PC-18:** Evaluate nature of material below the sediment at the 1F1 X-100B penetration location to determine if fuel debris is present.[a] Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes. **PC-19:** Chemical analysis (XRF) of black material discovered on CRD exchange rail in 1F2 at X-6 penetration location **PC-20:** Chemical analysis of black material on 'existing vertical wall structure' in 1F1 picture outside pedestal doorway **PC-22**: Chemical analysis of debris from locations at different axial and radial positions (bores, if possible). Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes. Include neutron and gamma detectors in examinations. Evaluations of bore samples indicating axial composition, including identification of short-lived isotopes. (1F1-1F3). These five information requests focus on the chemical composition of materials observed in 1F1 (i.e., sediment and underlying material on the drywell floor below the X-100b penetration, and on existing vertical structure near the pedestal doorway), and black material discovered on the CRD exchange rail in 1F2 from the X-6 penetration. Elemental analysis of samples should look for fuel, structural, and concrete components and should also include a measurement of oxygen content if possible. Evaluations should also consider data to address recriticality concerns and debris cutting, drying, and storage requirements [e.g., debris composition and morphology (e.g., crack, gaps, porosity, permeability, particle size and shape distribution)].
• **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 molten core concrete interaction (MCCI) phenomena. Additional information is also needed to benchmark and reduce uncertainty in models for predicting vessel failure, in-vessel cladding oxidation and hydrogen production, holdup on ex-vessel structures, and MCCI phenomena. Vessel failure, holdup on ex-vessel structures, and MCCI phenomena are important for assessing combustible gas generation during late phase accident progression. It is important to reduce uncertainty in these phenomena because they affect strategies for venting and water addition. Additional PC-19 analysis can be used to assess the extent of in-vessel cladding oxidation. PC-18 evaluations can be used to determine if core debris is present at the X-100B location, thereby providing insights on the extent of core debris relocation which is also a critical uncertainty impacting accident management strategies. Knowledge gained from these analyses will be used to enhance these strategies. Data from PC-17 can be used to determine if sediment composition varies with height. Recent chemical analysis results indicate a high presence of Na but little Cl, indicating the potential for NaCl decomposition and potential formation of CsCl which could impact source term evaluations.
• **Methods/Tools Needed to Collect Information or Data:**
• Hot cell elemental analysis system and/or in-situ elemental analysis using Laser Induced Breakdown Spectroscopy (LIBS) and/or X-ray Florescence (XRF). • Robotics systems for collecting samples, and for probing / determining the sediment (loose material) depth at X-100B. • Sample examinations should also consider identifying short-lived fission product isotopes. If present, this would indicate low-level recriticality and thus impact approaches for debris removal and storage. • In addition, future robot entries could be instrumented with a neutron detector (to augment gamma detector) that also provide of low-level criticality, if it is occurring. • Any low-level criticality could impact plant heat balance calculations, given the current low overall level of decay heat.
• **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.[79] 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.[80] 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.[80]

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

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

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

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

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

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

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

APPENDIX C. Selected FY22 Presentations

This appendix contains presentations from participants wishing to have them published in this report. Presentations are organized according to topics identified in the meeting agenda found in Appendix A.1. Session 1 presentations, which describe new information from Japan, are found in Appendix C.1; Sessions 2 and 3 presentations are organized according to topic: U.S. introductory remarks are found in Appendix C.2, recent U.S. systems analysis code development and application activities are found in Appendix C.3; and U.S. topic area presentations are found in Appendix C.4. Section 2 highlights key points discussed during these and other presentations during the meeting.

C.1. New Information from Japan

C.1.1. Nuclear Damage and Decommissioning Facilitation Corporation

- This slide is explanatory material. Please check the URL below for the full text and overview.
  - ◆Full text
    - https://www.dd.ndf.go.jp/files/user/pdf/en/strategic-plan/book/20211210_SP2021eFT.pdf
  - ◆Overview
    - https://www.dd.ndf.go.jp/files/user/pdf/en/strategic-plan/book/20211210_SP2021eOV.pdf

- You can download past Technical Strategic Plans from the NDF website.

NDF Website(English)
https://www.dd.ndf.go.jp/english/

NDF

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NDF Nuclear Damage Compensation and Decommissioning Facilitation Corporation

Technical Strategic Plan 2021 for Decommissioning of the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company Holdings, Inc.
(Explanatory Material)

October 29, 2021

Nuclear Damage Compensation and Decommissioning Facilitation Corporation

NDF

Table of Contents

1. Introduction

Four key aspects in Technical Strategic Plan 2021

- The Technical Strategic Plan 2021 presents mid-to-long-term technical strategy by focusing on the following four key aspects of this year.

First Proposed Prospects of a processing/disposal method and technology related to its safety

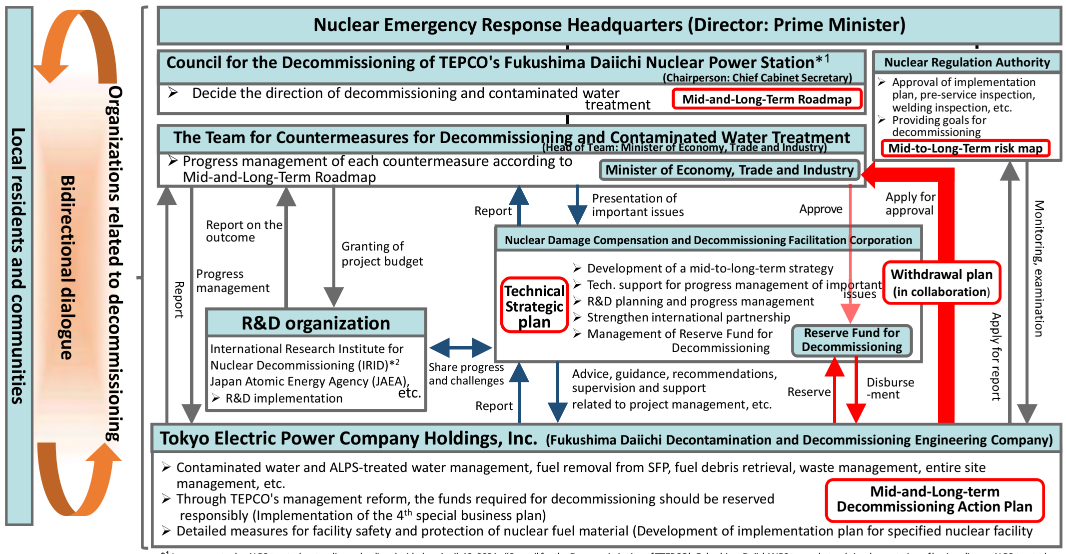
Second Issues to be addressed for the trial retrieval to minimize the impact of the COVID-19 infection

Third Summary of issues to be discussed for the selection of methods for further expansion of fuel debris retrieval

Fourth Efforts for the ALPS-treated water

1. Introduction

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

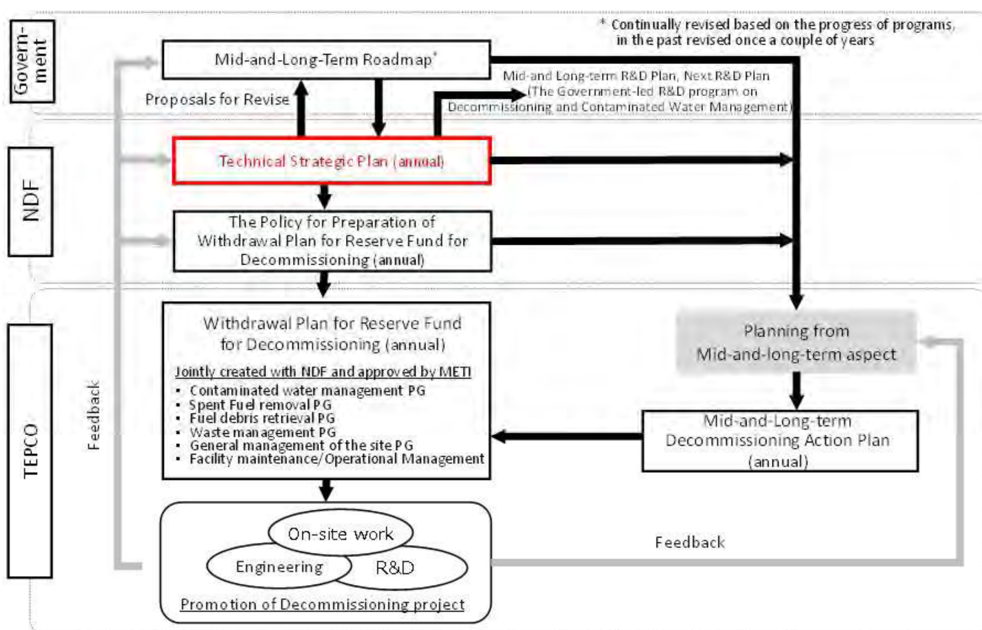


*1 In response to the ALPS-treated water disposal policy decided on April 13, 2021, "Council for the Decommissioning of TTEPCO's Fukushima Daiichi NPS toward steady implementation of basic policy on ALPS-treated water disposal" was founded.

*2 TEPCO, a decommissioning project operator, participates as a member of IRID and shares the needs, challenges, and results of research and development.

1. Introduction

Positioning of the Technical Strategic Plan



2. Concept on risk reduction and safety assurance for decommissioning of the Fukushima Daiichi NPS

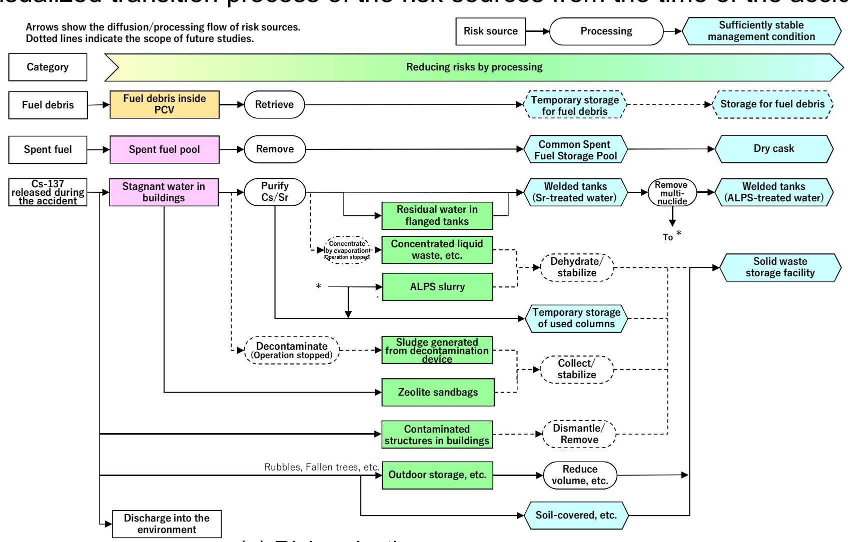
Concept on risk reduction

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

2. Concept on risk reduction and safety assurance for decommissioning of the Fukushima Daiichi NPS

Risk reduction process for major risk sources and its progress

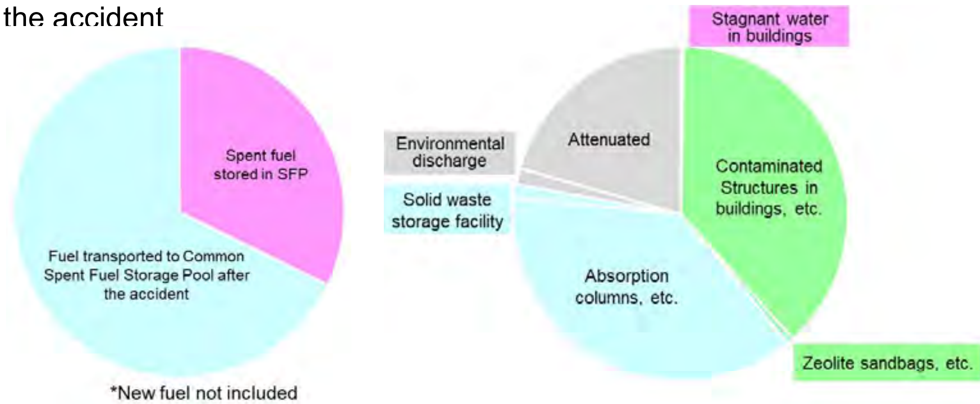
➢ Risk reduction process for major risk sources and an example of representing the decommissioning work progress
(Visualized transition process of the risk sources from the time of the accident)



(a) Risk reduction process

Risk reduction process for major risk sources and its progress

- Status of transition (in what proportion) to the "Sufficiently stable management" region for each risk source compared to the beginning of the accident



(b) Number of fuel assemblies of spent fuel (units 1 to 4)

(c) Radioactivity of Cs-137 released at the accident (units 1 to 3)

Fig. Risk reduction process for major risk sources and its progress
(example as of March 2021)

Approach to ensuring safety during decommissioning

Basic policy

- 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.

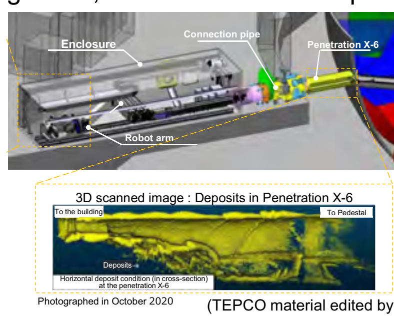
Major targets and progress for fuel debris retrieval

Major targets

- As for trial retrieval in unit 2, which is stated in Mid-and-long-term Roadmap as to be conducted in 2021, the process has been delayed due to the COVID-19 infection. In order to limit the delay to about one year, preparations will be made for starting retrieval.
- With regard to 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 internal investigations, research and development.

Progress

- The arm-type access equipment has arrived in Japan and started testing.
- Deposit contact investigation and 3D scanning investigation in the penetration X-6 was conducted.



(TEPCO material edited by NDF)

Fig. Investigation result of 3D scanning

Strategies for trial retrieval

Strategy Trial retrieval

- Although small in scale, the operation in which an opening will be newly provided to extend the containment barrier outside the PCV, is a fundamental form of site construction for future retrieval work, since the conventional containment barrier was located in the closed flange part (convex edge) of the penetration X-6. This presents an approach that enters a new stage.

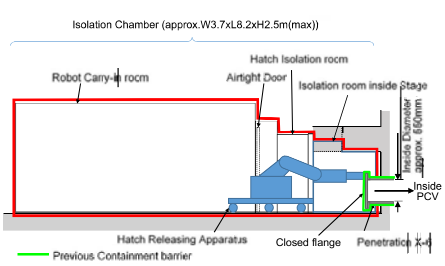


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Fig. Schematic drawing of isolation chamber to be installed at penetration X-6

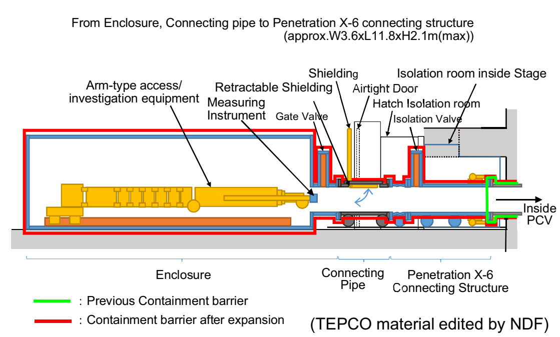

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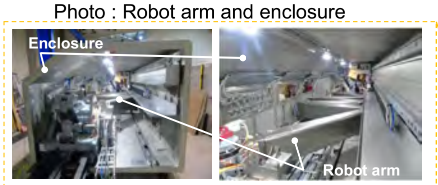
Fig. Schematic drawing of enclosure to penetration X-6

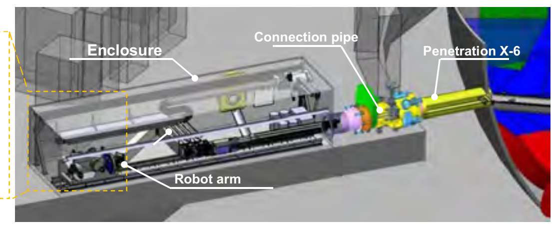
3. Technological strategies toward decommissioning of the Fukushima Daiichi NPS 3.1 Fuel debris retrieval

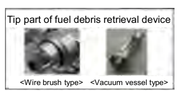
Strategies for trial retrieval

- While minimizing delays caused by the COVID-19 infection, mockup testing that takes full account of uncertainties on site is important in terms of actual site applicability and ensuring safety.
- It is necessary to maintain the backup system on the UK side, while sharing information and communicating smoothly with the UK engineers who fabricated the equipment.

Photo : Robot arm and enclosure







(TEPCO material edited by NDF)

Fig. Conceptual image of fuel debris retrieval system

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

Strategies for further expansion of fuel debris retrieval

Strategy **Further expansion of the retrieval scale**

① How to select retrieval methods

- In selecting the method, it is necessary to use evaluation items such as schedule and resources as indexes for selection while satisfying the target of safety level.
- In the process of selecting the method, it is most important to quantify each of these evaluation items, to use what evaluation items as indexes for selection, and how to set the weighting of these indexes.
- In a situation with many uncertainties, it is necessary to proceed with examination based on the currently available information and then to feed back the results gained from the investigation.

② Development of retrieval scenarios

- Several scenarios of fuel debris retrieval by each unit should be examined and clarified several paths. Then, it is important to narrow down the pathways thereafter to take according to the information obtained afterward.

Strategies for further expansion of fuel debris retrieval

③ Clarification of requirements

- Followings are important because operations, devices and equipment, and facilities will be larger, and the scope of construction will be wider.
  - ✓ Consideration in overviewing the entire Fukushima Daiichi NPS
  - ✓ Specification and optimization of the requirements (containment, criticality, operability, maintainability, throughput ※, etc.) for operations and devices

④ Process for narrowing down promising retrieval methods

- Diverse ideas are expected to be derived, but it is important to conduct objective evaluation and to narrow down the methods by the gradual process.

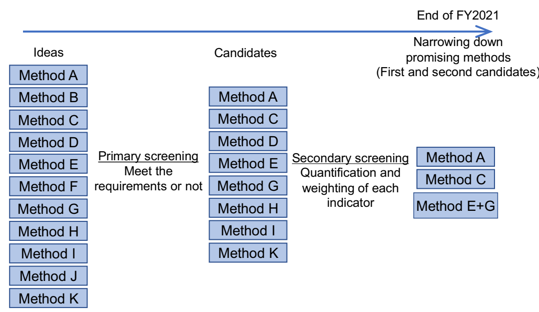


Fig. Image diagram of the process for narrowing down retrieval methods

※ Fuel debris retrieval capacity that indicates the processing time and operational efficiency

Major targets and progress for solid waste

Major targets

- As an effort towards implementing processing/disposal, prospects of a processing/disposal method and technology related to its safety should be made clear by around FY 2021.

Progress **Providing prospects of a processing/disposal method and technology related to its safety**

a. **Present measures toward reducing the volume** of solid waste

b. **Develop analytical/evaluation method** for efficient characterization

c. **Develop methods to reasonably select** safe **processing/disposal methods** at the time when the necessary information such as solid wastes' properties are proven

Prospects of a processing/disposal method and technology related to its safety

■ Present measures toward reducing the volume

- ➢ The priorities for measures to be taken as waste management are ① prevention of waste generation, ② minimization of waste volume, ③ reuse, ④ recycling. In waste management, it is important to consider ⑤ disposal as the last option for volume reduction of waste.

①: Consider in the design and construction plan to reduce the volume of materials to be used. Not to bring in substances that affect processing/disposal as much as possible

②: Strict segregation

③: Reuse should be promoted after contamination checks, decontamination, repair and parts replacement

④: Consider the contamination condition, separate and process recyclable materials, and use them as new materials and products

Fig. Summary of waste hierarchy at the NDA, UK, and countermeasures at the Fukushima Daiichi NPS

Prospects of a processing/disposal method and technology related to its safety

■ Develop analytical/evaluation method

- ➢ Automation of pretreatment and simplified analytical methods compared to the conventional radioactivity measuring method, etc.
- ➢ Establishing the method in which statistical methods have been applied to identify variable distribution and the width (quantify uncertainties in evaluated values).

■ Develop methods to reasonably select processing/disposal methods

- ➢ For the waste for which properties have been identified to some extent, repeating examination steps from ① to ③, and an appropriate combination of processing (waste form) and disposal※ methods would be examined,

① Establish several feasible disposal methods suitable for waste characteristics.

② Establish several processing methods suitable for waste characteristics to be considered and set the specifications of waste package after applying each processing method.

③ Evaluate the safety of several selected disposal methods based on the specifications of waste form after processing to verify whether risk to the public and environment can be sufficiently low, and to consider more effective processing/disposal methods based on the evaluation results.

※ The location and scale of the facility are not specified.

Prospects of a processing/disposal method and technology related to its safety

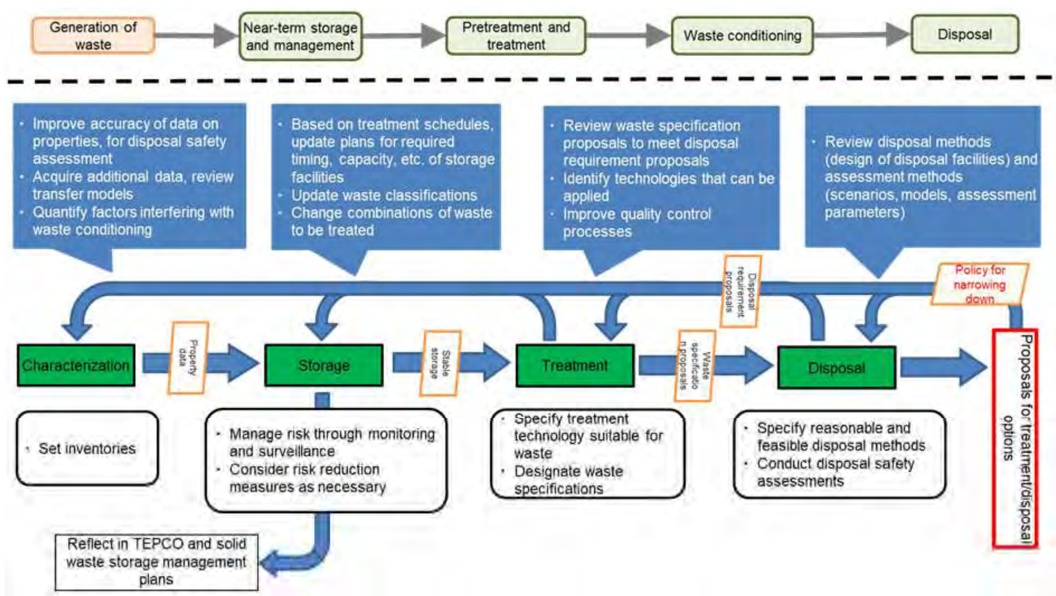


Fig. Develop methods to reasonably select safe processing/disposal methods

Technical strategy by sector related to waste management

Strategy

Characterization

➢ It is important to develop a medium-to-long-term analysis strategy that defines the solid waste to be analyzed, its priority, and quantitative targets for analysis, etc., and to proceed with analysis/evaluation accordingly.

Storage

➢ It is important to reconsider measurement items and timing, etc., while acquiring necessary information through continuous monitoring and surveillance of the storage status commensurate with the risks involved.

Processing/disposal

➢ In order to establish safe and reasonable processing/disposal methods, and to widely obtain knowledge for optimizing each individual stream※, it is necessary to continue development/research of processing/disposal technologies required for the series of studies.

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

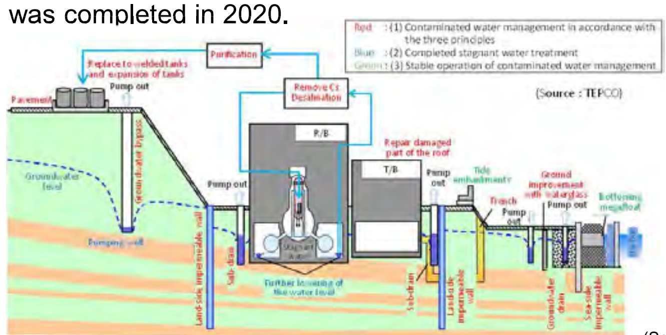
Major targets and progress for contaminated water management

Major targets

- To reduce the stagnant water in the reactor buildings in FY 2022 to FY 2024 to about the half of the amount of the end of 2020, while controlling the generation amount of the contaminated water to 100 m^3/day or less in 2025

Progress

- Excluding the reactor buildings of units 1 to 3, the process main building and high-temperature incinerator building, the treatment of stagnant water in buildings was completed in 2020.



(Source : TEPCO)

Fig. Outline of contaminated water management

Reducing stagnant water in reactor buildings

Strategy **Reduction of the stagnant water in the reactor buildings**

- In considering the removal methods for preventing the spread of α-nuclides, it is necessary to collect samples from as many places as possible and to understand the variation in their properties.

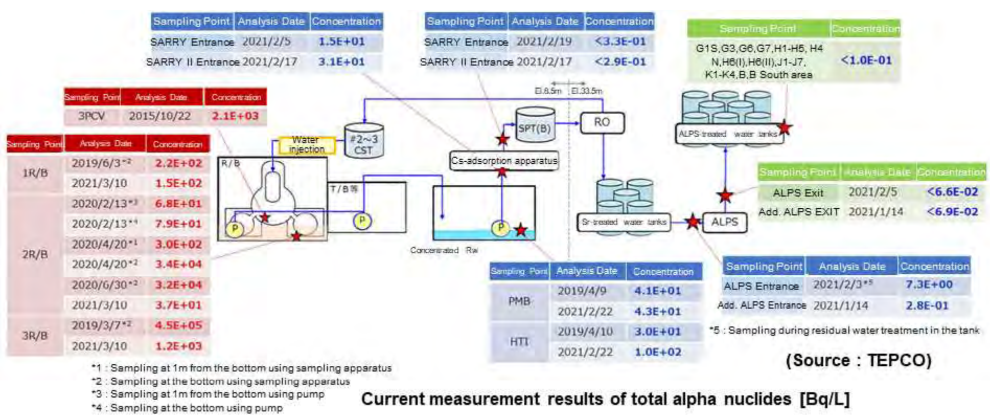


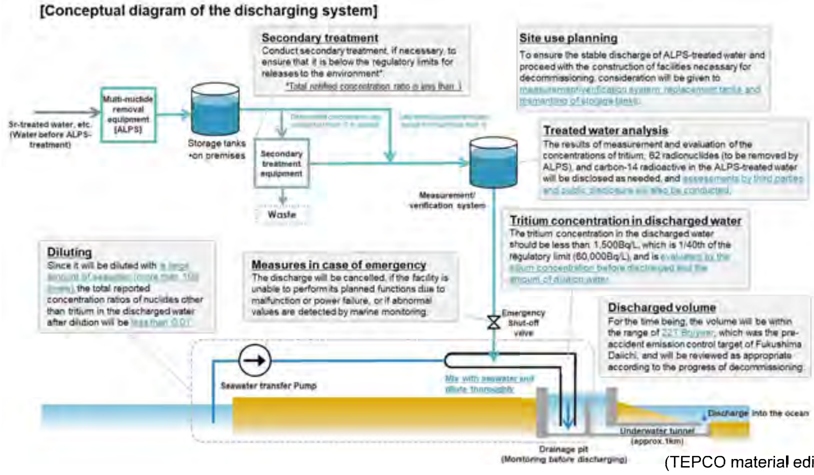
Fig. Water treatment systems for stagnant water in buildings and measurement results of total α-nuclides

Strategies for ALPS-treated water

Major targets

- ALPS-treated water currently being stored in tanks will be handled in accordance with the government’s basic policy decided in April 2021.

[Conceptual diagram of the discharging system]



(TEPCO material edited by NDF)

Fig. Conceptual diagram of ALPS-treated water discharge system planned by TEPCO

Efforts for releasing ALPS-treated water into the sea

■ Issues for discharging ALPS-treated water into the ocean

- The government’s decision on the policy is in line with international guidelines. It is an important decision from the perspective of ensuring the sustainability of decommissioning work.
- The discharge system is based on existing domestic and international experience, and safe discharge can be achieved through thoroughly maintaining manuals and strictly following the plan.
- In order to ensure the reliable operation of the implementation plan, it will be necessary to increase the transparency of the plan execution status, including confirmation and monitoring by third parties such as the IAEA.

Major targets and progress for fuel removal from spent fuel pools

Major targets

- Fuel removal from SFPs will start in FY 2027 to FY 2028 for Unit 1 and FY 2024 to FY 2026 for Unit 2. (For Unit 3, completed in February 2021)

Progress

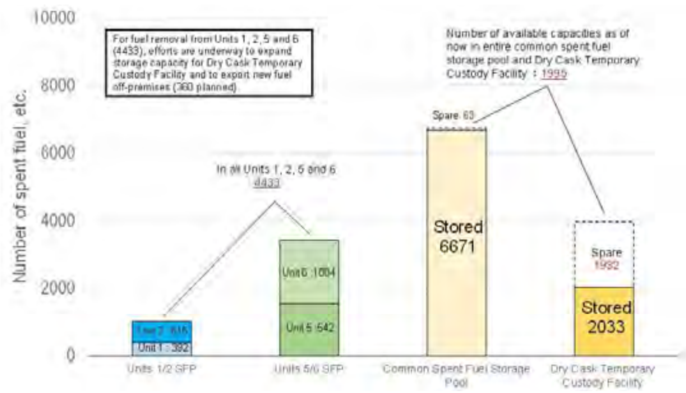


Fig. Storage status of spent fuel (As of March 2021)

Strategies for fuel removal from spent fuel pools

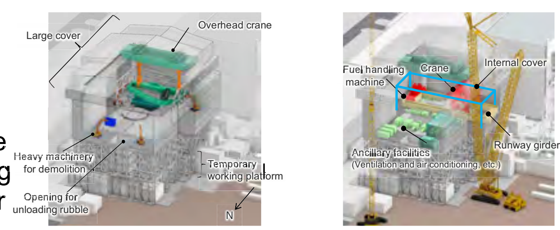
Strategy

Unit 1

- It is important to comprehensively consider removal of overhead crane and how to handle well-plugs, taking into account the impact on the other operations by performing thorough safety assessments.

Unit 2

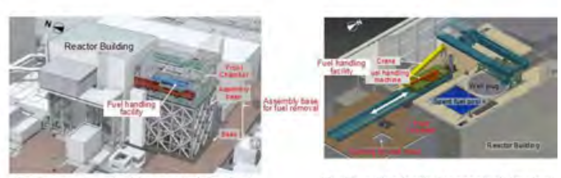
- For a fuel handling machine to be introduced as a new system, it is important to perform mockup tests and to be sufficiently familiar with remote operation.
- For further dose reduction of the operating floor, it is important to incorporate new survey results in decontamination and shielding installation methods .



During rubble removal (Conceptual drawing)   During fuel removal (Conceptual drawing)

(TEPCO material edited by NDF)

Fig. Fuel removal method from SFP (Unit 1)



Fuel removal method (conceptual drawing)   Fuel handling facility (conceptual drawing)

(TEPCO material edited by NDF)

Fig. Fuel removal method from SFP (Unit 2)

4. Analysis strategy for promoting decommissioning

- Analysis is one of the important factors in considering solid waste and fuel debris with significant uncertainty.
- To obtain good analysis results, it is effective to properly maintain (i) the methods and systems for analysis, (ii) the quality of the analysis results, and (iii) the size and quantity of sample.
  - ✓ It is necessary to organize division of roles according to the characteristics of the of facilities for analysis including the Ibaraki area.
  - ✓ Securing of analytical engineers and developing human resource are needed.
  - ✓ It is important to comprehensively review/evaluate at what stage of the accident, what elements were mainly contained, and what properties they have.
  - ✓ It is effective to diversify and expand the analysis methods and to perform comprehensive evaluation.

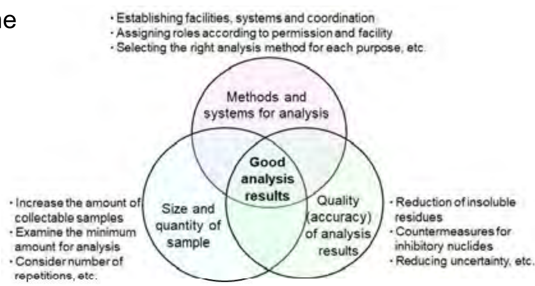


Fig. Three elements of the fuel debris analysis strategy

5. Efforts to facilitate research and development

- Updating the R&D medium-and-long term plan for fuel debris retrieval and others
- Strengthening of the functions of project planning and management in the Project of Decommissioning and Contaminated Water Management. (NDF has been participating in the secretariat)
- TEPCO's own R&D activities, and strengthen their structure. (Establishment of Decommissioning Technology Development Center)
- Strengthening and accelerating the perspective of the needs in the Nuclear Energy Science & Technology and Human Resource Development Project (TEPCO joined as the screening member for selection.)

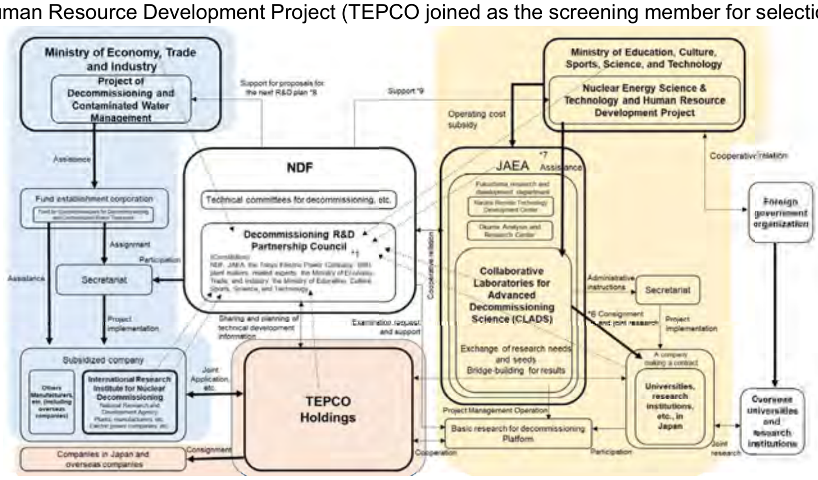


Fig. Overview of the R&D structure of the decommissioning of Fukushima Daiichi NPS

6. Activities to support our technical strategy

Project management approach

■ Significance and current status of project management

- For smooth promotion of the decommissioning project, it is necessary to establish and enhance a management system in which the organizations work together to achieve the goals.
- TEPCO has been working to build and strengthen its project management system, and the general framework was established, it is important to rooted in the on-site operations.

■ Key issues and strategies to be strengthened in the future

- Every employee needs educational materials and opportunities to learn about safety in a systematic way for establishing safety as an organizational culture.
- TEPCO needs to improve the owner’s engineering capability
  - ✓ Ability to assess and manage process risks
  - ✓ Improving acquisition management capability (Acquiring the final outcome (product or deliverables) by "making things up", in considering everything from development to manufacturing and operation and maintenance)
  - ✓ Promotion of internalization to develop the ability to plan/design, maintain/operate themselves.
- Securing and developing human resources for the smooth implementation of decommissioning projects
  - ✓ Development of the Medium-to-long term human resources development plan and its systematic implementation

6. Activities to support our technical strategy

Strengthening international cooperation

■ Significance of international cooperation

- It is important to learn lessons from precedent overseas decommissioning activities, and to utilize the world's highest level of technology and human resources.
- It is important to secure the confidence of the international community by disseminating accurate information on the decommissioning, and to promote decommissioning in a mutually beneficial manner by actively returning to the international community the knowledge, etc., gained through the accident.

■ Key issues and strategies

- It is necessary to continue this mutually beneficial relationship while also working to return the results to the interna community.
- It is important to continue to build and strengthen relationships by utilizing online systems and other means so that international cooperation will not be diluted.
- For the steady implementation of decommissioning, it is necessary to disseminate accurate information that meets the interests of the recipients through various opportunities.

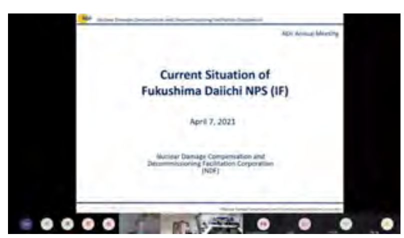


Fig. Annual meeting of NDF with foreign organizations concerned (held online in April 2021)

Local community engagement

■ Current status of initiatives for regional industrial and economic infrastructure

- The fundamental principle for the decommissioning of the Fukushima Daiichi NPS is "coexistence of reconstruction and decommissioning". Revitalization of decommissioning-related industries is an important pillar of TEPCO's contribution to the reconstruction of Fukushima.
- Efforts for the accumulation of decommissioning industries based on TEPCO's "Commitment" published at the end of March, 2020.
  - ① *Increased participation of local enterprises*
  - ② *Support for local enterprises to step up*
  - ③ *Creation of new local industries* → **Building an integrated decommissioning project implementation system in the Hamadori region※**

  (① and ②: **Matching support with prime contractors**; **Survey of the needs regarding human resource development**)

■ Key issues and strategies

- With the understanding of prime contractors, it is necessary to implement ordering and contracting methods that will make it easier for local companies to receive orders on a trial basis.
- Further strengthening of cooperation and collaboration with local governments, and local related organizations, including the Fukushima Innovation Coast Framework and the Fukushima Soso Recovery Promotion Organization, which are operating a joint consultation service and co-hosting matching business meetings.

※ TEPCO, " The recruitment of factory operation partners and Cask license partners for the establishment of a factory for decommissioning-related products in Hamadori Region of Fukushima Prefecture", announced on July 9, 2021

C.1.2. Nuclear Regulation Authority Japan Investigations and Analysis

Presentation Material

Summary of NRA's Research for Fukushima Daiichi Nuclear Power Plant Accident (December 2020-October 2021)

Masaya YASUI
Secretariat of Nuclear Regulation Authority, Japan

Overview

- The DOE requested the NRAJ to update its activities concerning Fukushima Daiichi Nuclear Power Accidents (FDNPA).
- This presentation covers mainly the information which has been obtained after the last presentation in November 2020.
- "Report 2021" was finalized by the NRAJ Commissioners on March 2021, and includes the presentation in 2020.
- "Report 2022" is planned to be finalized on March or April 2022, and will include the information in this presentation.

2

Note

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- In this presentation, radiation field is indicated with mSv/h.
- Scientifically speaking, mGy/h is more appropriate.
- However, the detectors/dosimeters which are used in the investigation provides data on mSv/h basis.
- In order to avoid the confusion accompanying with conversion, measured values are used as they are.

3

Main Contents

NRA , Japan
Nuclear Regulation Authority

1. Cs under the upper Shielding-plug of Unit 2
2. Information relating SGTS lines
3. Possible sources of organic gas inside the PCV
4. OECD/NEA international project of FDNPA

4

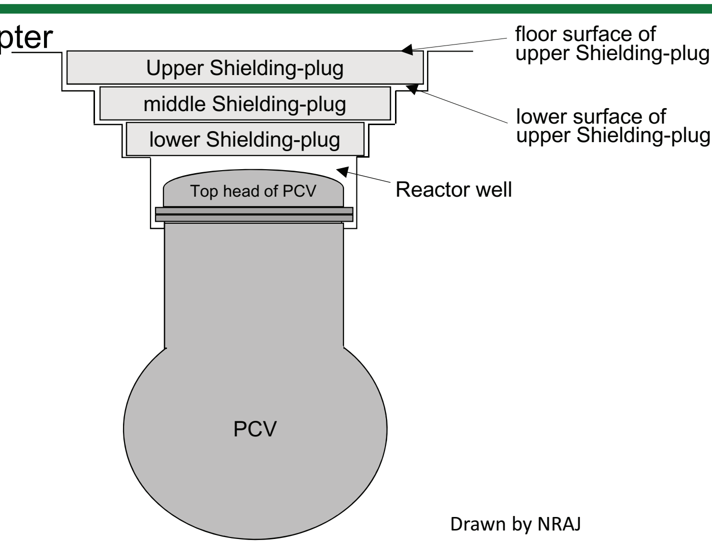
1. Cs under the upper Shielding-plug of Unit 2

Under the limitation of investigation activities due to COVID-19, the in-situ investigation by the NRAJ focused on the Cs distribution under the upper Shielding-plug of Unit 2

5

1. Cs under the upper Shielding-plug of Unit 2

Wordings in this chapter



Drawn by NRAJ

6

1. Cs under the upper Shielding-plug of Unit 2

Key points in the report 2021

□Around 30 PBq of Cs137 exists under the upper Shielding-plug of Unit 3.

□More than 30 PBq of Cs137 possibly exists under the upper Shielding-plug of Unit 2.

Confirmatory measurement is planned.

□Curious distribution of contamination by Cs137 in the SGTS lines of Unit 1 & 2 was find out. (At Fukushima Daiichi, vented gas flows in SGTS lines.)

□Computer simulation by the JAEA could not reconstruct the measured contamination pattern of SGTS lines

7

1. Cs under the upper Shielding-plug of Unit 2

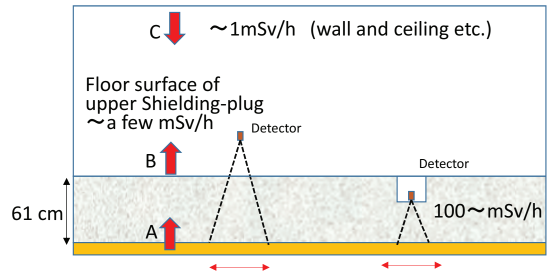
The aims of the in-situ investigation in the year 2021 are;

1. Confirm the existence of large amount Cs under the upper Shielding-plug of Unit 2.
2. Know the Cs distribution under the upper Shielding-plug of Unit 2
3. Obtain the clue of Cs movement route through Shielding-plugs of Unit 2.

Investigation in 2021 consists of three steps.

8

1.1 Step1 Trial measurement using bore-holes



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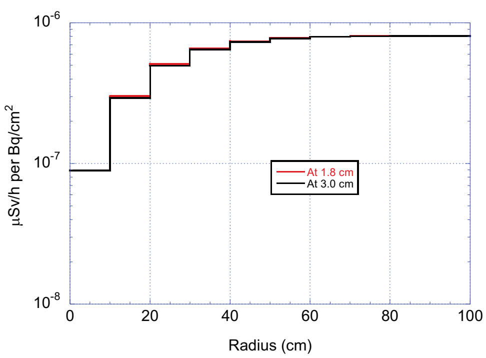
Measuring in bore-holes can reduce the effect of floor surface contamination.

9

1.1 Step1 Trial measurement using bore-holes



The relation between ambient dose rates and distances from the center of the shielding-plug (height from the surface: 1.8cm and 3.0cm)

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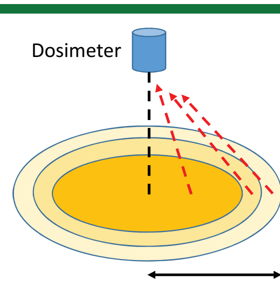

The effect from radiation sources according to the distance from the point it exists

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The examination revealed that the effect from radiation source is limited to the area inside a circle of radius of 50cm

10

1.1 Step1 Trial measurement using bore-holes

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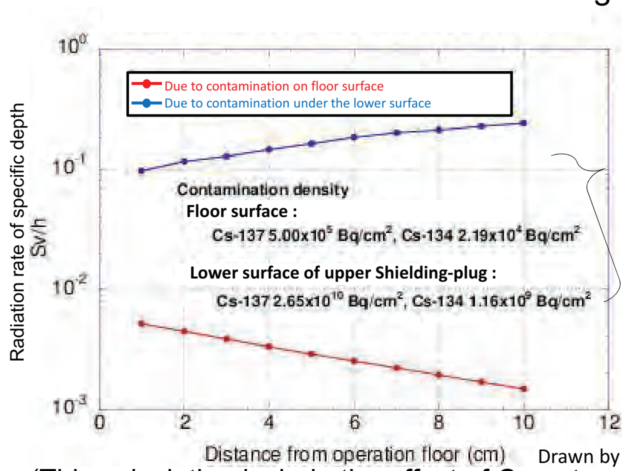
Calculated contribution of Cs137 existing under the upper Shielding-plug



These figures are assumptions

If strong Cs137 contamination exists under the upper shielding-plug, the measured radiation field should become higher as the detector goes deep in bore-holes.

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(This calculation include the effect of Compton scattering.)

1.1 Step1 Trial measurement using bore-holes

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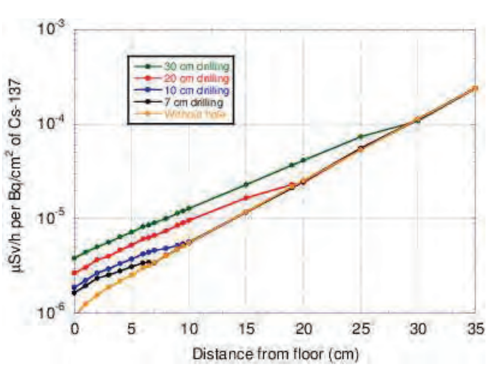
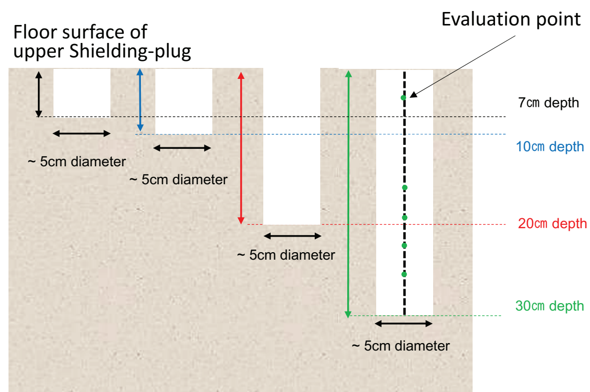
Simulated radiation rate in the bore-holes of various depth (0cm, 7cm, 10cm, 20cm, 30cm)





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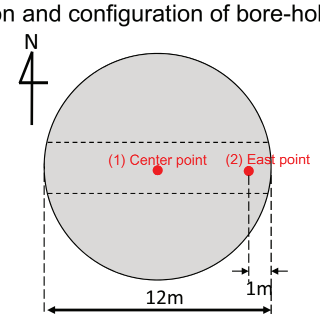
1.1 Step1 Trial measurement using bore-holes

Two existing bore-holes on the upper Shielding-plug of Unit 2

<u>Method</u>: Measuring the radiation field at various depth in the 7cm deep bore-holes on the Shielding-plug of Unit 2

<u>Advantage</u>: Avoiding the effect of Surface (including walls & ceiling of R/B) contamination

Position and configuration of bore-holes



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13

1.1 Step1 Trial measurement using bore-holes

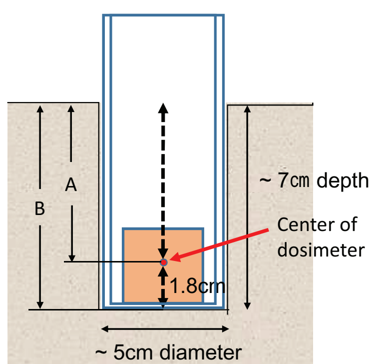

Fig. Configuration of Measuring Equipment

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* Each depth is measured from the floor surface of upper Shielding-plug

14

1.1 Step1 Trial measurement using bore-holes

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Results of measurement

(1) Center point

Depth* [cm]	mSv/h
4.2 (6.0)	1169
3.2 (5.0)	1070
2.2 (4.0)	944
1.2 (3.0)	825
0.2 (2.0)	682, 690

A (B)

(2) East point

Depth* [cm]	mSv/h
5.2 (7.0)	255
4.2 (6.0)	277
3.2 (5.0)	290, 300
2.2 (4.0)	292
1.2 (3.0)	255
0.2 (2.0)	225

A (B)

* Each depth is measured from the floor surface of upper Shielding-plug

The committee of Accident Analysis of Fukushima Daiichi Nuclear Power Station 22th meeting (September 2021, NRAJ)
Document 3-3
https://www.nsr.go.jp/data/000364989.pdf

15

1.1 Step1 Trial measurement using bore-holes

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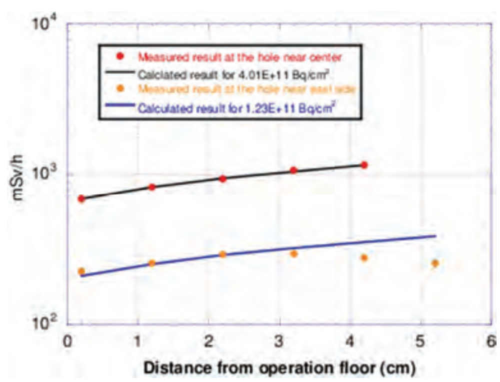
<u>Result</u>:

Measured results correspond reasonably well with the calculated result.



(A) value

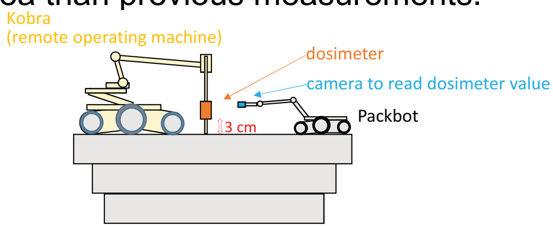
Drawn by NRAJ

16

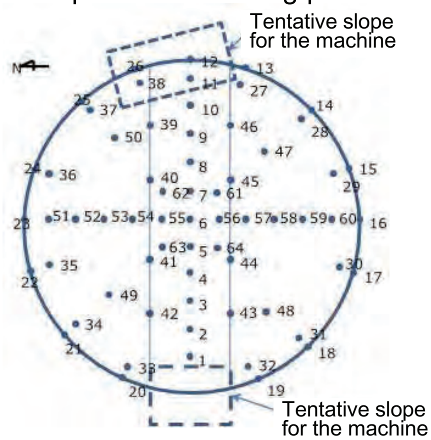
1.2 Step2 To obtain higher resolution data

Method: (measuring mechanism below)
Measuring the radiation field at the height of 3 cm above the floor surface of Shielding-plug of Unit 2

Advantage:
Able to know the radiation from narrower area than previous measurements.



A map of 64 measuring point



“Numbers” are numbering of the measuring points

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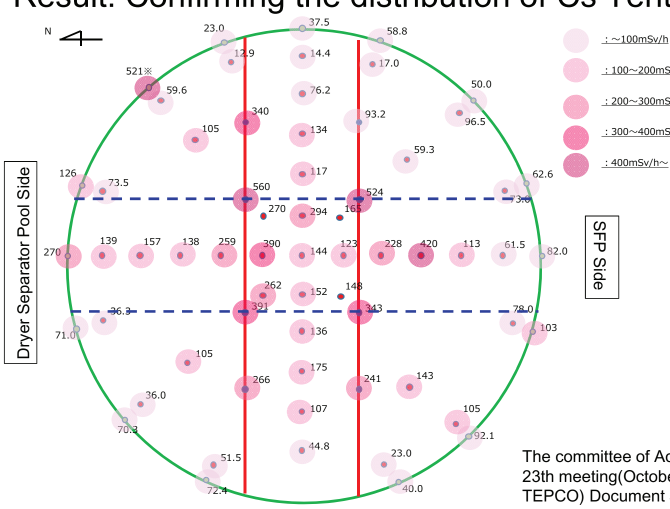
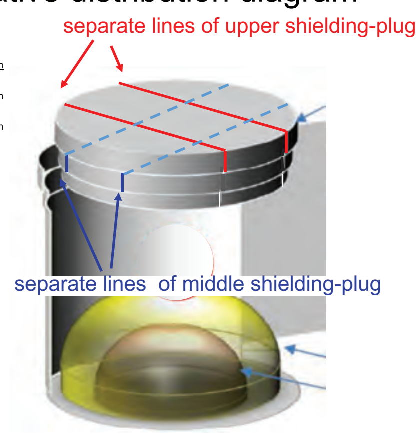
1.2 Step2 To obtain higher resolution data

Result: Confirming the distribution of Cs Tentative distribution diagram





The committee of Accident Analysis of Fukushima Daiichi Nuclear Power Station 23th meeting(October 2021, TEPCO) Document5-1, 21th meeting (July 2021, TEPCO) Document 5-3 https://www.nsr.go.jp/data/000358693.pdf 18

1.2 Step2 To obtain higher resolution data

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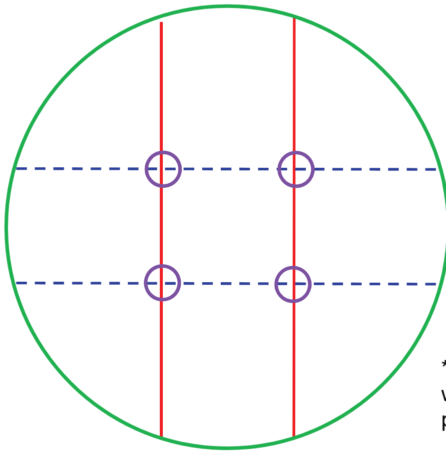

*This radiation field map illustrates dose-rates without including data of separate lines and peripheral gap

Radiation-field Map (Image)

Drawn by Dr. Hayashi, NRA

19

1.2 Step2 To obtain higher resolution data

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Nuclear Regulation Authority

1.2-2 To be considered

- Possibility of the effect of weak distribution of surface-contamination. (Maybe, not so important)
- Possibility of the effect of contaminated dust in the bore-hole.
  - (Although clean-up activity and preliminary measurement of contaminated dust was completed.)
- Existence of curiously high radiation points.
- The resulting radiation field might be too high.

20

1.2 Step2 To obtain higher resolution data

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Nuclear Regulation Authority

1.2-3 Possible release route of Cs

- Separate lines (red lines) of Shielding-plug shows considerably higher radiation field than the peripheral gap (green) of the Shielding-plug. (See p.18)
- The highest radiation field was measured at the cross-points of separate line.

21

1.2 Step2 To obtain higher resolution data

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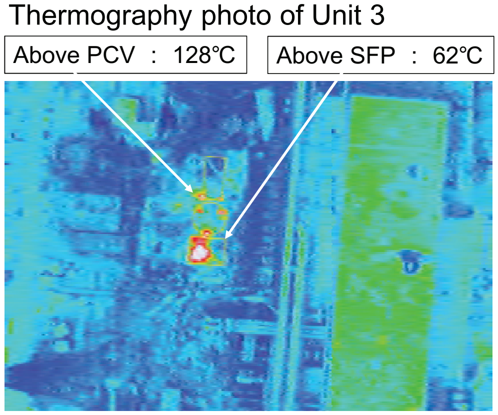
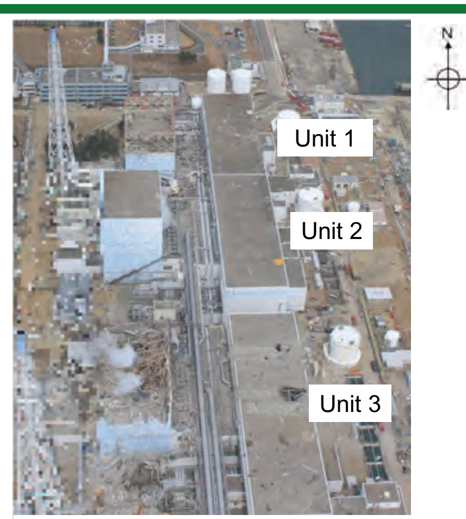
Nuclear Regulation Authority

1.2-3 Possible release route of Cs

Thermography photo of Unit 3



20 March 2011
Photo taken by Technical Research & Development Institute, Ministry of Defense



20 March 2011 Photo taken by TEPCO

22

1.3 Step3 Final measurement using bore-holes

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Method:
Boring and using new bore-holes
(in total 13 holes, 10cm deep)

Advantage:
Clean bore-holes

- No possibility of contaminated dust effect
- Avoiding the effect of surface contamination

Higher resolution (by more holes)

廃炉· 汚染水· 処理水対策チーム会合 / 事務局会議（第95回） 資料3-3 2号機オペフロ内シールドプラグ穿孔部調査について (October 2021, TEPCO) https://www.meti.go.jp/earthquake/nuclear/decommissioning/committee/osensuitaisakuteam/2021/10/95-3-3-4.pdf

23

1.4 Future plans

NRA, Japan Nuclear Regulation Authority

- NRAJ's measurements of Unit 2 Shielding-plug contamination for the purpose of accident-analysis are almost finished.
- More through (numerically and geologically) measurement are important for the decommissioning activity. TEPCO/METI will continue the appropriate researches with NRAJ's technical support.

24

1.5 Information of radiation field in the Reactor Well (Measured by TEPCO)

□February 2021, TEPCO investigated the inside of Unit 2 reactor well.

□Radiation field in the reactor well was substantially lower than expected.

- ➢This indicates that the amount of Cs under the Top-Head of the Unit-2 PCV is limited.

□The reasons have not yet been finalized.

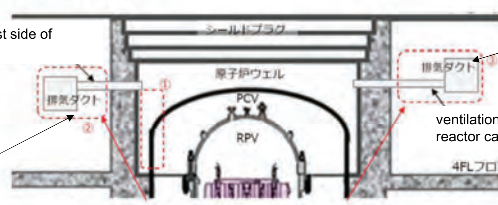
25

1.5 Information of radiation field in the Reactor Well (Measured by TEPCO)

Overview of investigation of Reactor Well of Unit 2

- ✓ Shoot a video of the reactor well & measure dose rates
- ✓ Investigation and sampling of dust in the ventilation line and corroded area of duct of west side of reactor cavity
- ✓ Appearance survey of the ventilation line and the corroded area of duct of east side of reactor cavity

ventilation line of west side of reactor cavity



ventilation line of east side of reactor cavity

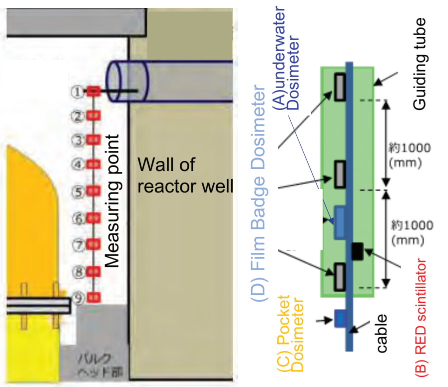
The committee of Accident Analysis of Fukushima Daiichi Nuclear Power Station 21th meeting(July 2021, TEPCO) Document 5-3 https://www.nsr.go.jp/data/000358693.pdf

26

1.5 Information of radiation field in the Reactor Well (Measured by TEPCO)

Result of the investigation in the reactor well of Unit 2 (dose-rates)

- Maximum dose-rate in the well was 530 mSv/h, measured by underwater dosimeter
- Dose-rates in the well increase as it goes down



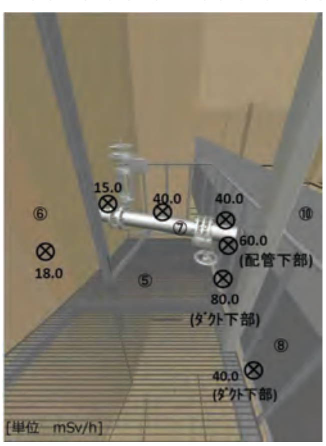
Measuring point	Distance (mm)	(A) Underwater dosimeter		(B) RED scintillator	(C) Pocket dosimeter	(D) Film Badge dosimeter
		Dose rate (mSv/h)				
		Measured on May 20, 24	Measured on June 23th			
①	0	74.6	-		Approx. 315	Top: 220 Middle: 270 Bottom: 360
②	500	150	-			
③	1000	330	-			
④	1500	300	280	Under lowest limit		
⑤	2000	310	-			
⑥	2500	380	-			
⑦	3000	440	-			
⑧	3500	530 (Max)	370	400±100m Gy/h		
⑨	4000	350	-			

The Committee of Accident Analysis of Fukushima Daiichi Nuclear Power Station 21th meeting (July 2021, TEPCO) Document 5-3 https://www.nsr.go.jp/data/000358693.pdf
Fukushima Prefecture Decommission Safety Monitoring Conference 3rd meeting in 2021 (September 2021, TEPCO) Document 3-2 https://www.pref.fukushima.lg.jp/uploaded/attachment/469770.pdf

27

1.5 Information of radiation field at the Reactor Well ventilation lines (Measured by TEPCO)

Result of the dose-rates survey and smear test of the ventilation line of west side of reactor cavity



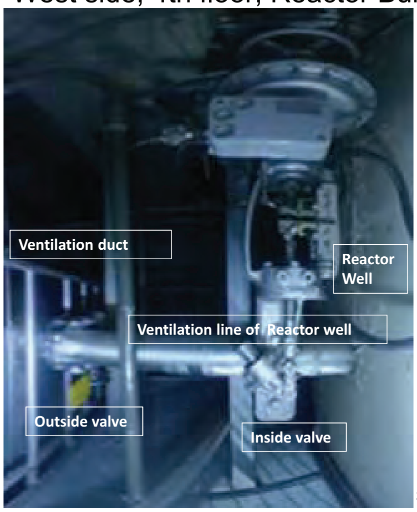
Smear sampling point	β (cpm)	α (cpm)	γ (mSv/h)	β+γ (mSv/h)
⑤	>100000	0	0.15	10.0
⑥	>100000	30	0.14	5.0
⑦	>100000	50	0.16	12.0
⑧	>100000	0	0.15	8.0
⑩	>100000	0	0.14	7.0

The committee of Accident Analysis of Fukushima Daiichi Nuclear Power Station 21th meeting (July 2021, TEPCO) Document 5-3 https://www.nsr.go.jp/data/000358693.pdf

28

1.5 Information of radiation field at the Reactor Well ventilation lines (Measured by TEPCO)

West side, 4th floor, Reactor Building of Unit 2



2020.10.8 taken by NRA

	Unit 1	Unit 2
Inside valve R/B 4th floor	Air-operated butterfly valve (Normal Open/Fail Close) BF-12 (west) BF-13 (east)	Air-operated butterfly valve (Normal Open/Fail Close) BF2-12 (west) BF2-13 (east)
Outside valve R/B 4th floor	Manual valve BF-18 (west) BF-19 (east)	Manual valve BF2-18 (west) BF2-19 (east)
The size of ventilation line of Reactor well	Diameter 150 mm	Diameter 150 mm
Investigation by TEPCO	2019/11/7 BF-13 is closed	2021/5/20,24 BF2-12, BF2-13 is open BF2-18 is open

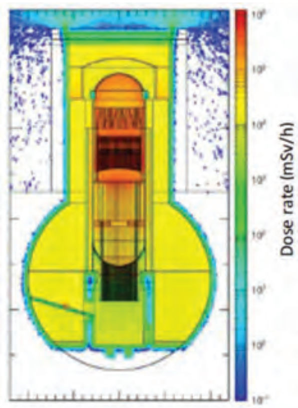
The committee of Accident Analysis of Fukushima Daiichi Nuclear Power Station 21th meeting (July 2021, NRAJ) Document 3-4 https://www.nsr.go.jp/data/000358689.pdf 29

1.5 Information of radiation field at the Reactor Well ventilation lines (Measured by TEPCO)

The mechanism of inside valves (diaphragm valve)

The committee of Accident Analysis of Fukushima Daiichi Nuclear Power Station 21th meeting (July 2021, TEPCO) Document 5-3 https://www.nsr.go.jp/data/000358693.pdf 30

1.5 Information of radiation field in the Reactor Well (Measured by TEPCO)



Dose rate distribution analysis of Unit 2 by PHITS (Particle and Heavy Ion Transport code System)

According to the study by Dr. Okumura (CLADS_JAEA)

Radiation field in the reactor-well was previously estimated to be around 40 Sv/h due to the radioactive material under the PCV-head.

However, the results of the measurements (including double-check measurement)

indicated less than 0.6 Sv/h.

Contamination inside PCV-head and reactor-well needs to be reconsidered substantially.

"Technical development of dose rate distribution analysis and underwater debris exploration for the decommission of Fukushima Daiichi Nuclear Power Station, (8)Estimation of 3D dose rate distribution in PCV (Unit 1 & 2)", Okumura et al., Atomic Energy Society of Japan Autumn conference in 2018

31

2. Information relating SGTS lines

32

2.1 SGTS filter trains of Unit 1 & 2

The NRA tried to measure the contamination level of SGTS filter trains of Unit 1 and Unit 2 in detail, but has not yet succeeded due to very high radiation field (in particular Unit 1).

Boundary conditions for vent gas flow analysis might be modified.

33

2.2 Water in the SGTS filter trains of Unit 3 & Unit 4

November 2020 TEPCO found water in the SGTS filter train of Unit 3 and Unit 4

100 litter in Unit 3 (A-train : 3 litter, B-train : 100 litter)

a little water in Unit 4 (A-train: a little, B-train: not investigated)

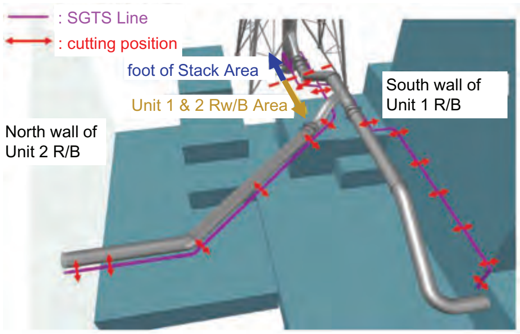
It is believed that the origin the water is the vented gas. But the reasons for difference between A-train and B-train are not known.

34

2.3 Investigation of inside of the SGTS line of Unit 1 & 2 (coming)

TEPCO plans to remove the SGTS lines of Unit 1 & 2 for decommissioning purposes.

Using this opportunity, detailed contamination measurement and sampling of radioactive material is under preparation.



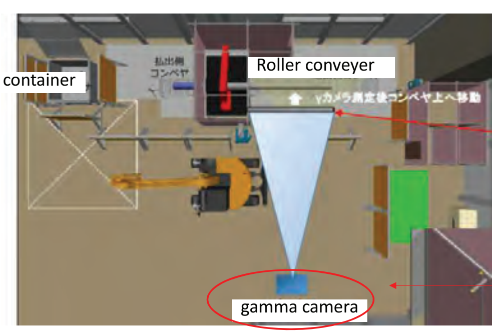
Container of cut-out sample

The committee of Accident Analysis of Fukushima Daiichi Nuclear Power Station 22th meeting (September 2021, TEPCO) Document 5-2 https://www.nsr.go.jp/data/000364993.pdf

35

2.3 Investigation of inside of the SGTS line of Unit 1 & 2 (coming)

Survey of pipes of SGTS line by gamma camera etc.



Pipe length
Max: 7m
Min: 2.5m

4 type of gamma camera

No.1
Hitachi Corp.
HDG-E1500 (pin-hole type gamma camera)

No.2
MIRION CANBERRA Corp.
iPIX (new type gamma camera)

No.3
Chiyoda Technol Corp.
gamma catcher
(Compton camera)

No.4
JAEA
Small Compton camera

Material data from TEPCO

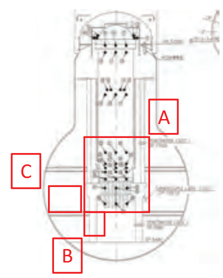
The committee of Accident Analysis of Fukushima Daiichi Nuclear Power Station 22th meeting (September 2021, TEPCO) Document 5-2
https://www.nsr.go.jp/data/000364993.pdf

36

3. Possible sources of organic gas inside the PCV

TEPCO offers the samples which consist of identical or similar materials as cables and thermal insulator of Unit 3

Offered sample	components	Pictures	Cross section image	Application
PN cable (insulator)	Flame-retardant ethylene propylene rubber			RPV bottom thermometer cable (140m) A
PN cable (sheath)	Special chloroprene rubber			
CV cable (insulator)	Cross-linked polyethylene			RIP pump power cable B
CV cable (sheath)	Flame-retardant heat-resistant vinyl			
Thermal insulator	Urethane			Thermal insulator of CCW system line ($8m^3$, 320kg) C



The committee of Accident Analysis of Fukushima Daiichi Nuclear Power Station 22th meeting (September 2021, TEPCO) Document 5-1 https://www.nsr.go.jp/data/000364992.pdf , 23th meeting (October 2021, TEPCO) Document 5-2 https://www.nsr.go.jp/data/000367851.pdf

37

3. Possible sources of organic gas inside the PCV

TEPCO's experiment plan (by the end of March 2022)

- Heating test of cables and painting material, which could have caused flammable organic gas because of large inventory in the PCV
- Identification and quantitative analysis of flammable organic gas of each temperature below
  - 200℃ : limit temperature of PCV
  - 1000℃: limit temperature of the experiment machine

sample	components
PN cable (insulator)	Flame-retardant ethylene propylene rubber
PN cable (sheath)	Special chloroprene rubber
CV cable (insulator)	Cross-linked polyethylene
CV cable (sheath)	Flame-retardant heat-resistant vinyl
Thermal insulator	Urethane
Other materials	･･･

Share the results of each experiment

sample

JAEA's experiment

STEP1 : TG-DTA-MS analysis

STEP2 : GC-MS analysis

The committee of Accident Analysis of Fukushima Daiichi Nuclear Power Station 23th meeting (October 2021, TEPCO) Document 5-2 https://www.nsr.go.jp/data/000367851.pdf

38

3. Possible sources of organic gas inside the PCV

General image of the experiment is as below.

The committee of Accident Analysis of Fukushima Daiichi Nuclear Power Station 23th meeting (October 2021, JAEA) Document 4-1 https://www.nsr.go.jp/data/000367849.pdf 39

3. Possible sources of organic gas inside the PCV

Fire at Unit 3	
March 14 11:01	Explosion at reactor building
March 21 15:55	Grey smoke on the top of south-east part of reactor building
March 23 16:20	Black smoke near the reactor building

March 21th 16:10

Photo taken by TEPCO

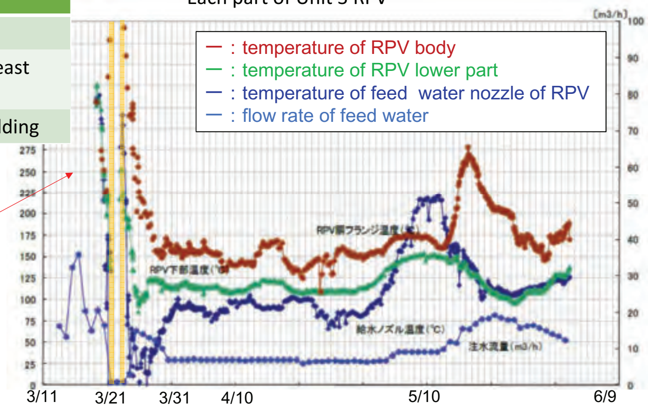
40

3. Possible sources of organic gas inside the PCV

Fire at Unit 3	
March 14 11:01	Explosion at reactor building
March 21 15:55	Grey smoke on the top of south-east part of reactor building
March 23 16:20	Black smoke near the reactor building

Two temperature peaks can be observed from March 21 to 23

Each part of Unit 3 RPV



Japanese Government report to the IAEA Ministerial Conference on Nuclear Safety about Fukushima Daiichi Nuclear Power Site Accident (June 2011, Nuclear Emergency Response Headquarters) https://japan.kantei.go.jp/kan/topics/201106/pdf/chapter_iv_all.pdf

41

4. OECD/NEA Next Project to ARK-F and PreADES

- NRAJ started the discussion of new project from June 2021.

- The contents of the project is agreed by the potential participants and the situation will be reported to the NEA-CSNI in December.

<u>For More information</u>

Joint Meeting of the PreADES and ARC-F Projects on FACE Project on 21st-22nd September 2021:
https://www.oecd-nea.org/download/arcf/FACE00-2021/index.html

42

4. OECD/NEA Next Project to ARK-F and PreADES

- The new project covers not only the ARK-F and PreADES, but also BSAF and Round Robin, and will be placed under one Managing Board.

- Object of the project is to know <u>what happened in the FDNPA</u>.

- NRAJ coordinates the related Japanese sectors including METI, therefore the project has <u>only one top in Japan</u>.

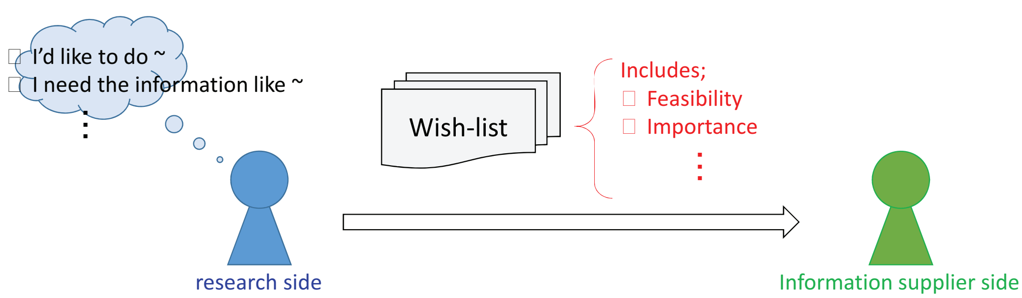
- NRAJ will supply the information basing on its <u>own investigation and analysis</u>.

43

4. OECD/NEA Next Project to ARK-F and PreADES

One of the key points is;

When the research side prepares a wish-list, feasibility and importance should be well considered and be explained to the information suppliers such as NRAJ and TEPCO.



44

5. Conclusion

The second phase of accident investigation/analysis of Fukushima Daiichi by the NRAJ still continues. Additional information/knowledge is coming and will be shared internationally.

Ceaseless Quest for Safety Improvement!!

Thank you for your attention!

C.1.3. Tokyo Electric Power Holdings, LLC (TEPCO Holdings) Investigations

C.1.3.1. Knowledge Management Mechanism

Knowledge Management Mechanism (Draft)

Dec. 1 2021 Tokyo Electric Power Company Holdings, Inc.

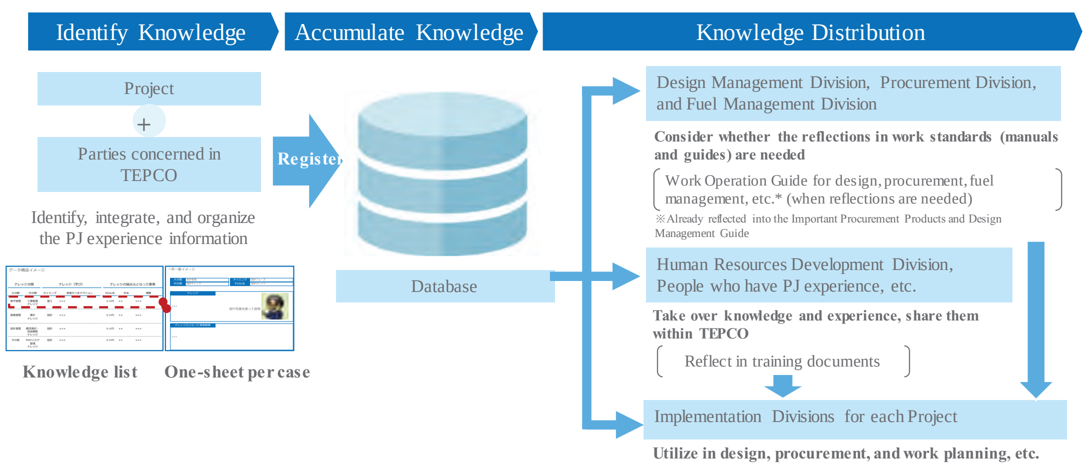
IAEA INTERNATIONAL PEER REVIEW OF MID-AND-LONG-TERM ROADMAP TOWARDS THE DECOMMISSIONING OF TEPCO’S FUKUSHIMA DAIICHI NUCLEAR POWER STATION

Acknowledgement 23
The IAEA Review Team acknowledges the establishment of a formal knowledge management information platform to identify, accumulate and disseminate lessons learned to internal stakeholders at Fukushima Daiichi NPS. This collection of knowledge should be useful in the future for carrying out the same or similar activities or processes.

Overview of Knowledge Identification and Utilization

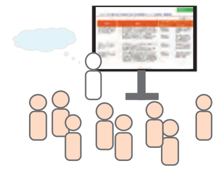
1

- Valuable knowledge will be selected by project members and parties concerned in TEPCO and will be organized into a list/one-sheet per case format and registered in the database.
- It is shared the registered knowledge with subsequent PJs and distributed through Decommissioning training course, etc.



Distribution Plan of Valuable Knowledges within TEPCO

2

	Apply into tools and manuals	Collection of Cases	In-house sharing
			
Description	Reflect learning from previous cases in tools and manuals such as risk checklists.	Create the list and one-sheet per case, of learnings obtained, and disseminate them.	Conduct in-house trainings through decommissioning engineer courses based on the list and one-sheet per case.
Effect to aim	Ensure learning as a mechanism by applying the learning from previous cases into tools and manuals.	Understand the actions to be taken for success by searching for previous learning in daily work.	Consider as a matter of yourself what kind of failure can occur and of thing can be taken to prevent by receiving the trainings based on actual cases.
	Knowledge into work procedure	Handbook on knowledges	Training using special knowledge

TEPCO

Example of one-sheet per case (necessity for time to address the issue found in the site)

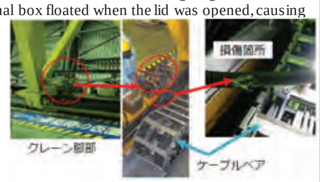
3

Major category	Facilities and Operation Management	Timing	On-site construction
Medium category	Knowledge on Creating On-site Schedule	PJ/Gr	Unit 3 Fuel Removal PJ

Event Overview of Knowledge Source

Case No. 51
[Event] While the hoisting gear for the cask gripping was being unloaded into the SFP, a submersible camera for monitoring contacted the hoisting gear, and the camera was damaged.
[Cause] : Based on the mock-up, it was installed the submersible camera additionally to the site in order to improve visibility when operating the hoisting gear, however operators did not recognize that the hoisting gear and the camera might interfere with each other.
[Loss] : Camera (¥30 million) + installation cost

Case No. 55
[Event] : During the test operation duration, an alarm occurred when the lid of the relay terminal box on FHM was opened or closed in the time of investigating another issue.
[Cause] : In addition to the poor fit of the cable in the terminal box, a shield was attached to the lid, therefore the terminal box floated when the lid was opened, causing noise in the cable. (lid that does not need to be opened or closed frequently during the mock-up operation)
[Loss] : Cause investigation and relaying the cable (¥ several ten million) + duration required for investigation and countermeasure (2.5 months)



Case No.58
[Event] : It was found that part of the crane could not be operated during operation training in the site.
[Cause] : The cable from terminal box on the crane feet, was caught with cable bear and damaged.
[Loss] : Duration required to procure/relay the cables (several months).

Summary of the cases: It had assumed that could be simulated by the full scall mock-up test at plant, and had confirmed (only confirmed whether each operation was possible or not)with minimal field test. In addition, it was planned the pre-use inspection as soon as completed the on-site test, therefore there was no duration to review.

Knowledge

・No matter how much simulated in mock-up, there are what you won't know unless you try them in the site!
・It is better to ensure the "test duration in the site" and "duration for addressing the issues found in the site" after installation!

Disseminate the Knowledge

・**PG/PJ planning plant mock-up (Unit1&2 Fuel handling Equipment Project Groups, Fuel Debris Retrieval PGs, etc.)**
※example: it will be conducted a final full test in the site test at Unit 2. In addition, it will be secured a review duration after the on-site test.

TEPCO

Thank you for your attention.

TEPCO

Reference | Knowledge Category (Draft)

Category Preparation Categorize to "QMS (Design/Procurement, Operation, Radiation Management, Maintenance, etc.)" and "Knowledge type" based on future utilization (searchability of documents, etc.)

Major category (QMS)	Medium category (Knowledge Type)
Design Management	Knowledge on Conceptual Study and Technology Development
	Knowledge on Design (basic design and detailed design)
	Knowledge on Technical Review
Procurement Management	Knowledge on Creating Requirements Specification
	Knowledge on Estimation
	Knowledge on Supplier Evaluation and Selection
	Knowledge on Negotiation
Facilities and Operation Management	Knowledge on Troubleshooting
	Knowledge on Creating On-site Schedule
	Knowledge on Supervising Construction
	Knowledge on Creating Maintenance and Operation Plan
	Knowledge on Creating Inspection Work Plan
Radiation Management	Knowledge on Alpha Nuclide
	Knowledge on Dose Reduction (shielding and decontamination)
Others	Knowledge on Mock-up
	Knowledge on Formulating Project Plan, and Progress Management
	Knowledge on Risk Management in Project
	Knowledge on Risk Management for Ageing Deterioration (operation and maintenance risk management)
	Knowledge on Decision-making
	Knowledge on Licensing
	Knowledge on Communication
	Knowledge on Keeping up and Improving Reputation

With respect to category, it will be reviewed while creating one-sheet per case considering the usability from users' point of view

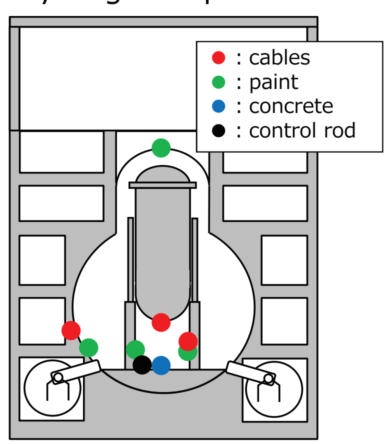
TEPCO

2021/11/29
OWADA Kenji
Tokyo Electric Power Company Holdings, Inc.

1. Overview

TEPCO

- The results of the video analysis of the hydrogen explosion in unit 3 reactor building conducted by the NRA suggest that not only hydrogen but also flammable organic gases may have been generated in the reactor building at the time of the explosion.
- Summarizing the sources of flammable organic gases and the possibility of their generation is useful for estimations such as that of existence of MCCI, the estimation of peak temperature of PCV, and the consideration of the contribution of flammable organic gases to hydrogen explosion.
- In this presentation, the following results of past investigations are summarized.
  - ① The state of structures that can be a source of flammable organic gases around RPV and PCV
    - ✓ cable coating
    - ✓ paint (epoxy)
    - ✓ concrete
    - ✓ control rod (B_4C)
  - ② The information related to the estimation of ambient temperature around RPV and PCV at the time of the accident
    - ✓ lead shielding, lead-wool mat



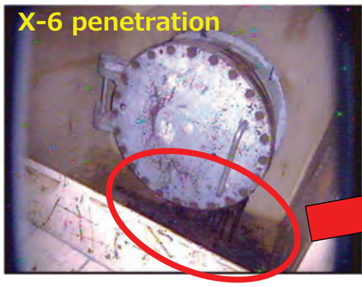
The image of the past investigation areas

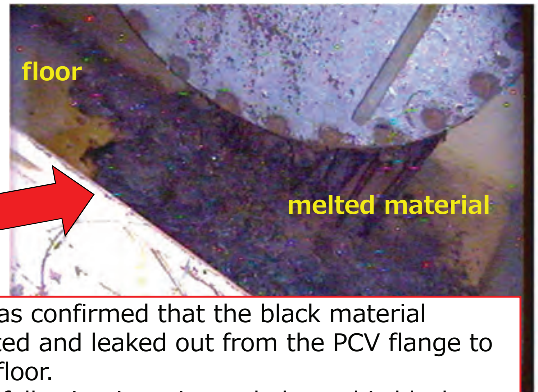
1

2. Results of past investigations on the sources of flammable organic gases

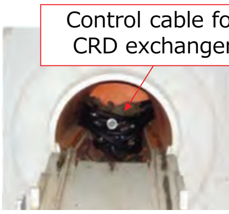
TEPCO

- Unit 2 PCV investigation (the state in the vicinity of the X-6 penetration) in 2015





Control cable for CRD exchanger



Inside of the X-6 penetration viewed from PCV in unit 5

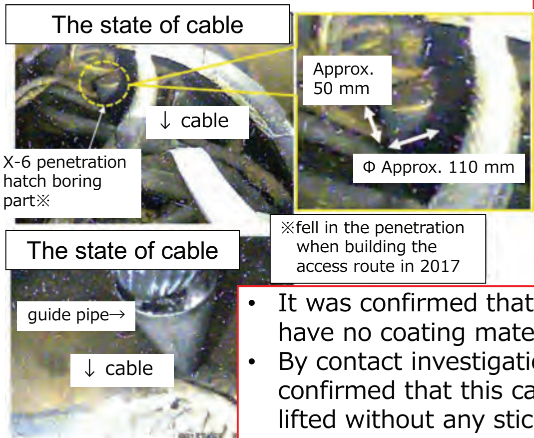
- It was confirmed that the black material melted and leaked out from the PCV flange to the floor.
- The following is estimated about this black material:
  - ✓ the O ring for the penetration flange seal
  - ✓ coating of control cable for CRD exchanger

2

2. Results of past investigations on the sources of flammable organic gases

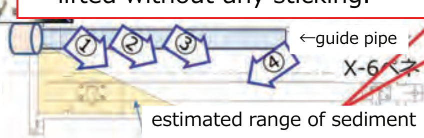
TEPCO

- Unit 2 PCV investigation (the state of the inside of the X-6 penetration) in 2020



Contact investigation of the cable

- It was confirmed that the cables have no coating material.
- By contact investigation, it was confirmed that this cable can be lifted without any sticking.

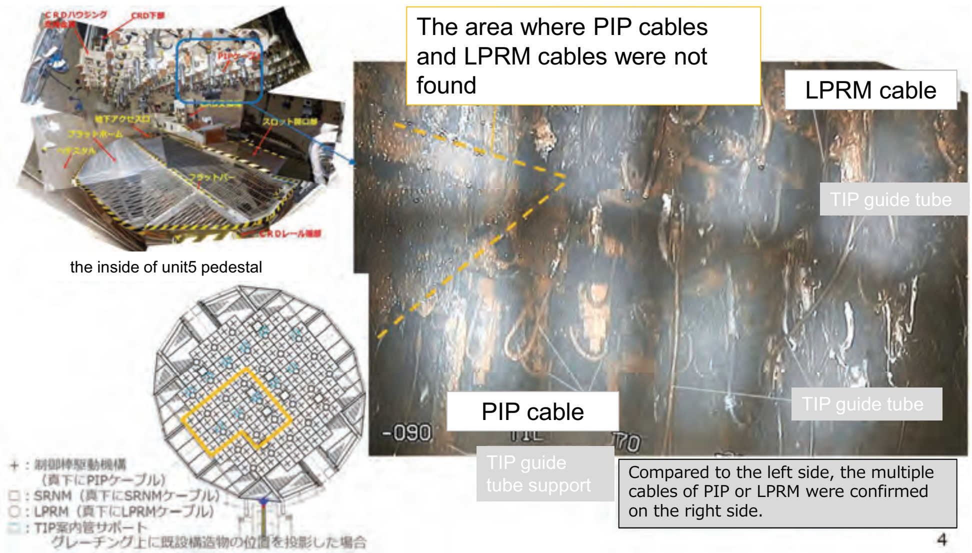


3

2. Results of past investigations on the sources of flammable organic gases

TEPCO

- Unit 2 PCV investigation (the state of the upper part of the pedestal) in 2017

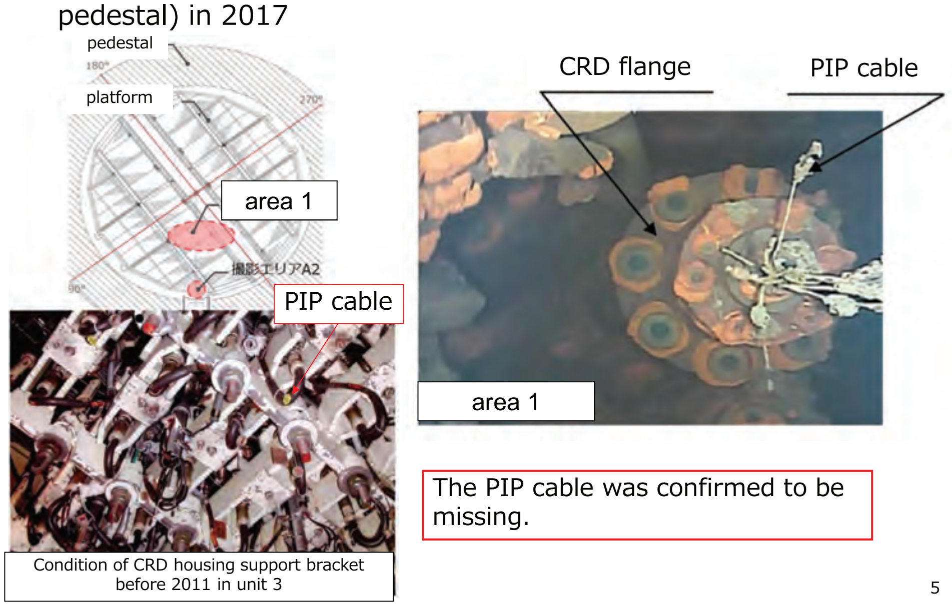


4

2. Results of past investigations on the sources of flammable organic gases

TEPCO

- Unit 3 PCV investigation (the state of the upper part of the pedestal) in 2017

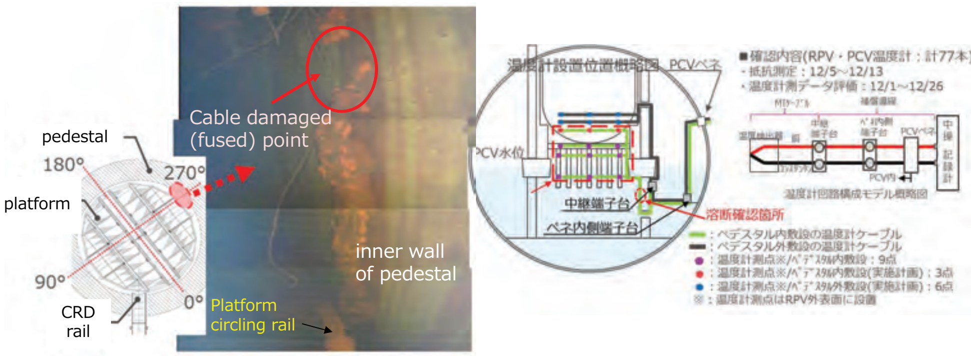


5

2．Results of past investigations on the sources of flammable organic gases

TEPCO

■ Unit 3 PCV investigation (the state of the inner wall of the pedestal) in 2017



- It was confirmed that the RPV bottom thermometer cables laid on the inner wall of the pedestal are damaged.
- These cables are assumed to be damaged by adhesion of high temperature molten material such as fuel which fell in the pedestal.
- The RPV bottom thermometer cables laid on the outside of the pedestal are unlikely to have had high temperature molten material adhered to them and to be damaged.

6

2．Results of past investigations on the sources of flammable organic gases

TEPCO

■ Unit 1 PCV investigation (the state of the inner wall of the PCV) in 2012

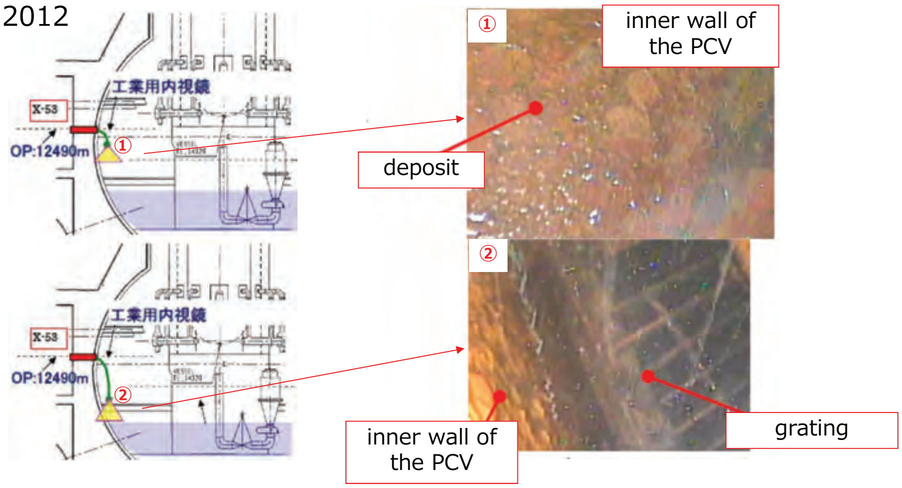
Although there were some peelings of the paint and roughness of the surface on the inner wall of the PCV, no major damage or deformation was observed there.

7

2. Results of past investigations on the sources of flammable organic gases

TEPCO

■ Unit 2 PCV investigation (the state of the inner wall of the PCV) in 2012



There were some peelins of the epoxy paint and roughness of the surface on the inner wall of the PCV.

8

2. Results of past investigations on the sources of flammable organic gases

TEPCO

■ Unit 3 PCV investigation (the state of the inner wall of the PCV) in 2017

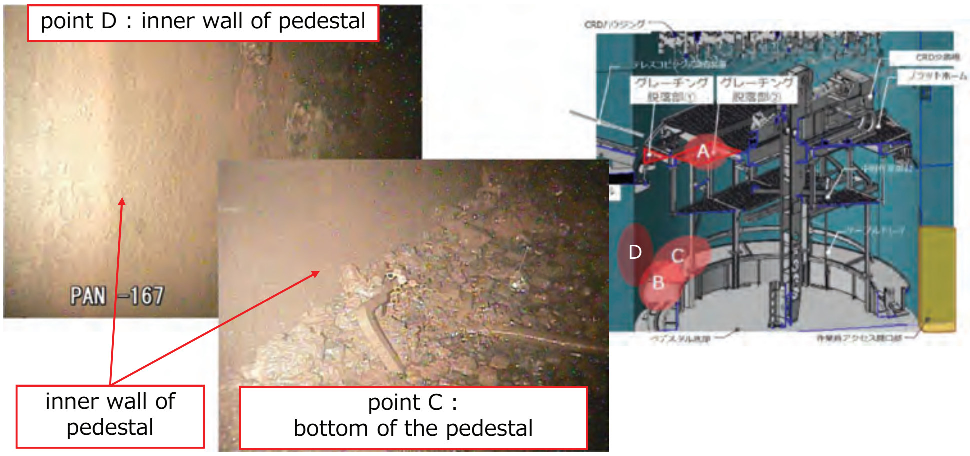
There were some peelings of the epoxy paint and roughness of the surface on the inner wall of the PCV.

9

2. Results of past investigations on the sources of flammable organic gases

TEPCO

■ Unit 2 PCV investigation (the state of the inner wall of the pedestal) in 2018



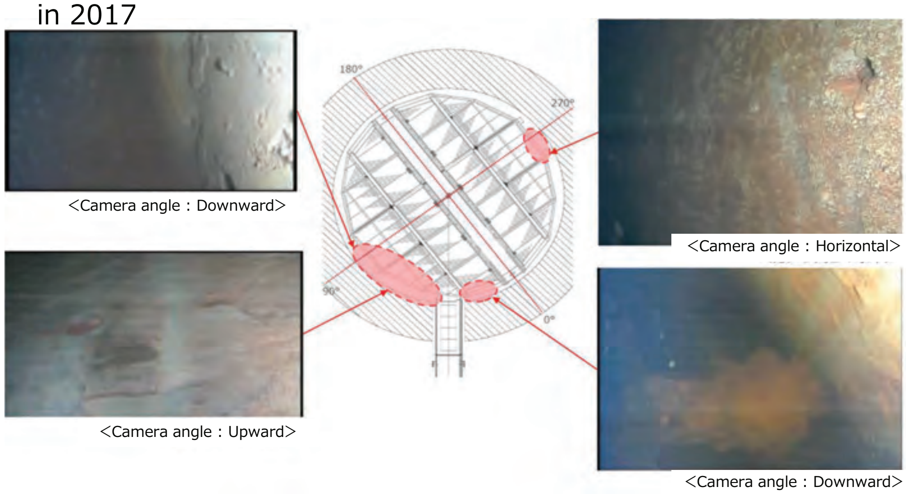
There were some peelings of the epoxy paint and roughness of the surface on the inner wall of the pedestal.

10

2. Results of past investigations on the sources of flammable organic gases

TEPCO

■ Unit 3 PCV investigation (the state of the inner wall of the pedestal) in 2017



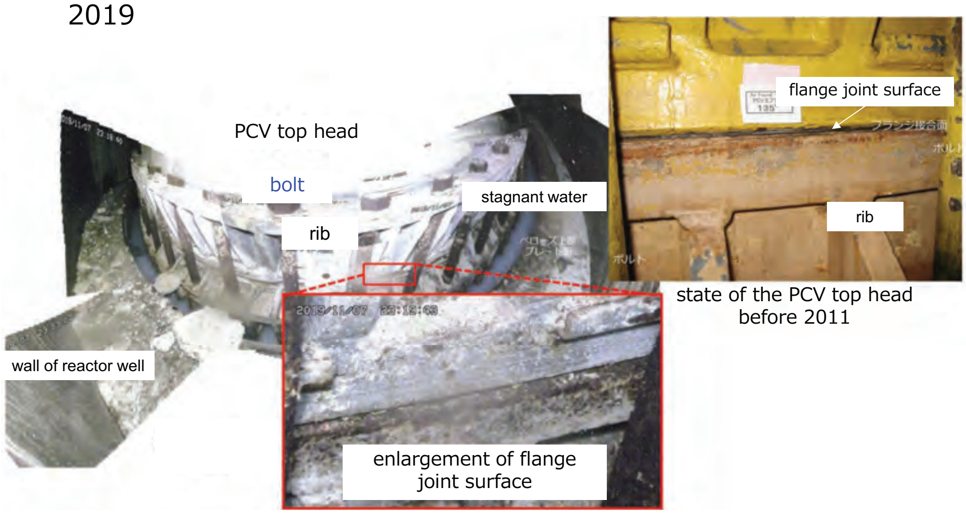
There were some peelings of the epoxy paint and roughness of the surface on the inner wall of the pedestal.

11

2. Results of past investigations on the sources of flammable organic gases

TEPCO

■ Unit 1 Reactor well investigation (the state of the PCV top head) in 2019



state of the PCV top head before 2011

Although there were some peelings of the paint on the PCV head flange, no major damage or deformation was observed there.

12

2. Results of past investigations on the sources of flammable organic gases

TEPCO

■ Unit 2 Reactor well investigation (the state of the PCV top head) in 2021

state of the PCV top head before 2011

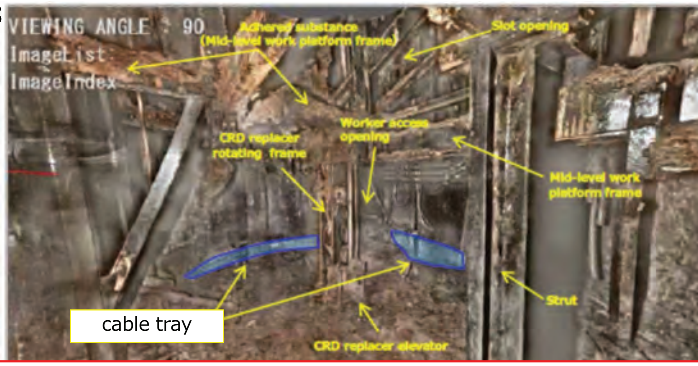
The paint on the PCV top head flange was confirmed to be peeled off.

13

2. Results of past investigations on the sources of flammable organic gases

TEPCO

■ Unit 2 PCV investigation (the state of the bottom of the pedestal) in 2018



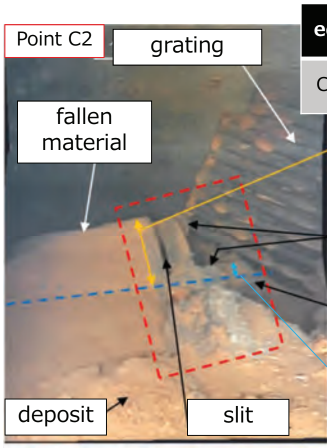
- The sediment at the bottom of the pedestal appears to be solidified molten material, but the deformation of the cable tray (stainless steel, 4 mm thick) was not confirmed.
- This suggests that the temperature of the sediment when it started to accumulate on the cable tray may not have been high enough to cause thermal deformation of the cable tray.
- Therefore, it is considered unlikely that flammable organic gas was generated by MCCI (molten core - concrete reaction) in unit 2.

14

2. Results of past investigations on the sources of flammable organic gases

TEPCO

■ Unit 3 PCV investigation (the state of the bottom of the pedestal) in 2017

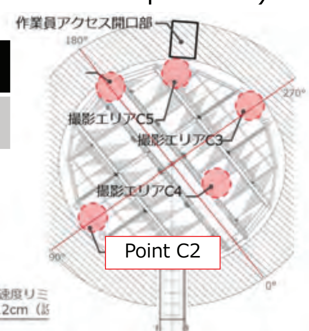


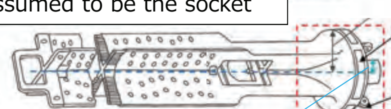
equipment	material	melting point
Control rod	SUS	Approx. 1450℃

Falling speed limiter (radius about 13cm)

shapes like a roller

cylindrical structure assumed to be the socket





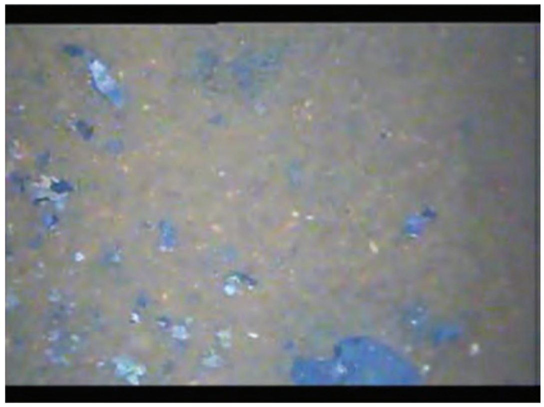
- A part of the control rod fell to the bottom of the pedestal.
- There is a possibility that flammable organic gas was generated from B_4C when the control rod melted due to contact between the molten fuel and the control rod.

15

3．The information related to the estimation of ambient temperature around RPV and PCV

TEPCO

■ Unit 1 PCV investigation (the state of the bottom of the PCV) in 2012



- Blueish fragmentary materials were observed in the sediment at the bottom of the PCV just below the X-100B penetration.
- These are assumed to be the result of the lead shielding melting, falling and solidifying.（melting point of lead：327.5℃）

16

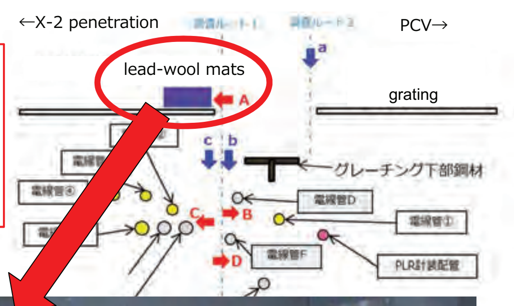
3．The information related to the estimation of ambient temperature around RPV and PCV

TEPCO

■ Unit 1 PCV investigation (the state of the inside of the PCV) in 2021

- Lead-wool mats were located near the X-2 penetration inside of the PCV.
- It is not clear whether lead remains inside of the mats.

（melting point of lead：327.5℃）



17

4. Summary

TEPCO

① The state of structures that can be sources of flammable organic gases around RPV and PCV
- ✓ Cable coating
  - ➢ Some cable bodies and cable coatings are missing in the PCVs of units 2 and 3.
- ✓ Paint (epoxy)
  - ➢ There are deterioration of paint and surface roughness on the walls of PCV in units 1-3.
- ✓ Concrete
  - ➢ MCCI is estimated not to have occurred in unit 2.
- ✓ Control rod (B_4C)
  - ➢ A part of control rod is confirmed to have melted and fallen in unit 3.

② The information related to the estimation of ambient temperature around RPV and PCV at the time of the accident
- ✓ lead shielding, lead-wool mat
  - ➢ lead shielding is estimated to have molten and fallen to the bottom of the PCV in unit 1.

18

C.1.3.3. 1F2 Reactor Well Investigations

Investigation of Unit 2 Reactor Well

TEPCO

2021/11/29

Michal Cibula

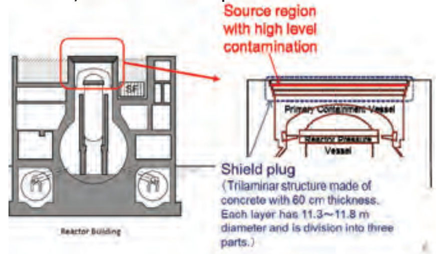
Tokyo Electric Power Company Holdings, Inc.

0. Background of this investigation

TEPCO

- NRA investigations presume massive contamination in between the shield plug layers of Units 1-3
- In particular, Unit 2 shield plug was estimated to contain several tens of PBq of Cs-137

- In 2011, AESJ established the 1F accident PIRT committee to discus about thermal hydraulics and FP behavior.
- Accumulation of FPs as deposition on the narrow part was pointed out because of the high radiation level at shield plug area in unit 2 after confirmation by robotic investigation in 2012.
- The phenomena was ranked as middle and unknown in the paper, Shoichi Suehiro, et.al., “Development of the source term PIRT based on findings during Fukushima Daiichi NPPs accident”, Nucl. Eng. Des. 286 (2015) 163-174. https://doi.org/10.1016/j.nucengdes.2015.02.005, which was published as an output of AESJ 1F acident PIRT committee.

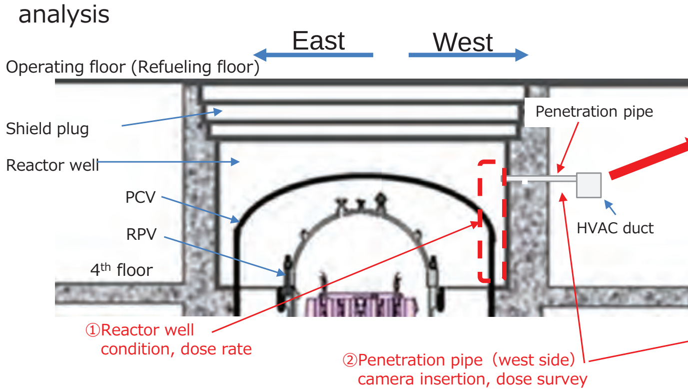


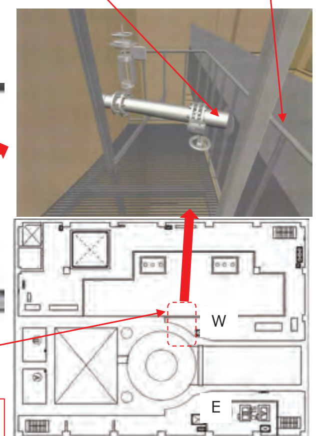
1

1. Outline of the reactor well investigation

TEPCO

- Unit 2 west side of reactor well wall investigated through the penetration pipe on 5/20,24 and 6/22 to confirm the status and dose rate inside the reactor well.
- Corrosion was observed in the HVAC duct and penetration pipe
- Sediments and duct fragments collected for analysis





R/B 4th floor

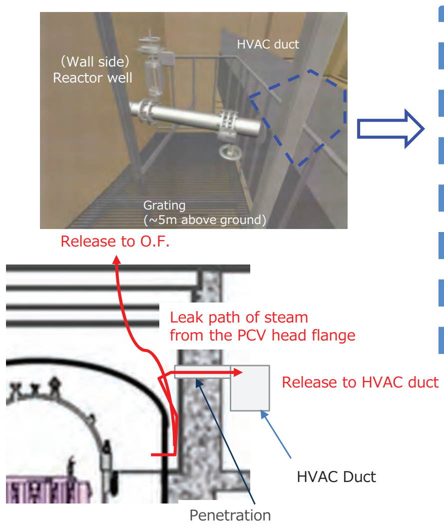
※ penetration pipe： A line to adjust the differential pressure between the reactor cavity (reactor well) and the operation floor during operation, connected to the exhaust duct of the reactor building ventilation and air conditioning system.

2

2. West Reactor cavity differential pressure adjustment line

TEPCO

- Deterioration in the part of the HVAC duct near the connection of the reactor cavity differential pressure adjustment line (other significant deterioration not observed)
- High dose rate area in the bottom of the duct and on the floor (4 m lower) below the duct



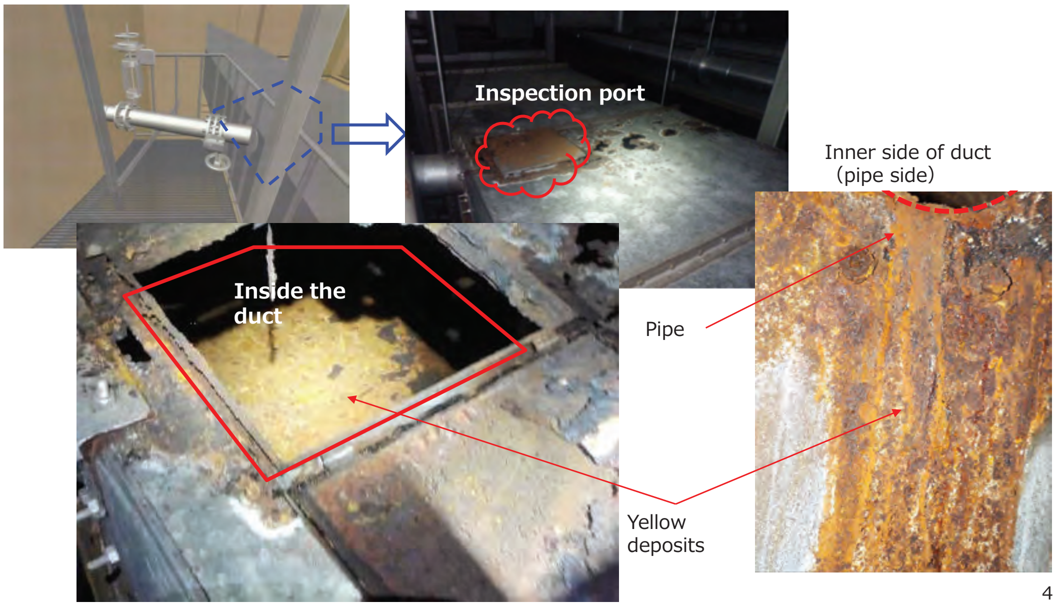
・Presumed inflow of steam containing CsOH from inside the well
・Effect of high temperature steam on galvanized plates and alkaline corrosion of ducts

3

2. West Reactor cavity differential pressure adjustment line

TEPCO

- Yellow deposits found on the inner surfaces of the duct



4

2. West Reactor cavity differential pressure adjustment line

TEPCO

- Sediments found inside the pipe

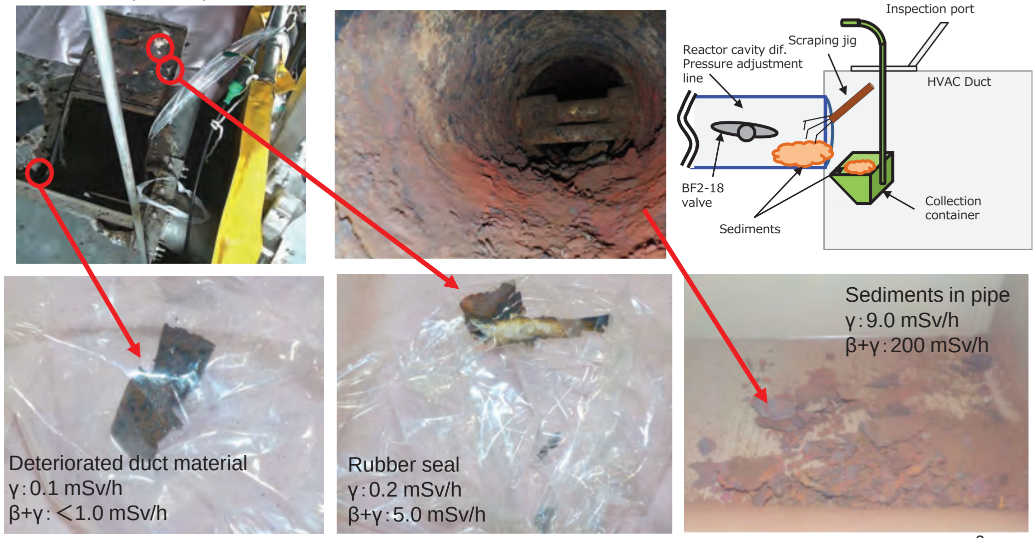
Condition inside the pipe : accumulated sediments

5

2. West Reactor cavity differential pressure adjustment line

TEPCO

- For the accident investigation, samples were collected at the following locations
  - Deteriorated duct material
  - Rubber seal on the back of the duct inspection port
  - Sediments in pipe
- Preliminary analyses confirmed Cs-134, Cs-137, Co-60, Sb-125, Sr-90

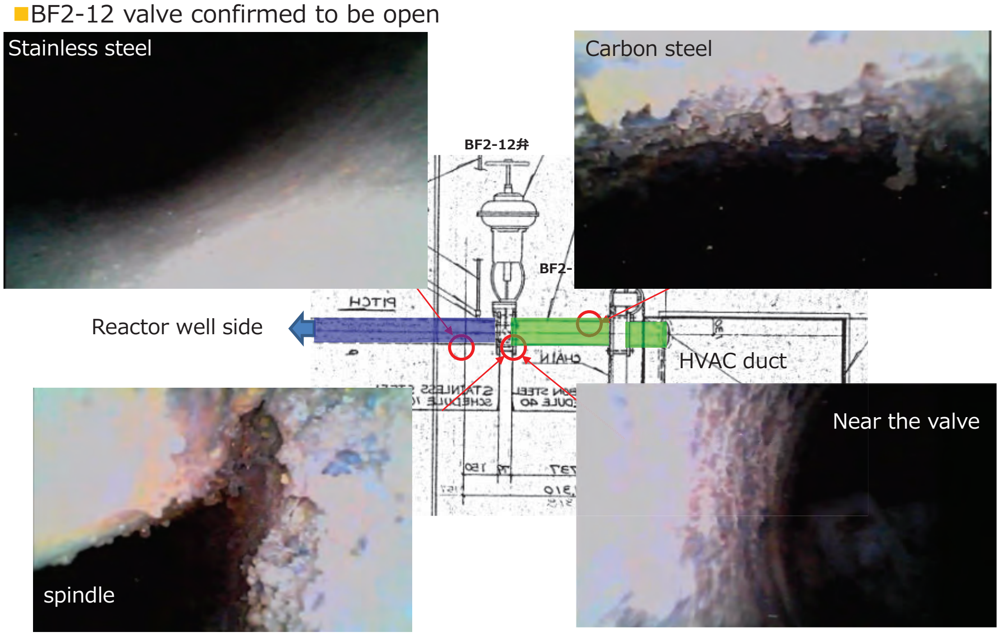


6

2. West Reactor cavity differential pressure adjustment line

TEPCO

- The piping (SUS) upstream of the BF2-12 valve did not show any skin roughness or deposits that were present in the piping and valve box (carbon steel).
- BF2-12 valve confirmed to be open



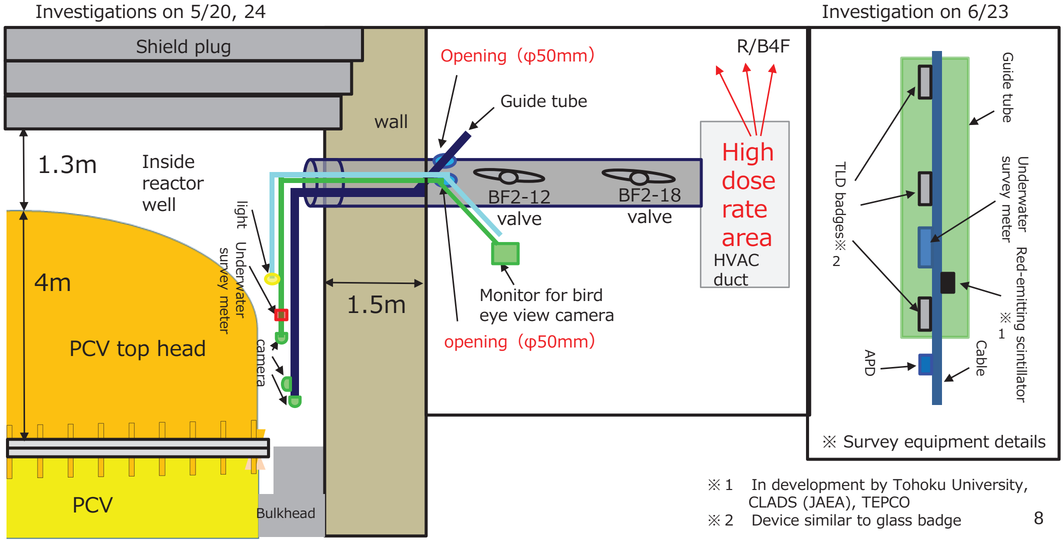
7

3. Reactor well investigation

TEPCO

- Survey meters and cameras inserted into the reactor well through the penetration
- Due to high dose rate in the vicinity of HVAC, investigation conducted through newly perforated opening in the pipe near the wall
- Initial surveys conducted by underwater survey meter (5/20, 24)
- Additional survey utilizing various types of equipment conducted on 6/23

Investigations on 5/20, 24 Investigation on 6/23

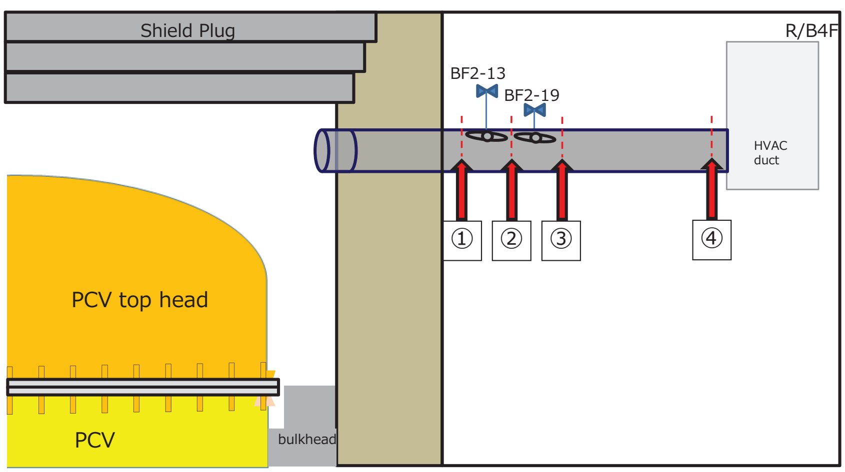


※1 In development by Tohoku University, CLADS (JAEA), TEPCO
※2 Device similar to glass badge

3. Reactor well investigation (dose rates)

TEPCO

- The results of the dose rate measurements are as follows.
  - The maximum dose rate (51 mSv/h) at the bottom of the pipe (measurement point 2)

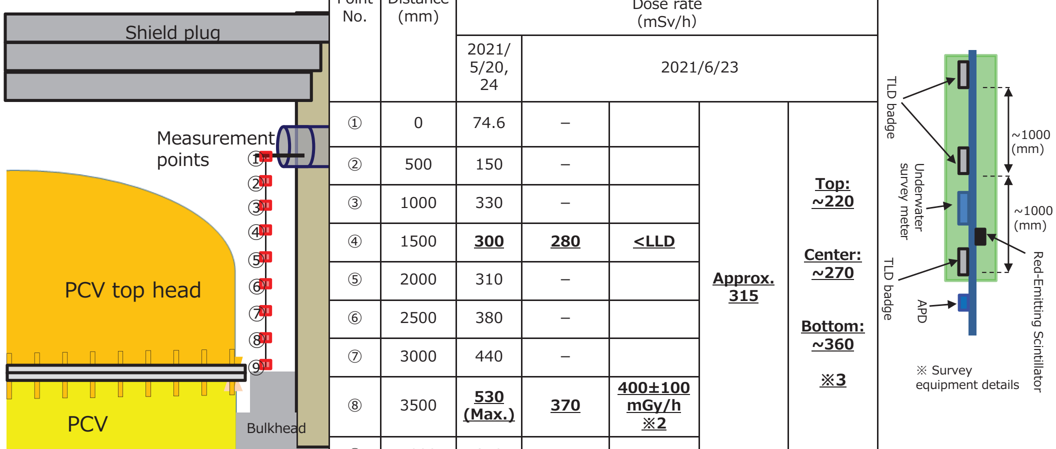


Measured points		①	②	③	④
pipe	upper(mSv/h)	13	41	25	18
	lower(mSv/h)	13	51	37	20

3. Reactor well investigation (dose rates)

TEPCO

- Initial survey showed maximum dose of 530 mSv/h near the PCV flange.
- Measured values were lower than expected. Cf. max. on shield plug was ～560mSv/h
- Re-measurement results by several dosimeters also showed same tendency.



Point No.	Distance (mm)	Dosimeter 1		Red-emitting scintillator	APD	TLD badge
		Dose rate (mSv/h)				
		2021/5/20, 24	2021/6/23			
①	0	74.6	–			
②	500	150	–			
③	1000	330	–			
④	1500	**300**	**280**	**<LLD**		
⑤	2000	310	–		**Approx. 315**	**Top: ~220** **Center: ~270** **Bottom: ~360** **※3**
⑥	2500	380	–			
⑦	3000	440	–			
⑧	3500	**530 (Max.)**	**370**	**400±100 mGy/h ※2**		
⑨	4000	350	–			

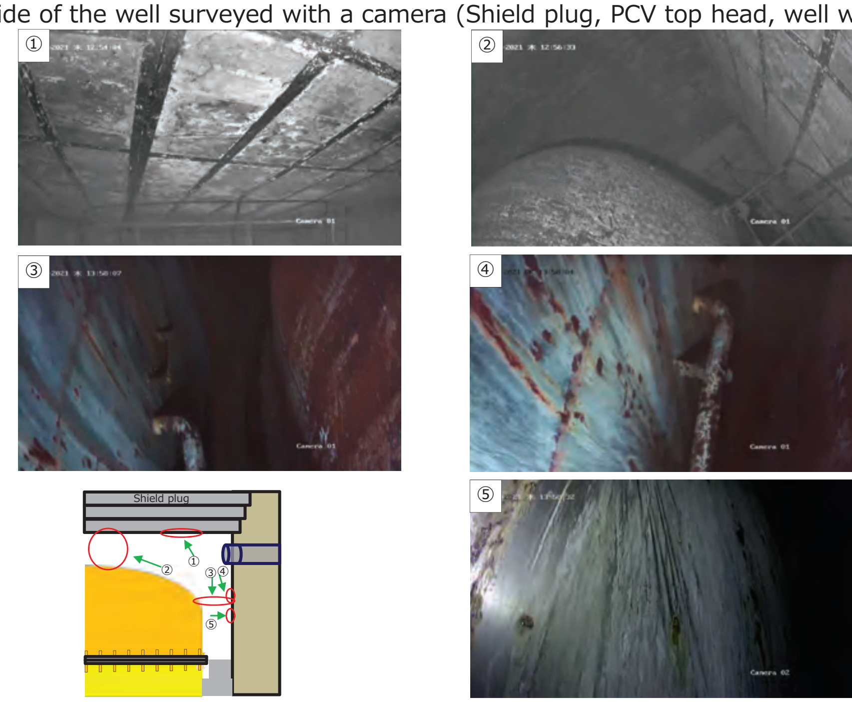
※1 Calibrated for Co-60
※2 Red-emitting Scintillators units are mGy/h
※3 Converted from cumulative dose considering the survey time in well

10

4. Reactor well investigation (visual)

TEPCO

- Inside of the well surveyed with a camera (Shield plug, PCV top head, well wall, etc.)



11

(Picture) Shield plug bottom surface (1/3)

TEPCO

(Picture) Shield plug bottom surface (2/3)

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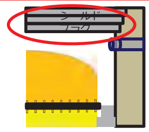
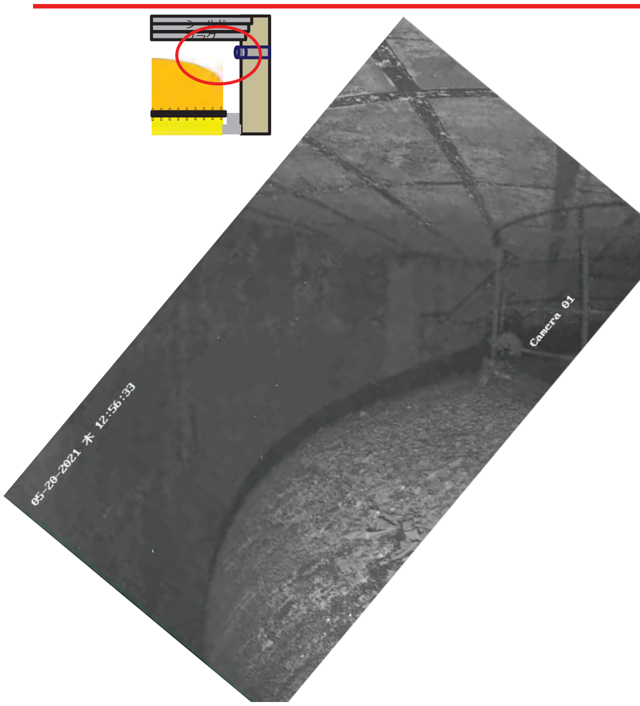
13

(Picture) Shield plug bottom surface (3/3)

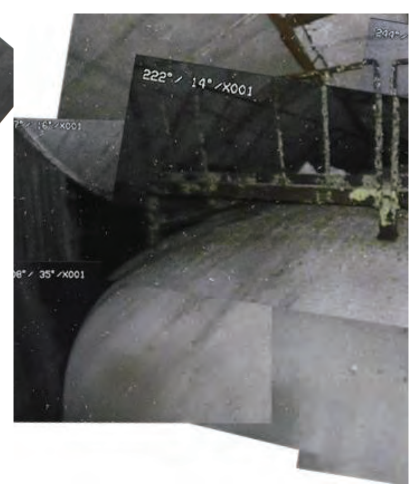


14

(Picture) PCV top head and bottom of shield plug (3/3)



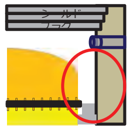
Ref.: Unit 1 PCV top head



15

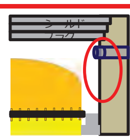
(Picture) Well wall and PCV top head

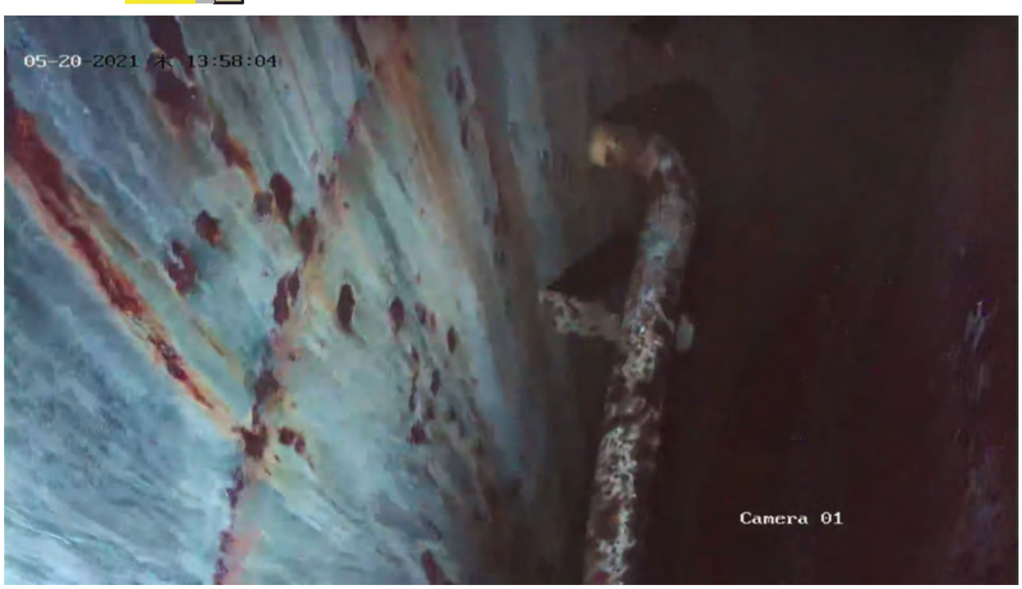
TEPCO



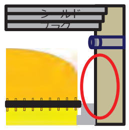
(Picture) Well wall (1/3)

TEPCO

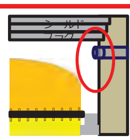


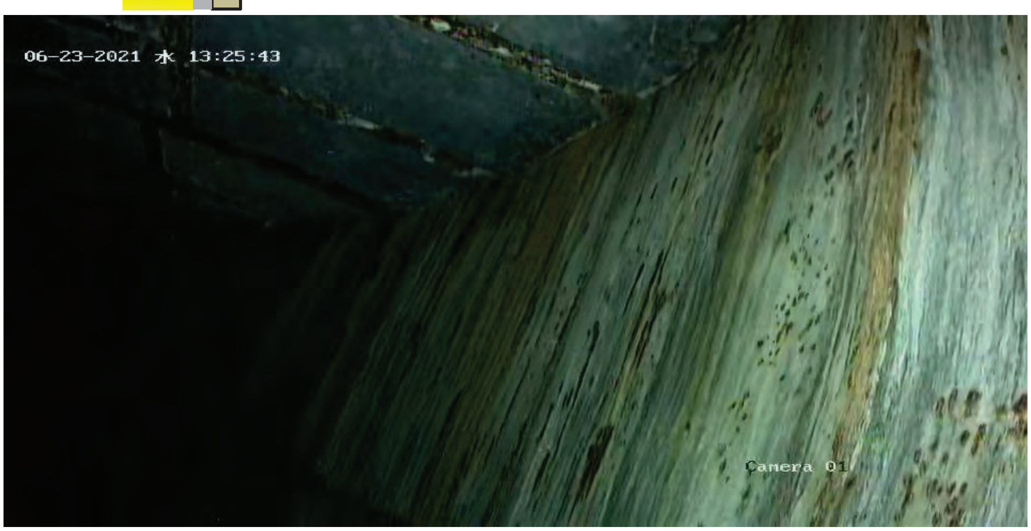


(Picture) Well wall (2/3)



(Picture) Well wall (3/3)

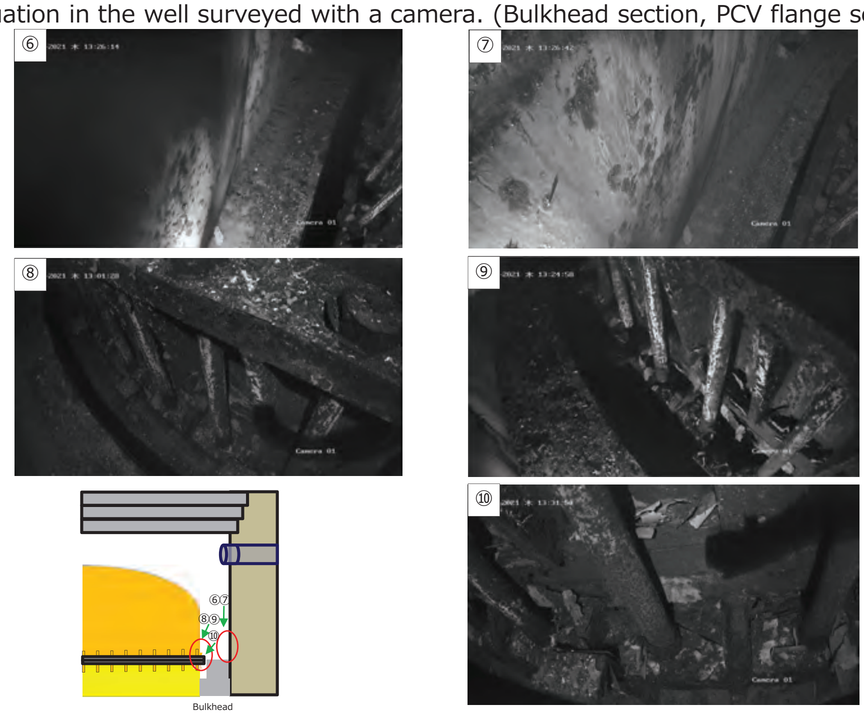




4. Reactor well investigation (visual)

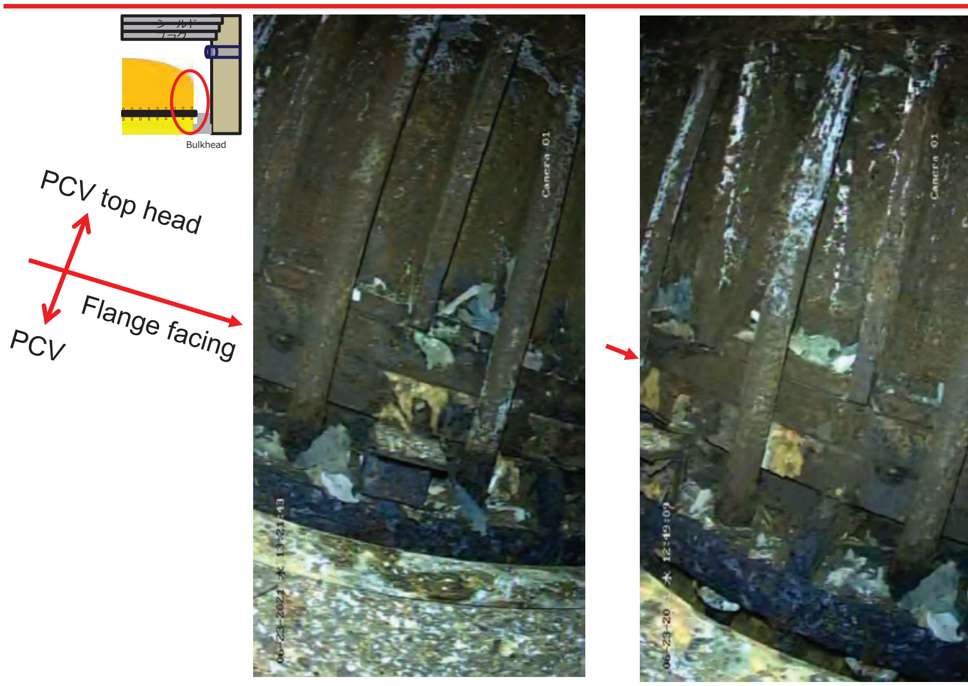
TEPCO

■ Situation in the well surveyed with a camera. (Bulkhead section, PCV flange section)



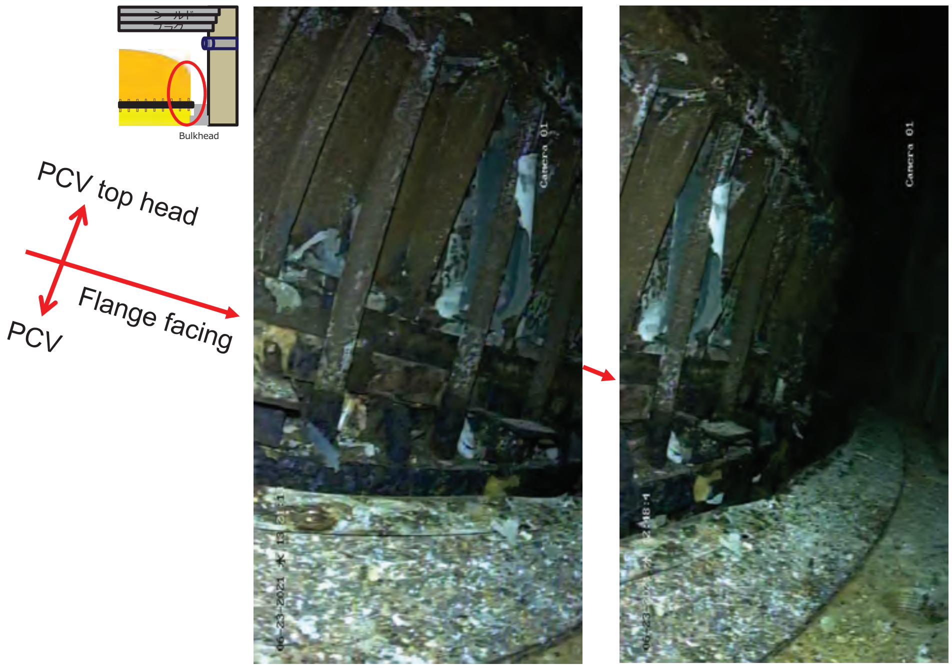
(Picture) PCV flange (1/2)

TEPCO



(Picture) PCV flange (2/2)

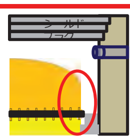
TEPCO

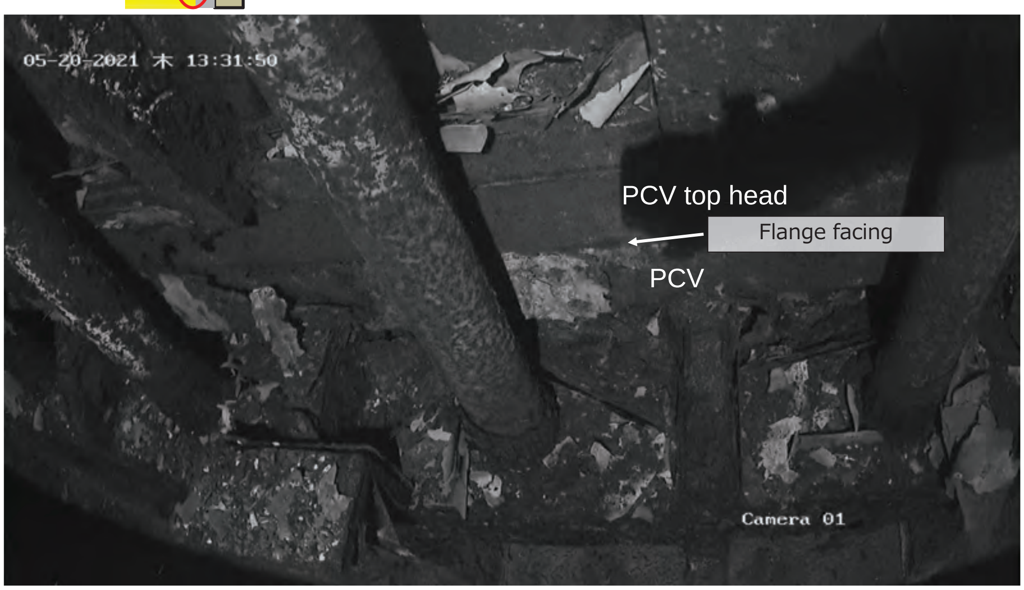


22

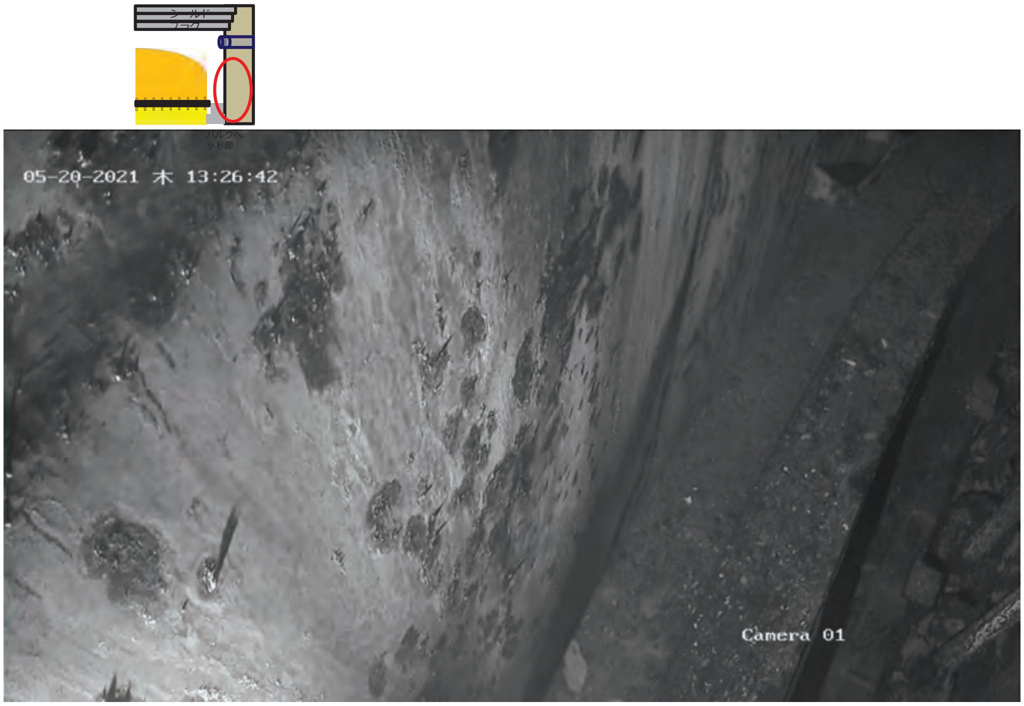
(Picture) PCV flange joint

TEPCO

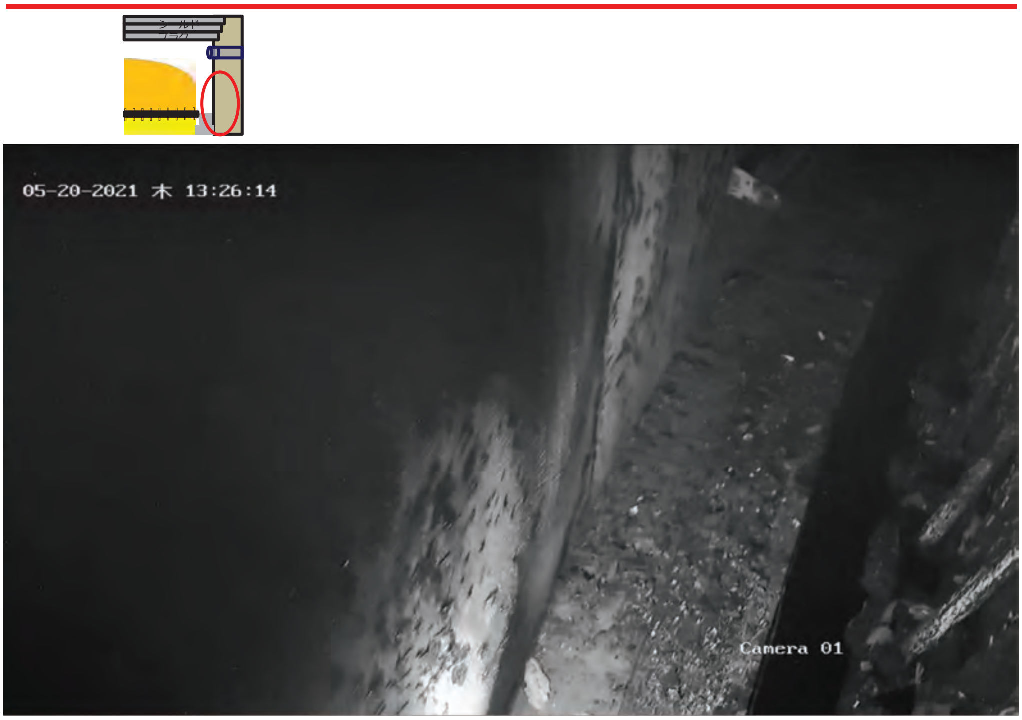




(Picture) Well wall and bulkhead section (1/3)

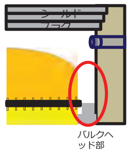


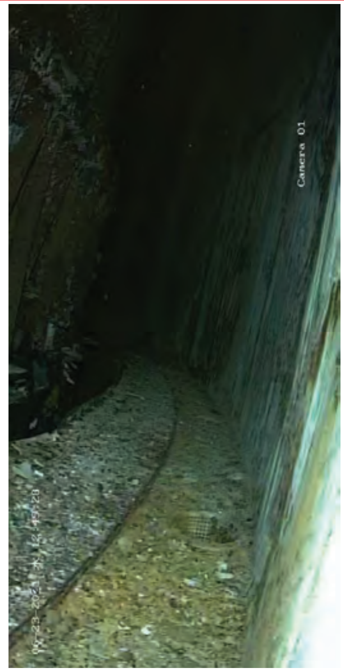
(Picture) Well wall and bulkhead section (2/3)



(Picture) Well wall and bulkhead section (3/3)

TEPCO





5.1 The condition of shield plug at the operation floor

TEPCO

- Unit 2 Shield plug was covered by curing sheet.
- There is a possibility that contamination is top heavy due to the cover sheet.

5.2 Unit 2 operation floor video 6 months after accident

http://www.tepco.co.jp/en/news/library/archive-e.html?video_uuid=ce7g4s6v&catid=61793 28

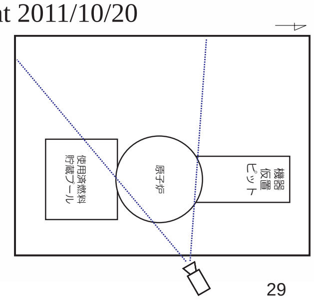
5.3 Humidity conditions on operating floor after accident

Steam can be seen at 2011/09/17

Paint extremely deteriorated at 2011/10/20

Steam discharge from PCV top head stopped after CS injection started on Sep. 14 because of temperature decrease

No steam discharge from October, O.F. in dry condition (Paint chipping as evidence)



29

5.4 Consequences of the SA and current conditions of 1F

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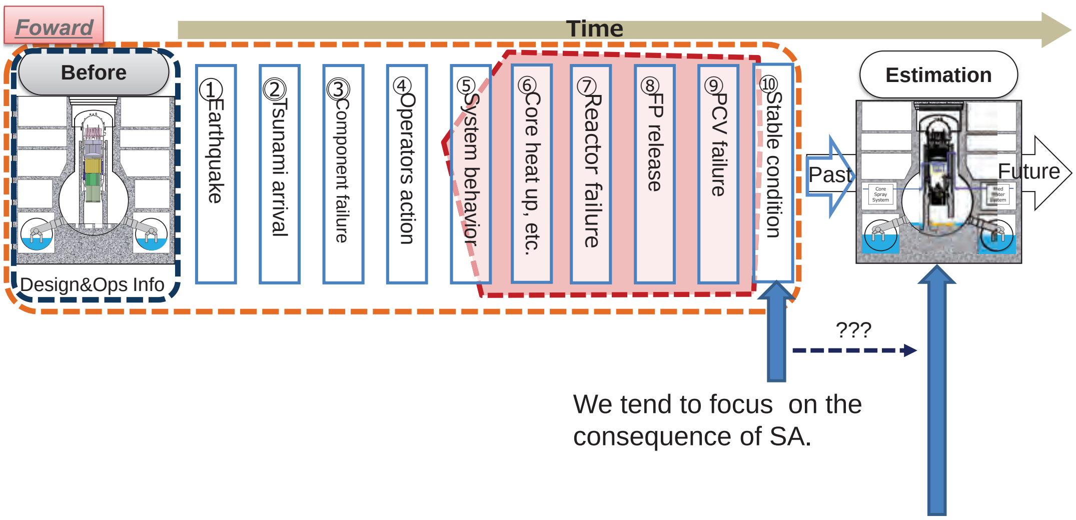


We tend to focus on the consequence of SA.

However, investigation results which derived from 1F plants included additional 10 year events after the accident.

TEPCO

Thank you for your attention

31

(Picture) 4th floor east side on grating

TEPCO

32

(Picture) Duct condition on the east side of the 4th floor

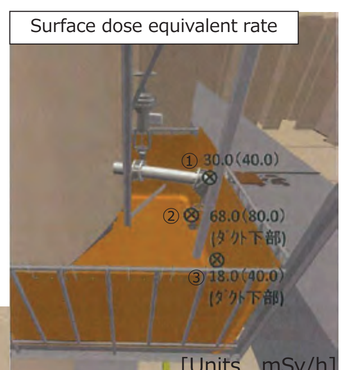
TEPCO

33

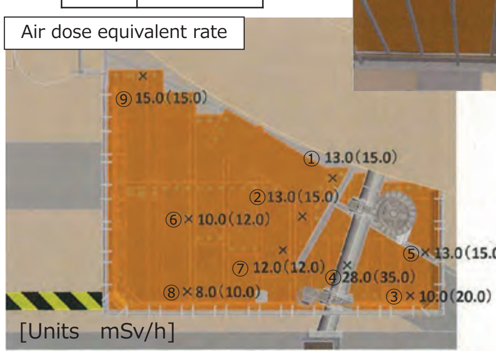
(Picture) 4th floor east side piping and duct connections

(Ref.)Dose rate in the work area and exposure dose to workers

Point	Air dose eq. rate [mSv/h]
①	13
②	13
③	10
④	28
⑤	13
⑥	10
⑦	12
⑧	8
⑨	15



Point	Surface dose eq. rate [mSv/h]
①	30
②	68
③	18

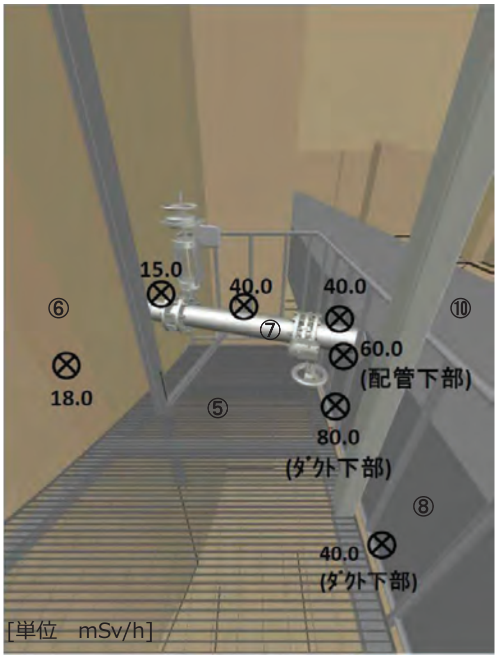


■ Exposure dose to workers

Date	Work contents	Exposure dose (mSv)		
		Total	Personal max.	Average
5/19	Pipe drilling, investigation preparation	24.58	1.86	1.54 (16 people)
5/20	Well investigation	25.10	1.82	1.48 (17 people)
5/24	Well investigation, clean up	11.52	1.93	1.28 (9 people)

(Ref.) Survey data (reactor cavity differential pressure adjustment line (west side))

TEPCO



Measurement date : 2021/3/5

Used equipment

ICW,ICWBL,ICWBH,GMAD,α

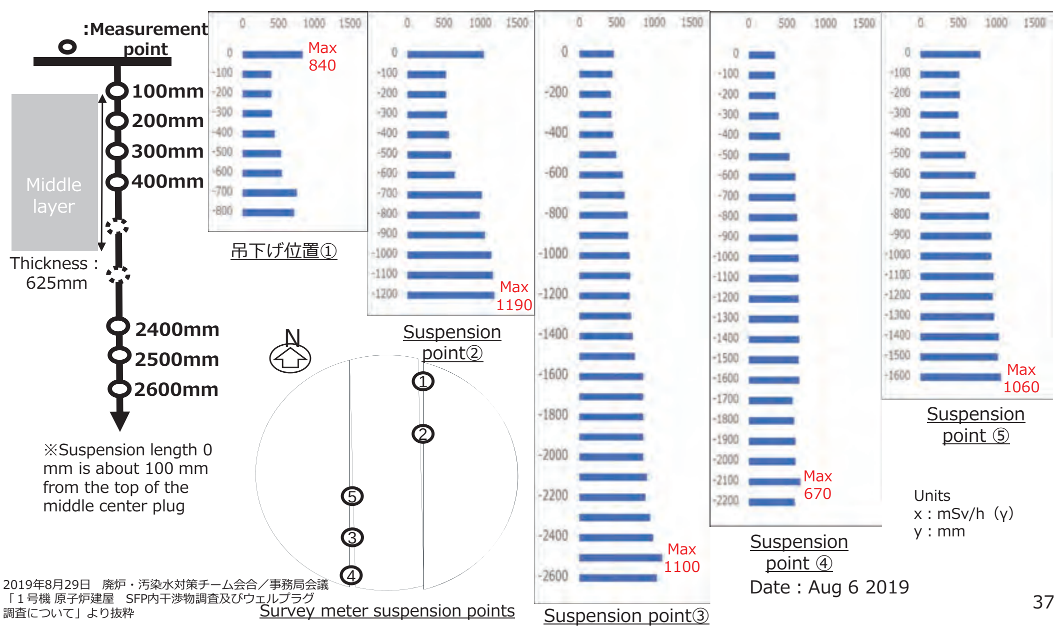
Smear collection point	β(cpm)	α(cpm)	γ(mSv/h)	β+γ(mSv/h)
⑤	>100000	0	0.15	10.0
⑥	>100000	30	0.14	5.0
⑦	>100000	50	0.16	12.0
⑧	>100000	0	0.15	8.0
⑩	>100000	0	0.14	7.0

36

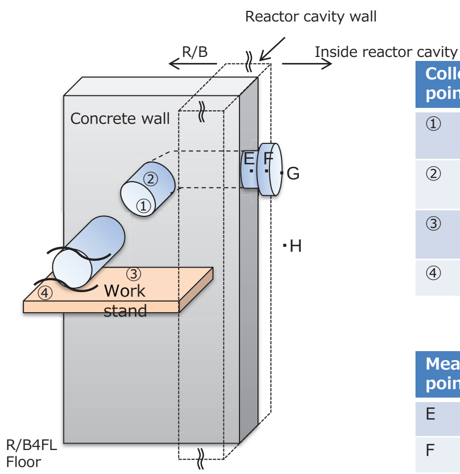
Dose rate measurements under the middle plug of Unit 1

TEPCO

- As a result of survey, it was confirmed that each measurement position tended to be higher below the middle plug



37



Smear collection/dose rate measurement points of Reactor cavity differential pressure adjustment pipe

Smear analysis (Bq/cm²)

Collection point	Alpha emitters	B+γ emitters	Reference
①	1.1×10	1.8×10^2	Inside pipe (near cutting)
②	3.5×10	$>2.6\times10^2$	Inside pipe (near elbow)
③	<LOD	$>2.6\times10^2$	Work stand (Below pipe)
④	<LOD	$>2.6\times10^2$	Work stand floor

Dose rate measurements near the pipe outlet

Measurement point	Dose rate [mSv/h]	Reference
E	400	10 cm from pipe outlet (in pipe)
F	700	10 cm from pipe outlet (in pipe)
G	1100	Pipe outlet
H	2600	50 cm below pipe outlet

C.1.3.4. Mid-and-Long-Term Plan for 1F Investigations

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

TEPCO

2021/11/29

Michal Cibula

Tokyo Electric Power Company Holdings, Inc.

Background

TEPCO

- **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 independently** proceed with the **investigation** in the future

1

Outline

TEPCO

The plan is formulated and implemented as follows

- 4 categories:
Immediate (~half 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

Unit 1

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

Document 1

			~half year	~2023	~2032	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	・PCV leak location investigation (RB-10) ・Stagnant water analysis (TRB-1)
		Nuclide analysis of indoor samples (RB-5,9a,14)		Implemented as appropriate		
		Investigations of penetrations for electrical conduits etc. (RB-9b)	Unit 1 PCV internal investigation Unit 1 R/B upper floors investigation	▼ Unit 1 PCV internal investigation		
	O.F. ☆	※Included in general bldg. work			▼ Unit 1 rubble removal ▼ Unit 1 O.F. decontamination/shielding	
	Torus room					
	Specific equipment /system	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			
T/B, yard	-	Vent line, SGTS investigation (RB-11) ☆	Unit 1 R/B upper floors investigation			・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)	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 ▼ Unit 1 PCV internal investigaton		
	Specific equipment/ system	PLR investigation (PC-4,11)	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 investigaton		
	RPV itself, peripheral piping	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				

3

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.

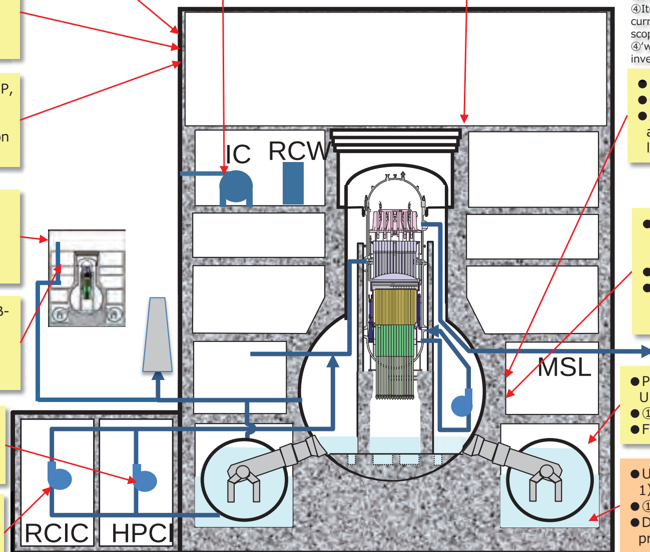
- ●U4 HVAC duct RN analysis (RB-6)
- ●①④'
- ●U4 R/B explosion cause (H_2 source) clarification.

- ●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

4

Unit 2

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

Document 1

			~half year	~2023	~2032	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 RPV internal investigation	
	O.F. ☆	O.F. investigation (RB-4) ※Included in general bldg. work	Unit 2 shield plug investigation	▼ Unit 2 FHM remote control room dismantling/existing objects removal; ▼ Unit 2 O.F. decontamination, shielding		
	Torus room	PCV leak location investigation (RB-10)		▼ Unit 2 R/B water level lowering		
		Torus room investigation (TRB-8)		▼ Unit 2 R/B water level lowering		
	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			
T/B, yard	-	Vent line, SGTS investigation (RB-11) ☆	Unit 2 R/B upper floors investigation			· 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	Genaral	PCV liner, pedestal, etc. state confirmation, FP nuclide analysis (PC-3,9,10,16)		Unit 2 PCV internal investigation ▼ Unit 2 PCV internal investigation		
	Debris, sediments	Investigation of FD, sediments, etc. (PC-3,15,22)		Unit 2 trial debris retrieval, PCV internal investigation ▼ Unit 2 trial debris retrieval, property analysis	Unit 2 small scale debris retrieval ▼ Unit 2 debris retrieval, property analysis	
	Specific equipment/ systems	RPV instrumentation investigation (PC-7,8)		Unit 2 PCV internal investigation ▼ Unit 2 PCV internal investigation		· PLR investigation (PC-4,11) ☆ · MSL、SRV investigation (PC-5,6) · Conduit cables/liner survey (PC-14)
	RPV itself, peripheral piping	RPV, peripheral piping state confirmation (PC-3,12,13) ☆		Unit 2 PCV internal investigation ▼ Unit 2 PCV internal investigation		
RPV	-	※Consideration of the content and timing of the investigation based on the progress of D&D				

5

Unit 3

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

Document 1

			～half year	～2023	～2032	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)
		Nuclide analysis of indoor samples （RB-5,9a,14）	Implemented as appropriate			
		Stagnant water analysis (TRB-1)	Unit 3 MSIV room stagnant water detailed analysis	▼ Unit 3 MSIV room stagnant water detailed analysis		
	O.F. ☆	※Included in general bldg. work				
	Torus room					
	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) ☆
		AC investigation (TRB-6)		Unit 3 R/B upper floors investigation		
T/B, yard	-	Vent line, SGTS investigation (RB-11) ☆	SGTS filter nuclide analysis	Unit 3 R/B upper floors investigation ▼ 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)
	Debris, sediments	Unit 3 PCV internal investigation (PC-21)			▼ Unit 3 PCV internal investigation	デブリ、堆積物等の調査（PC-3,15,22）
	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)
	RPV itself, peripheral piping ☆	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				

6

Unit 4

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

Document 1

			～half year	～2023	～2032	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)		SGTS filters nuclide analysis		AC investigation (TRB-6)
T/B, yard	-			▼ Units 3/4 exhaust stack removal		

7

TEPCO

Thank you for your attention

Important input information ①

TEPCO

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

No.	D&D work content	Implementation	Target area
1	Unit 1 PCV Internal Investigation	Immediate, short term	Whole PCV
2	Unit 1 indoor/outdoor environment improvement (Dose rate reduction/interferences removal, etc.)	Immediate, 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 2 R/B water level lowering (lowering below the S/C RCIC nozzle)	Immediate, short term	R/B torus room
6	Unit 2 O.F. decontamination/shielding	Short term	R/B O.F.
7	Unit 2 building environment improvement (Dose rate reduction/interferences removal, etc.)	Immediate, medium term	Whole R/B
8	Unit 2 FHM remote control room dismantling/existing objects removal	Short term	R/B O.F.
9	Unit 2 trial debris retrieval/internal investigation	Short term	PCV debris, Whole PCV
10	Unit 2 debris property analysis (At the time of trial retrieval)	Short term	PCV debris

Important input information ①

TEPCO

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

No.	D&D work content	Implementation	Target area
11	Unit 2 debris retrieval (Gradual expansion of retrieval scope)	Medium term	PCV debris
12	Unit 2 debris property analysis (At the time of gradual expansion of retrieval scope)	Medium term	PCV debris
13	Unit 3 MSIV room stagnant water detailed analysis	Immediate, short term	Whole R/B
14	Unit 3 indoor/outdoor environment improvement (Dose rate reduction/interferences removal, etc.)	Medium term	Whole R/B, Yard
15	Unit 3 PCV internal investigation	Medium term	Whole PCV
16	Units 1/2 bottom part of exhaust stack removal	Immediate, medium term	Yard
17	Units 3/4 exhaust stack removal	Shor term	Yard

10

Important input information ②

TEPCO

○Needs of internal/external stakeholders

No.	Organization	Site investigation item	Implementation	Target area
1	NRA	Units 1/2 SGTS filter train, AC system contamination	Immediate	R/B specific equipment/system
2	〃	Dose rate measurement of vent lines, SGTS, exhaust stack, etc.	Immediate	R/B specific equipment/system
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)	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	NRA／TEPCO	Experiment to confirm generation of organic compounds by heating of cables, etc. (Confirmation of In-PCV combustible gases generation)	Immediate	R/B specific equipment/system
8	TEPCO	Unit 1 R/B 2FL on-site reactor pressure gauge soundness	(TBD)	R/B specific equipment/system
9	〃	Units 1~3 SRV state Confirmation	(TBD)	R/B specific equipment/system
10	〃	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
11	〃	Possibility of leaks from RPV flanges	(TBD)	PCV RPV body

11

Important input information ③

TEPCO

○ Our external commitments

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

12

R/B-2

Ref.

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

● U1~4 AC piping contamination/rust related investigation, dose survey, photos/videos (TRB-6)
● ④④'
● Deepening understanding of accident progression, effects of venting, FP behavior

● U2 vent line rupture disc (non)rupture investigation, photos/videos (TRB-3)
● ④'
● Deepening understanding of accident progression, success/failure of venting

● U4 SGTS filters RN analysis (TRB-4)
● ④'
● Deepening understanding of accident progression, composition of FP in vent

● Dose survey and FP analysis of U1/2 exhaust stack base high dose rate area (TRB-7)
● ②
● Deepening understanding of accident progression, composition of FP in vent

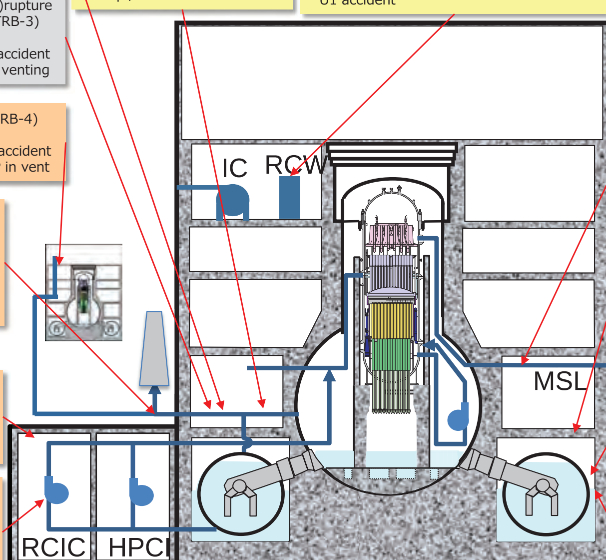
● Photos/video, RN analysis of U2 RCIC room upper wall surface (TRB-2)
● ④
● FP leak path clarification

● U2 S/C liquid leakage point investigation(RCIC room, etc.) (TRB-5)
● ④
● PCV leakage point clarification

● Photos/video, dose surveys related to PCV hardpipe venting pathway, SGTS and related R/B ventilation system (RB-11)
● ①④④'
● Evaluation of sealing performance in high temp./radiation conditions

● Information related to SW piping soundness/exterior photos to confirm mechanical soundness (TRB-10)
● ④
● Deepening understanding of accident progression

● U1 RCW surge tank investigation, water level, dose rate measurement (RB-15)
● ②④'
● Determining the role of the RCW during U1 accident



● Information related to instrumentation soundness
・exterior photos to confirm mechanical soundness
・electrical inspections results (including records after the disaster) (TRB-9)
● ①④
● Deepening understanding of accident progression

● Photos/videos of MSL piping outside of PCV (RB-13)
● ④①
● Identification of PCV failure mode

● Chem. analyses of high radiation deposits/particles found in the R/B (RB-14)
● ②④①
● Contribution to debris location estimation related to materials generated by MCCI

● Photos/videos of the vicinity of leakage points in R/B (RB-12)
● ①
● Clarification of the reason for leakage from the R/B into the T/B (investigation of U2 PCV leakage point)

● U2 torus room inundation marks (TRB-8)
● ④
● Deepening understanding of accident progression

Gray color : Investigation conducted, information acquired

13

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

- 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

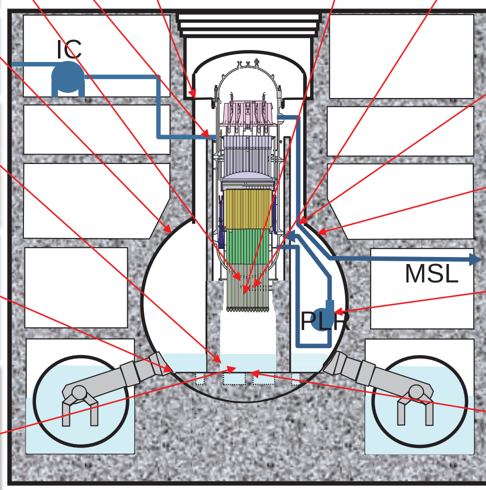
- Photos/videos of RPV lower head, structures, penetrations (PC-3e)
- ①②
- Damaged around the bottom of RPV, code evaluation of corium behavior, model improvement

- 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.



- 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

14

原子炉格納容器-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

- 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

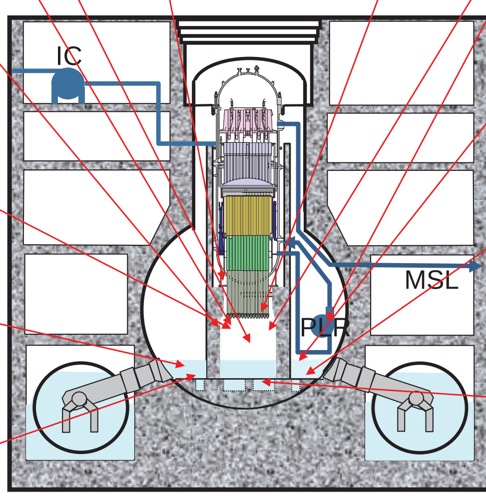
- Samples of conduit cabling and paint for RN analyses (PC-14)
- ③ (②)
- Dose code assessments, model improvements

- 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 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

15

Gray color : Investigation conducted, information acquired

RPV

Ref.

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

- 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.

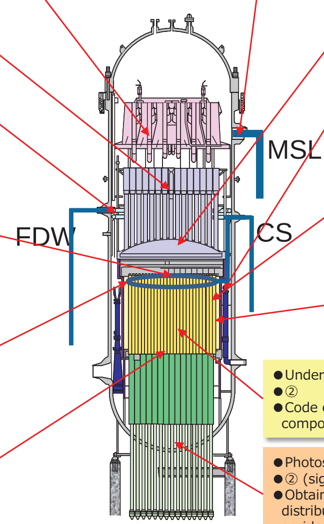
- Photos/videos, sampling of separator (RPV-3)
- ② (sampling③)
- Code evaluation and model improvement based on findings from soundness or displacement measurement, peak temperature evaluation, and metallurgical testing

- Photos/videos, 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)

16

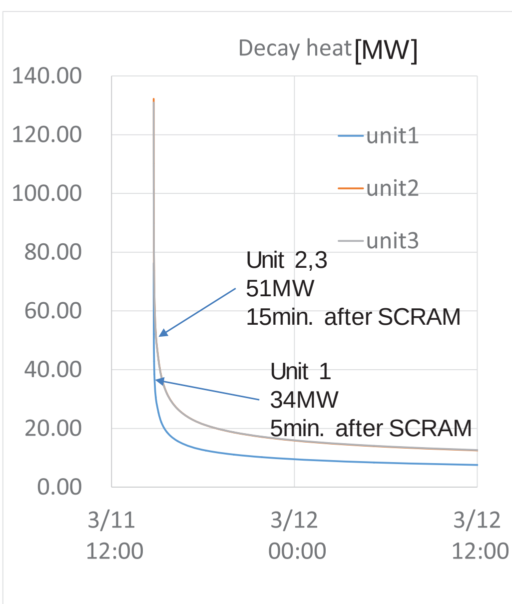
C.1.3.5. Long Term Cooling Insights

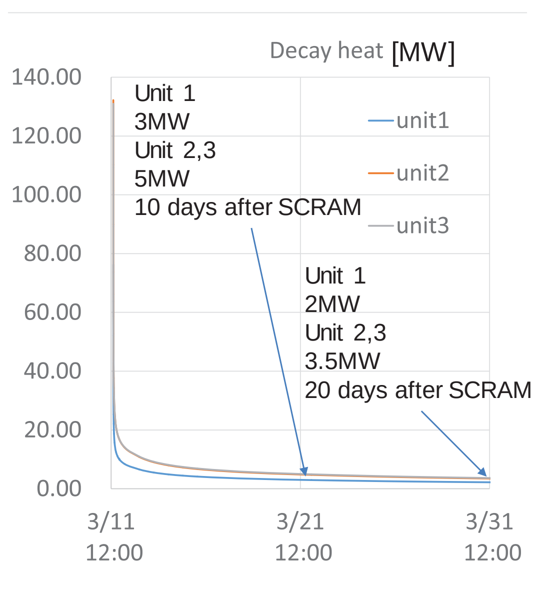
Lessons learned from 1F accident by focusing on the long term cooling

TEPCO HD
S. Mizokami

2021/10/20

1. Decay heat history since 3.11

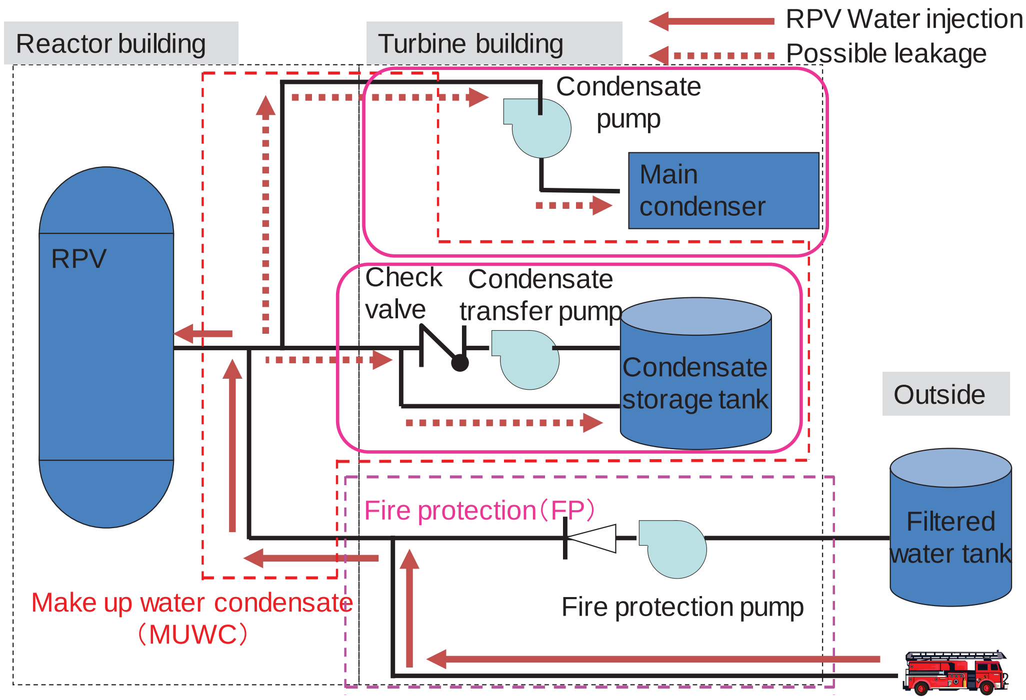




1

4. ALTERNATIVE WATER INJECTION

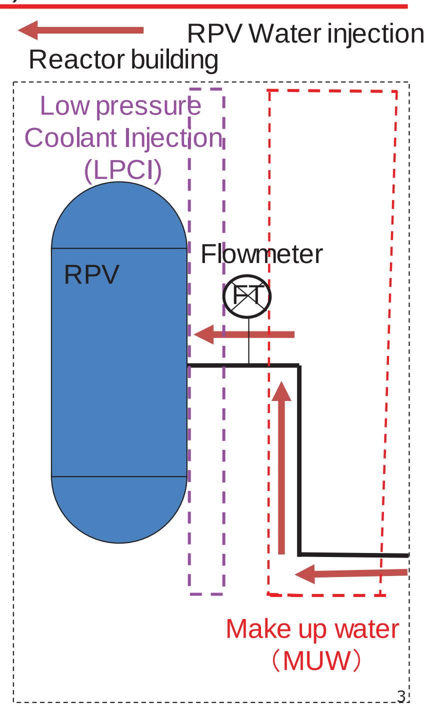
TEPCO



4. ALTERNATIVE WATER INJECTION (AWI)

TEPCO

- Water leakage to other equipment connected to MUW was recognized as a concern.
- Therefore, flowmeter was installed in between MUW and LPCI. Because the water reached to LPCI can be injected to RPV.
- The operating procedure require to confirm the flow rate is no less than 55 m^3/h, which corresponds to about 30MW.
- However, the range of this flowmeter was 20 to 200 m^3/h. If flow rate was less than 20m^3/h, the flowmeter indicate 0 due to low cut filter.
- Decay heat at end of March was 2 to 4 MW. In case of such low decay heat, the flowmeter can not measure the flow rate corresponding to the water to remove the decay heat.
- The flowmeter was the only instrument which can tell us correct value of injected water to RPV. The information was totally lost.



Water injection reported during accident

TEPCO

YYYY/MM/DD	Water injection per day (unit 1)
2011/03/19	449m³/day
2011/03/20	48m³/day ➡ 2m³/h
2011/03/21	38m³/day ➡ 1.6m³/h
2011/03/22	42m³/day ➡ 1.75m³/h
2011/03/23	301m³/day

The amount of water injection reported from 3/20 to 3/22 was less than mensurable range, 20 m^3/h to 200 m^3/h.

4

TEPCO

Thank you for your kind attention.

5

TEPCO

Original presentation

6

1. Introduction

TEPCO

- On 2011/3/11, Fukushima Daiichi NPS was attacked by huge earthquake and successive tsunami. Then 1F 1 to 3 finally lost both of AC and DC.
- The existing reactor cooling systems were IC , RCIC and/or HPCI.
- Even after the core meltdown, TEPCO tried to cool the debris by using fire trucks through FP and MUWC systems.
- These cooling systems were used to remove decay heat. The decay heat monotonically decrease and the variation range is 10^1 to 10^0 MW.
- Plant behaviors in each units were affected by loss of electric power and also the decay heat history.

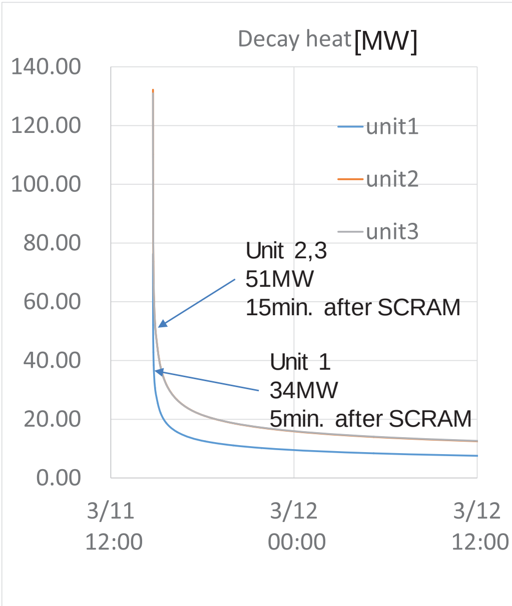
- The accident progressions in units 1-3 are divided into three phases from the viewpoint of reactor cooling:
  i: Reactor cooling just after Earthquake before Tsunami arrival
    The operator's action was normal operation when reactor scram occurred.
  ii: Reactor cooling without AC/DC power before core damage
    There are no operator's actions or operations without accurate recognition of reactor condition.
  iii: Debris cooling by using alternative water injection line
    This time period started in the middle of March and ended in December.

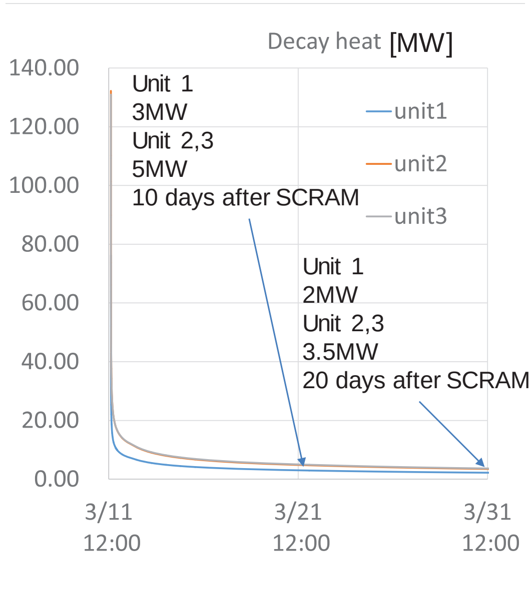
- In this paper, I will show you the lessons learned from 1F accident in each accident phase.

7

1. Decay heat history since 3.11

TEPCO





8

1. Introduction

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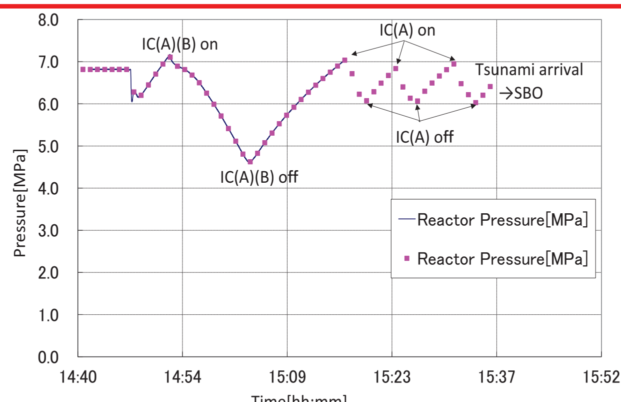
9

i: Reactor cooling just after Earthquake before Tsunami arrival

The operator's action was normal operation when reactor scram occurred.

2. REACTOR COOLING JUST AFTER EARTHQUAKE

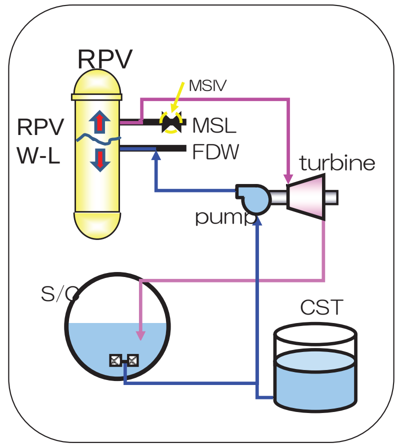


- Decay heat decrease monotonically. Hx of IC has ability to remove the decay heat at the time of 5 minutes after SCRAM.
- The decay heat during IC operation was less than cooling rate of IC Hx. Therefore, over cooling of RPV make the pressure decrease.
- The reason why operator selected on/off operation of IC is to control the RPV pressure within certain range to maintain cooling rate below 55 ºC per hour, therefore IC was operated by switching on/off to keep reactor pressure within certain range by opening or closing the valve.

2. REACTOR COOLING JUST AFTER EARTHQUAKE

TEPCO

- In units 2 and 3, when water level decreased to a certain level, RCIC was activated automatically.
- Because RCIC injection rate was set equal to water boil off rate at 15 minutes after scram, water injection rate was excessive compared to the water loss by boil off.
- When reactor water level increased to certain level, RCIC was tripped automatically.
- Therefore, operator controlled the valve opening to keep the reactor water level within proper range.



12

- However, maintaining water level is not easy task even for skilled operators because of rapid decrease of decay heat and SRV activation.
- In fact, RCIC in unit 2 and 3 was automatically tripped several times due to high RPV water level.

Decay heat decrease required operators' action to control RPV behavior.
IC, which is passive system, was not exception.

2

TEPCO

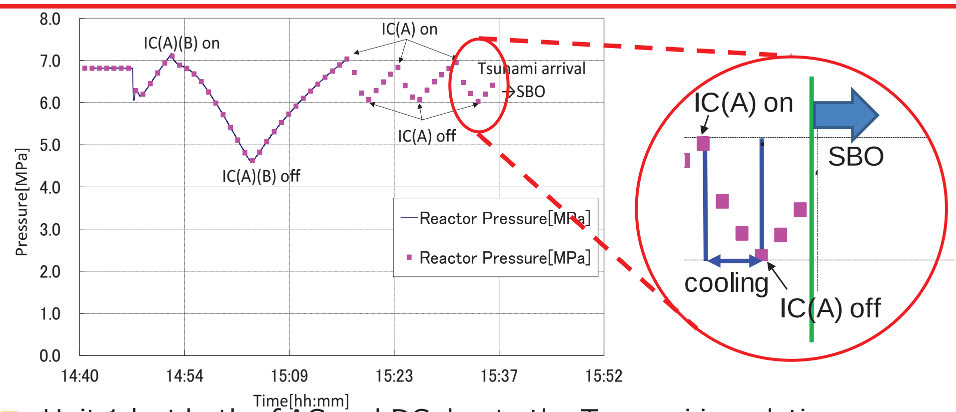
ii: Reactor cooling without AC/DC power before core damage

There are no operator's actions or operations without accurate recognition of reactor condition.

13

3. AVAILABILITY/UNAVAILABILITY OF REACTOR COOLING SYSTEM AFTER SBO

TEPCO

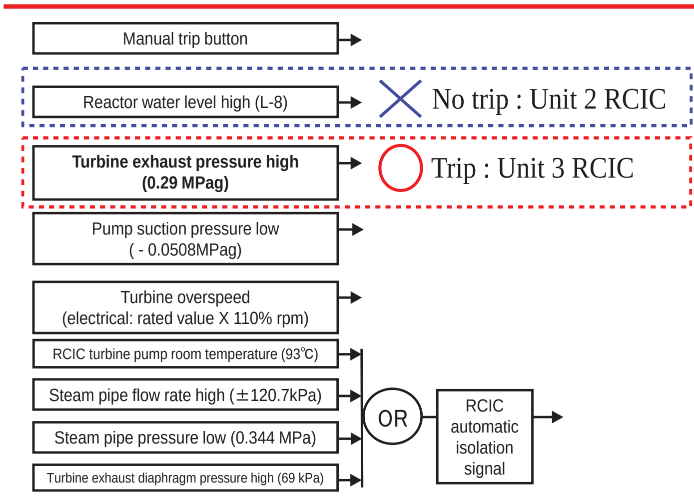


- Unit 1 lost both of AC and DC due to the Tsunami inundation. We call this total SBO.
- At the time of SBO, IC was not operated. And the valve to activate the IC was DC driven so that operators could not restart the IC.
- This loss of cooling was the direct cause of unit 1 to fall into SA.
- There is a discussion the need of RPV pressure control in such situation. However, operators were trained for the SCRAM scenario.
- Adding to that, IC was designed to close the valve when DC was lost.

14

3. AVAILABILITY/UNAVAILABILITY OF REACTOR COOLING SYSTEM AFTER SBO

TEPCO

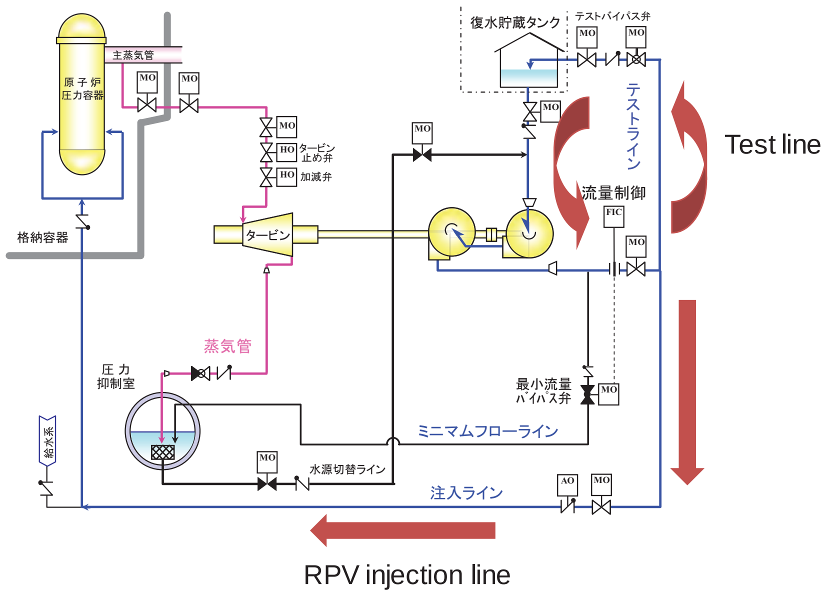


- Unit 3 RCIC tripped due to turbine exhaust pressure high.
- Unit 2 RCIC didn't trip reactor water level high because water level could not measure due to loss of DC power.
- Even though unit 2 RCIC was out of design condition, loss of DC power enabled longer time operation of unit 2 RCIC than that of unit 3 RCIC.

15

3. AVAILABILITY/UNAVAILABILITY OF REACTOR COOLING SYSTEM AFTER SBO

TEPCO



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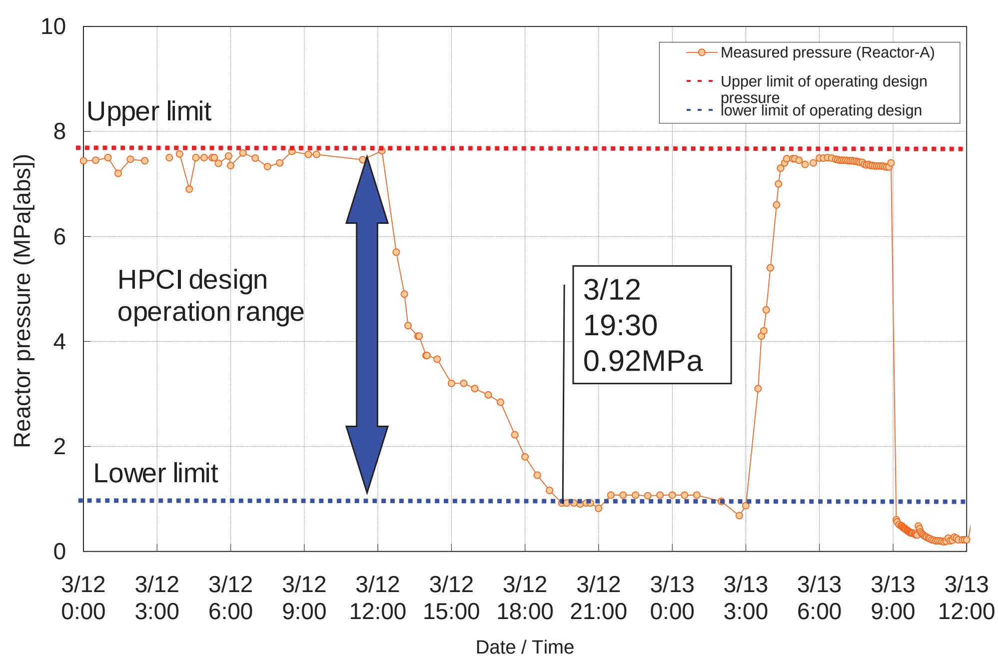
3. AVAILABILITY/UNAVAILABILITY OF REACTOR COOLING SYSTEM AFTER SBO

TEPCO



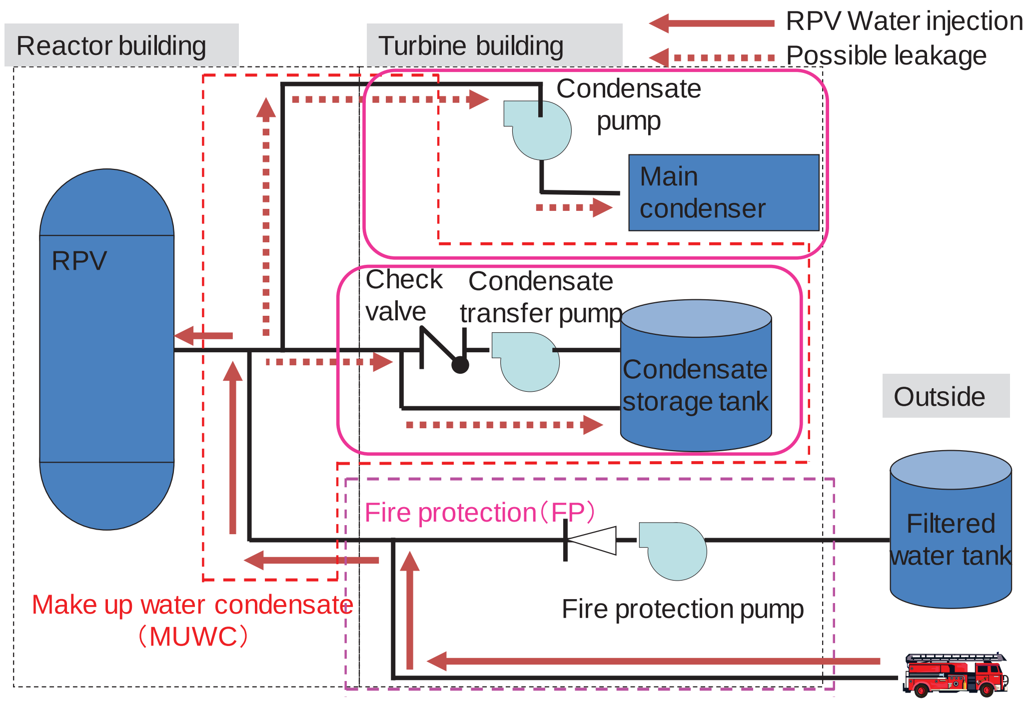
17

iii: Debris cooling by using alternative water injection line

This time period started in the middle of March and ended in December.

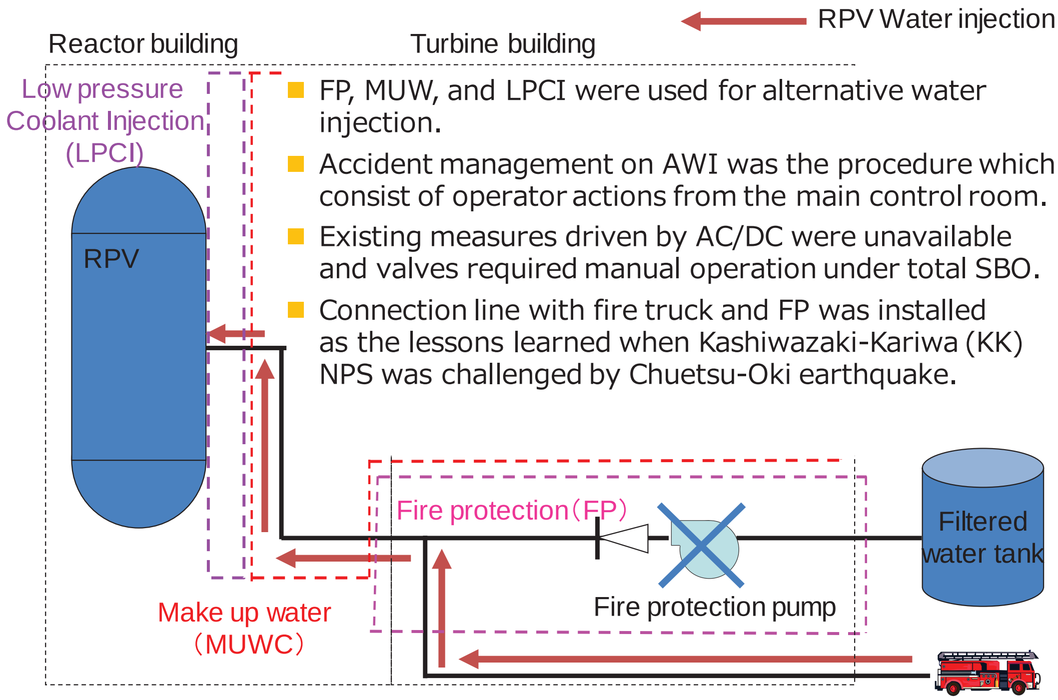
18

4. ALTERNATIVE WATER INJECTION



4. ALTERNATIVE WATER INJECTION (AWI)

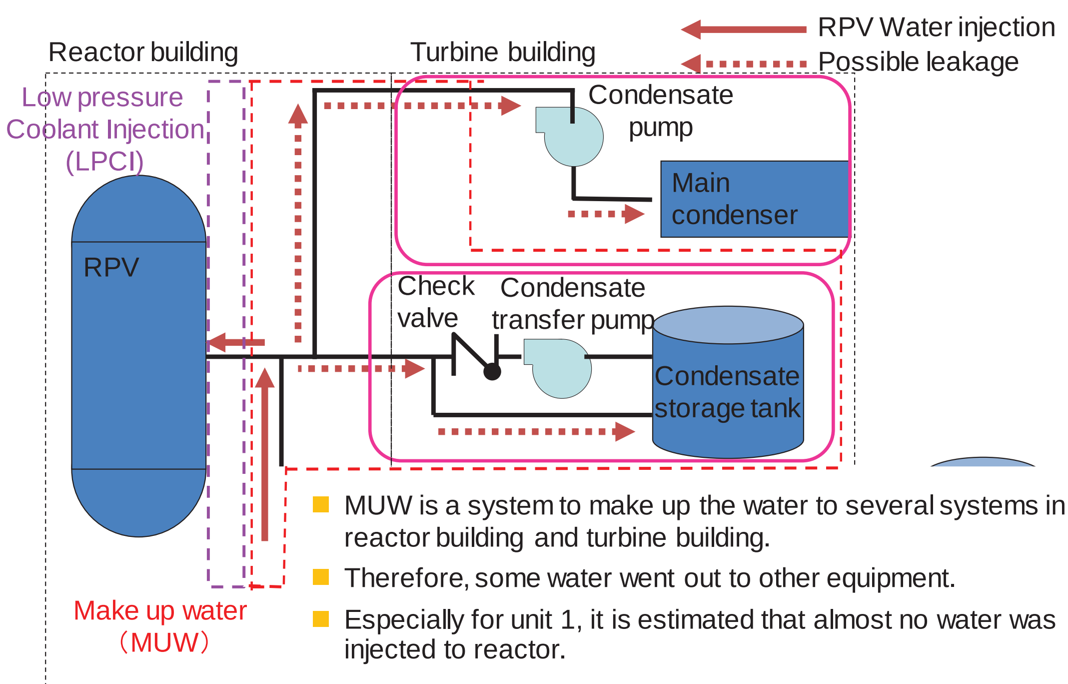
TEPCO



- FP, MUW, and LPCI were used for alternative water injection.
- Accident management on AWI was the procedure which consist of operator actions from the main control room.
- Existing measures driven by AC/DC were unavailable and valves required manual operation under total SBO.
- Connection line with fire truck and FP was installed as the lessons learned when Kashiwazaki-Kariwa (KK) NPS was challenged by Chuetsu-Oki earthquake.

4. ALTERNATIVE WATER INJECTION (AWI)

TEPCO

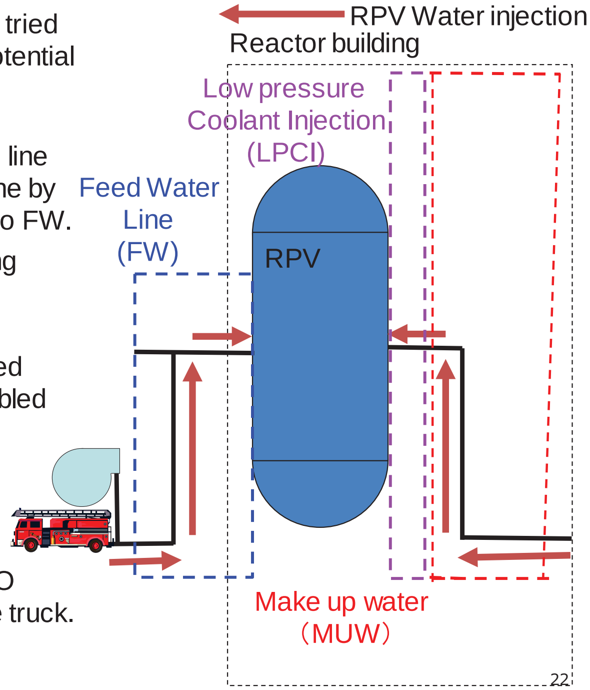


- MUW is a system to make up the water to several systems in reactor building and turbine building.
- Therefore, some water went out to other equipment.
- Especially for unit 1, it is estimated that almost no water was injected to reactor.

4. ALTERNATIVE WATER INJECTION (AWI)

TEPCO

- Around the end of March, operators tried to close the valves which had the potential leakage.
- TEPCO changed the water injection line from AWI to the Feed Water (FW) line by direct connection firetruck or pump to FW.
- FW line is a line to inject water during normal operation. So, there are no leakage potential.
- Unit 1, to which no water was injected through AWI, was the first plant enabled injection from FW. March 23 2011.
- Unit 2: end of May
- Unit 3: middle of May
- In Kashiwazaki-Kariwa NPS, TEPCO introduced new injection point of fire truck.
- This enables direct injection to RPV without leak potential.



22

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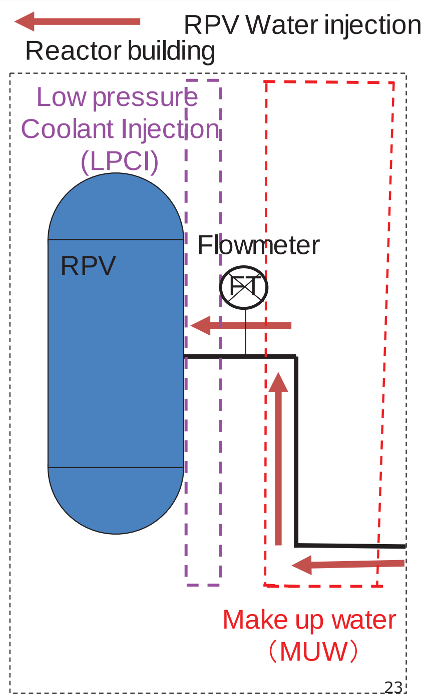


23

5.Conclusion

TEPCO

- During the Fukushima Daiichi accident, several methods of reactor cooling were used.
- The behavior of each equipment and the differences among three units are almost clarified in 10 years after the accident.
- The accident was extremely severe compared to the prepared scenarios in accident management. Some worked and many failed in accident progressions.

- It is expected that lessons learned from this accident will contribute to the enhancement of future nuclear safety to prevent the accident like Fukushima Daiichi.

24

TEPCO

Thank you for your kind attention.

25

C.1.4. JAEA Update and Discussion

This slide includes results obtained under research program entrusted to International Research Institute for Nuclear Decommissioning, including Japan Atomic Energy Agency, by Agency for Natural Resources and Energy, Ministry of Economy, Trade and Industry (METI) of Japan. <1/30>

Reactor Safety Technology Expert Panel Forensics Meeting

November 28-30, 2021

Session 1

JAEA Update and Discussion

Collaborative Laboratories for Advanced Decommissioning Science (**CLADS**),

Japan Atomic Energy Agency (JAEA)

International Research Institute for Nuclear Decommissioning (IRID)

Contents

<2/30>

Updates on JAEA activity related to fuel debris analysis (1/2) (Outline) — **S. Koyama**

Update on 1F sample analysis — **H. Ikeuchi**

Updates on JAEA activity related to fuel debris analysis (2/2) (Analysis in Ibaraki area) — **S. Koyama**

integrated Radiation Imaging System based on Compton Camera * — **Y. Sato**

* Presented in the separated document

C.1.4.1. Updates on JAEA Activity related to Fuel Debris Analysis

Forensic meeting
Nov. 29th 2021

Updates on JAEA activity related to fuel debris analysis

Shin-ichi Koyama

Fuel debris research and analysis division
Collaborative Laboratories for Advanced Decommissioning Science
Japan Atomic Energy Agency

This slide includes results obtained under research program by Agency for Natural Resources and Energy, Ministry of Economy, Trade and Industry (METI) of Japan.

1

JAEA report for fuel debris analysis

- Design, planning and control of debris-related processes, namely retrieval, storage management, processing and disposal of the debris, are required for the safe and steady decommissioning of 1F.
- For this purpose, it is indispensable to conduct an internal survey inside the containment vessel / reactor pressure vessel and to analyze the characteristics and deposition state of fuel debris, the distribution of fission products / radiation doses, and damage to structural materials and the state of corrosion.
- Continuous updating and improvement of the process design are important through ascertainment of the cause of the accident.
- JAEA promoted to summarize a report required debris analysis in relation with issues for the retrieval, storage management, processing and disposal, and ascertainment of the cause of the 1F accident based on past experience and achievements. Practical analysis plan is expected to be prepared based on this report .

2

<5/30>

Fuel debris analysis report

Published as a JAEA report (May,18,2020)

Published English version (Dec.8, 2020)

Introduced at OECD / NEA International Project

Base for future international research cooperation

The report will be updated based on

- Comments from USA
  - Dr. Mitch Farmer,
  - Dr. Martin Plys,
  - Dr. David Luxat,
  - Dr. Damian Peko,
  - Dr. Joy Rempe.
- Recent activity

Now update is now underway on 1F task force

Organized in JAEA.

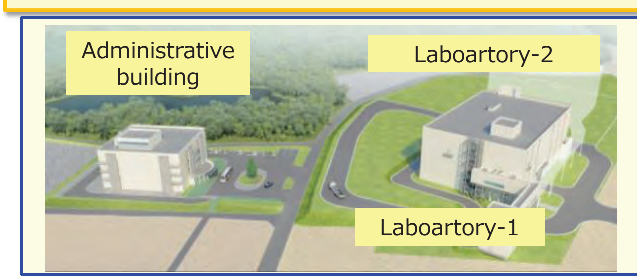
3

<6/30>

Outline of Okuma analysis and research center

〇The Okuma Analysis and Research Center is a facility that conducts research and development related to the analysis of radioactive waste and fuel debris for evaluation of characteristics and safety during storage, etc.

〇As a facility dedicated to analysis of rubble and fuel debris, it will contribute to the remediation and decommissioning of 1F.



Administrative building
: Office work, cold mockup of analysis equipment

Laboartory-1
: Analysis of radioactive wastes

Laboratory-2
: Analysis of fuel debris

Developing of human resources for analysis and facility operation are in progress in parallel with facility construction.

4

Reactor Safety Technology Expert Panel Forensics Meeting
Nov. 29, 2021

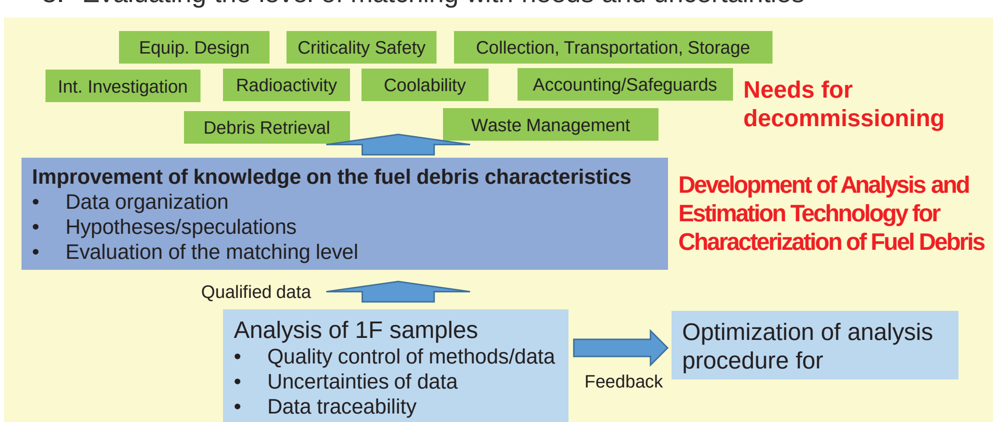
Update on 1F sample analysis

Hirotomo IKEUCHI
Japan Atomic Energy Agency
International Research Institute for Nuclear Decommissioning

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

Introduction

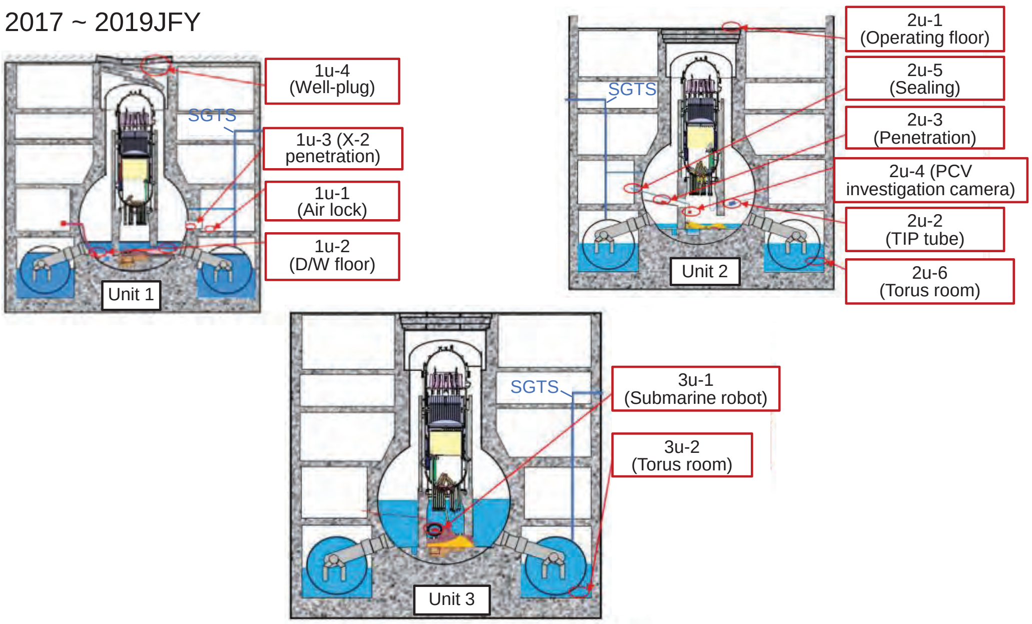
- Various kinds of data obtained by the 1F sample analyses are able to contribute to the 1F accident forensics.
- As the debris characteristics is still highly uncertain, the acquired data should be utilized carefully considering not only the validity of the results, but also how it would match with the needs for decommissioning.
  1. Organizing the qualified data (fact) systematically
  2. Setting the rational, and realistic hypotheses/speculations based on the fact
  3. Evaluating the level of matching with needs and uncertainties



1F Samples Analyzed So Far

Though the samples obtained so far are not categorized to “fuel debris”, some radionuclides or U-bearing particles were found.

2017 ~ 2019JFY



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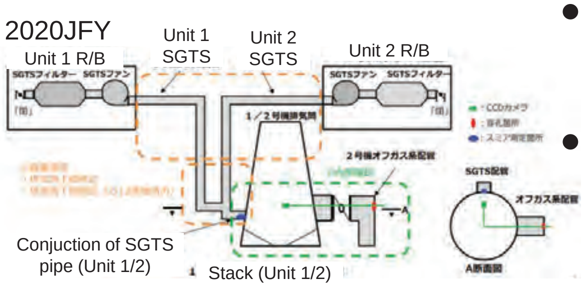
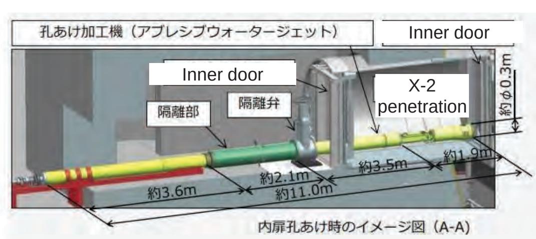
2020JFY





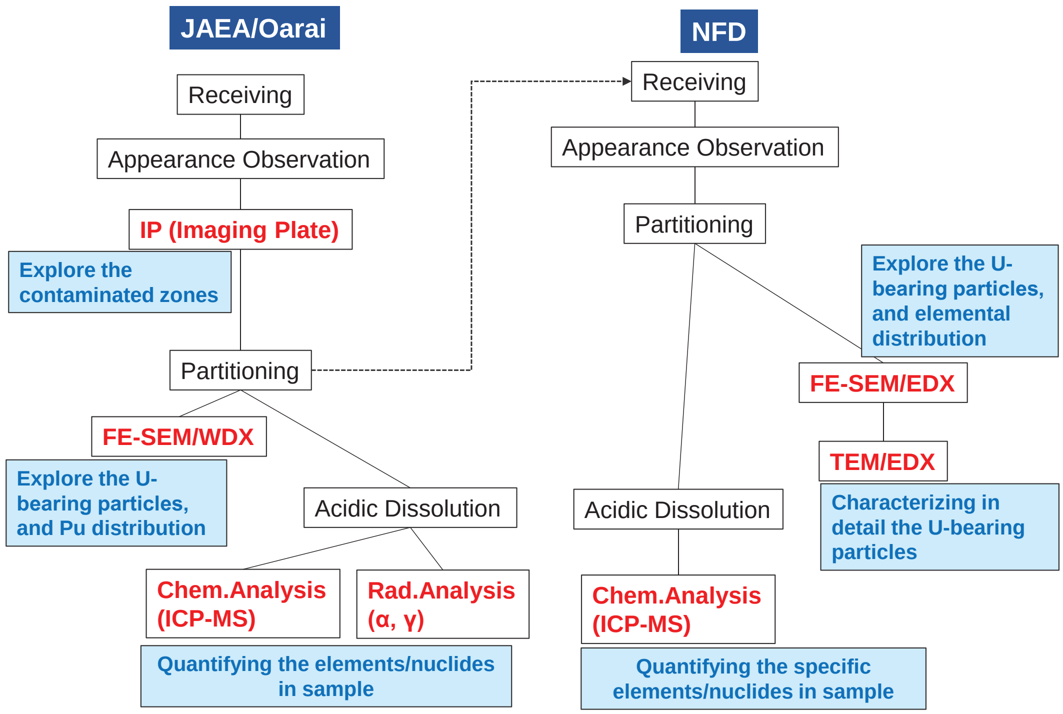
- Smear samples of the SGTS pipe (between Unit 1/2)
  ⇒ Gas from the Unit 1 PCV atmosphere could have passed.
- Smear/Filter samples from the gas control equip. and AWJ* equip. from Unit 1.
  ⇒ Information on the inner wall of Unit 1 PCV is expected.

Specific scenario for Unit 1

- Fuel debris could have dropped down to the PCV via short time
- The fuel debris on the pedestal floor could have been kept at high temperature for almost 1 week, without an efficient water injection
- The inner wall of the PCV could have been kept at approx. 400 °C

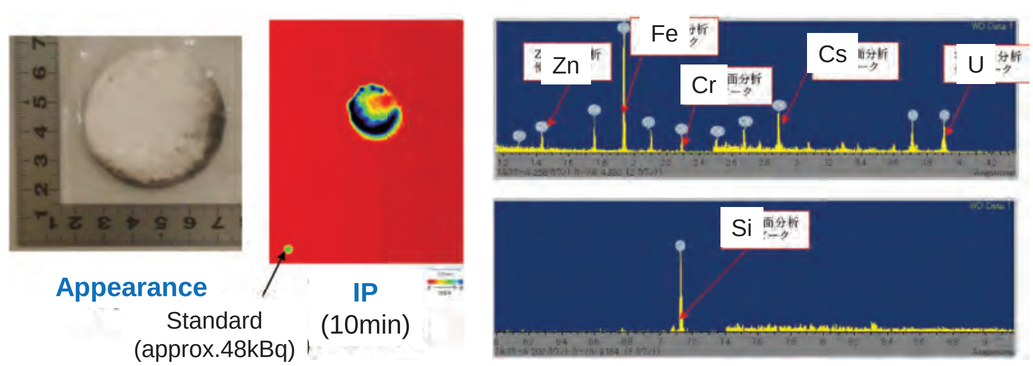
* AWJ : abrasive water jet

Basic Flow of the Analysis



Results (outside PCV)

Smear sample of the Unit 1/2 SGTS pipe



Appearance
Standard (approx.48kBq)
IP (10min)
FE-SEM/WDX

- ◆ **Appearance, IP**
  - Identifying and partitioning the highly contaminated area
- ◆ **FE-SEM/WDX**
  - Qualitative analysis based on the peak identification, and mapping analysis.

→ Zn, Fe, Cr, Cs, U, Si identified
→ No Pu identified in this sample

Results (outside PCV)

<13/30> 7

Smear sample of the Unit 1/2 SGTS pipe

◆ **Rad. and Chem. Analysis**

- U → Fuel originated
- B, Cr, Fe, Zr, Mo → Close to the natural isotopic ratio (Mo could contain a contribution from FP)

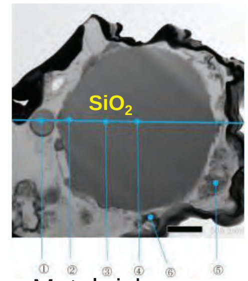
Isotopic ratio of U

	XM20111	XM20121
U235/ U238	1.65×10^{-2}	1.9×10^{-2}
U236/ U238	2.53×10^{-3}	1.9×10^{-3}

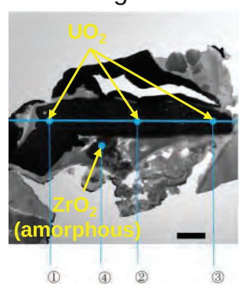
◆ **SEM/EDX, TEM/EDX (analysis of a single particle)**

- U-bearing particles with various types
  1. UO_2–ZrO_2–FeO_x mixture
  2. Separated between UO_2 vs ZrO_2 (possibly a piece of irradiated fuel)
- Metal-rich particles
  - ➢ Ag–Te, Ag–Te–Pb, etc. (possibly contains volatile FPs such as Ag, Te)
- Si–Cs rich particles
  - ➢ Glass-like spherical particles

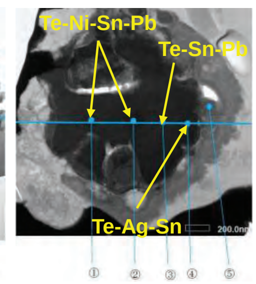
Si–Cs rich



U-bearing



Metal-rich



Typical examples of TEM/EDS analyses of particles

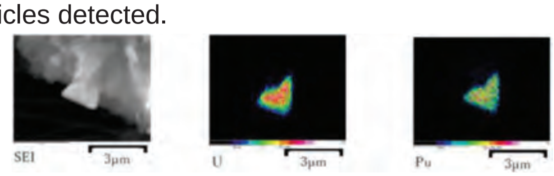
Results (inside PCV)

<14/30> 8

Smear/Filter samples of access route of Unit 1

◆ **Filter samples from the gas control equip.**

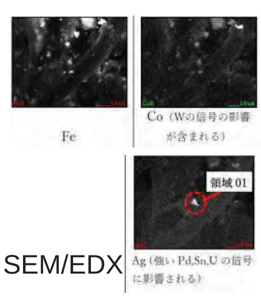
- Chem. Analysis : Mo, Zr, Cr, Fe, B, U detected.
- Rad. Analysis :^{134}Cs, ^{137}Cs detected
- SEM/WDX : Detected Fe-based particles, and U-bearing particles (with Pu)
- TEM analysis : No U-bearing particles detected.



SEM/WDX

◆ **Smear samples from the AWJ equip.**

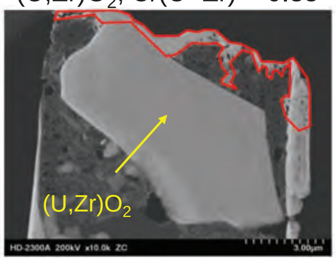
- Chem. Analysis : Mo, Zr, Cr, Fe, B, U detected.
- Rad. Analysis :^{134}Cs, ^{137}Cs dected
- SEM/WDX : Detected U-bearing particles (with Pu)
- SEM/EDX : Fe, Co, Ag detected at different regions
- TEM analysis : No U-bearing particles detected.



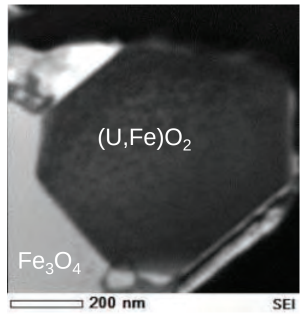
SEM/EDX

Revisiting the Previous Data

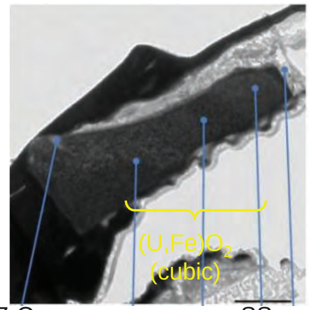
(U,Zr)O$_2$, U/(U+Zr) > 0.85



1u-2
(D/W floor)



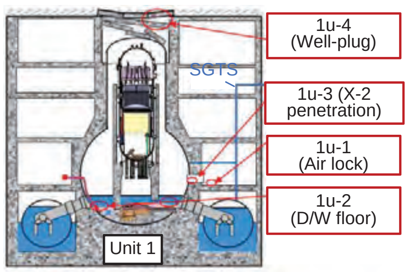
1u-4
(Well plug)



ZrO$_2$
(monoclinic)

SS comp.
(unidentified)

1u-3
(X-2 penetration)

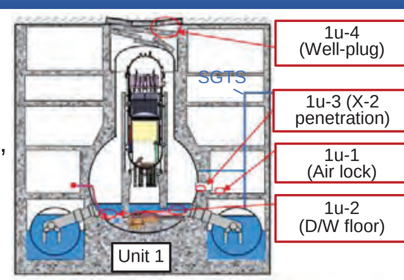


- **Previous data of Unit 1 samples (obtained by 2019) data were also reviewed** to reinterpret the probable mechanism of formation of the U-bearing particles.

→ **Characteristics were quite different according to their locations.**

Hypothesis/Speculations (Quick look)

- Elements detected and their origins (evaluated by referring to the initial load materials of RPV/PCV) :
  U (fuel), **Mo*** (grease), **Zr** (cladding and canister), **Zn** (paint), **Pb** (shielding), **Fe**, **Ni**, **Cr** (steels), **B** (neutron absorber), **Na** (sea water), **Si**, **Al** (thermal shield)
  ** Possibly contains contributions from FPs, steels, etc.*



→ Fuel debris in the pedestal floor could have much various origins such as sea water, grease, thermal shield, shielding, as well as the main core materials including concrete and steels.

Differences by samples

- **The D/W floor (1u-2)** : Fluorite phase with low Zr contents dominated.

→ Indicating the **solidification of molten mateials (followed by a slow cooling)**

- **The X-2 penetration (1u-3), and the well plug (1u-4)** : Separation between (U,Fe)-based oxide and ZrO$_2$

→ The **condensation from the gas phase** could contribute the formation mechanism.

- Metallic Zr has not been identified yet.

To be updated being based on the recent results...

Conclusions

Summary

- Extensive data has been acquired by the analyses of radioactive samples obtained from various locations in 1F (in- or outside PCV).
  - In 2020JFY, the newly obtained samples were characterized to deepening the knowledge especially on the Unit 1 accident progression.
    - Smear obtained from the SGTS guide pipe (Unit 1/2)
    - Smear and filter samples between Unit 1 and 2, and access route samples (incl. the gas control equip. and the AWJ equip.)
  - The previous data (obtained up to 2019 JFY) were reviewed to evaluate probable mechanism of formation of U-bearing particles.

Perspective

- Continuous effort to update the hypothesis/speculations based on the highly qualified experimental data.
  - The data and evaluations will be used in the backward analysis of the core melt progression.
  - Better dissemination for the international collaboration

Analysis in Ibaraki area

Discussion by hot lab researcher and engineer with TEPCO HD experts

■ Objectives
・Analytical method and system should be established that can be evaluate the characteristics when receiving fuel debris whose properties are not clear and analysis is not easy.
・Collaboration of a representative Japanese analytical organization (having a hot laboratory) responsible for fuel debris analysis.

■ Hot laboratory in Ibaraki area as candidate of fuel debris analysis
・JAEA NSRI（RFEF、BECKY、WASTEF、Lab No.4）
・JAEA NCL（CPF）
・JAEA Oarai（FMF、AGF）
・Nuclear fuel development Co.Ltd.（NFD）
・Nuclear development Co. Ltd.（NDC）

■ Simulated fuel debris prepared by an organization independent of the analytical facilities in the Ibaraki area is analyzed by each facilities, the results were compared and discussed in order to construct and standardize each analytical technique and evaluation method.

5

Efforts to standardize fuel debris analysis technology

METI「The project of decommissioning and contaminated water management」

Development of Technologies for Enhanced Analysis Accuracy

Objectives

・Analytical method and scheme should be established that can evaluate the characteristics when receiving fuel debris whose properties are not clear and analysis is not easy.

・Collaboration of a representative Japanese analytical organization (having a hot cell laboratory) responsible for fuel debris analysis.

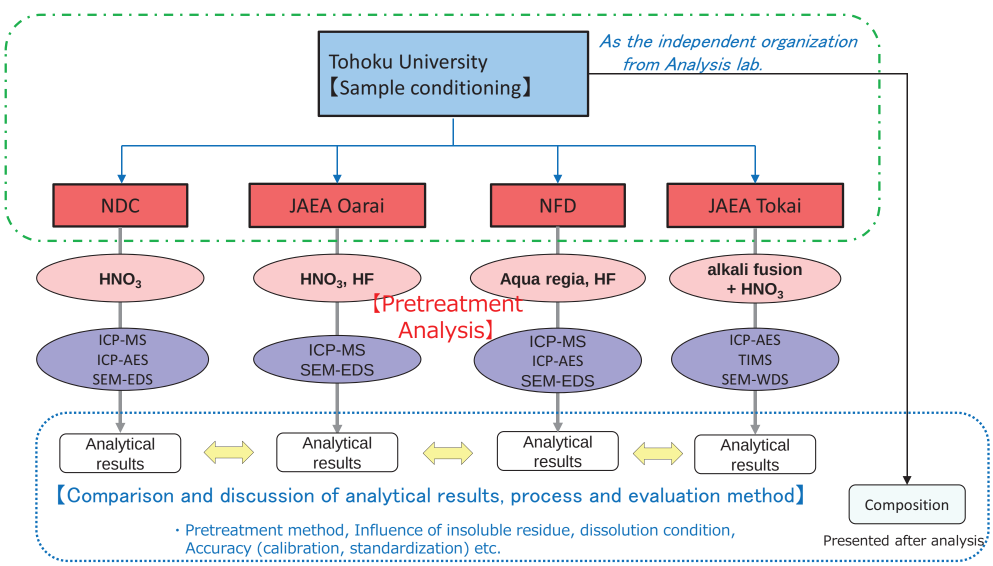
- To understand and deal with issues and difficulties in measurement by using current analytical techniques.
- To timely evaluate and establish analytical scheme when unexpected analysis results and difficulties occur in analysis work.
- To share evaluation methods regarding the level of "certainty" of the obtained results.

standardization

- To define analytical techniques for four important basic quantities (morphology, nuclide/element amount, phase, density/porosity) as fuel debris analysis items, and to be shared among the parties concerned.
- To cope with the analysis needs of fuel debris analysis evaluators by appropriately combining the analytical techniques of each hot lab.

【出典】【JAEA】令和2年度開始「廃炉・汚染水対策事業」に関する補助事業（第一次公募、燃料デブリの性状把握のための分析・推定技術の開発（燃料デブリの分析精度の向上及び熱挙動の推定のための技術開発））の成果の概要（https://dccc-program.jp/category/result）より抜粋。
(Ref) JAEA's summary results of subsidy program (the First solicitation) for the" Project of Decommissioning and Contaminated Water Management (Development of Analysis and Estimation Technology for Characterization of Fuel Debris (Development of Technologies for Enhanced Analysis Accuracy and Thermal Behavior Estimation of Fuel Debris))" starting FY2020 [in Japanese]

Participant organization and the sample flow



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Preparation of simulated fuel debris sample (element, requirements)

Target elements

- U, Zr, Gd
- Fe, B
- Cr, Ni
- Si

《Basis of selection》

U, Zr	: main component, affect to the formation of insoluble residues
Gd	: burnable poison components affecting criticality assessment
Fe	: main component of fuel debris
B	: control rod components affecting criticality assessment
Cr, Ni	: major component of SUS, viewpoint of accident progress
Si	: component of concrete and insulator materials, affect to formation of insoluble residues

Sample preparation

(1) Simulated fuel debris containing Uranium

Prepared by mixing the following powders

- ・Oxide solid solution*1 【Component A】
- ・FeB 【Component B】
- ・SUS304, SiO_2 【Component C】

《 Basis of decision 》

Requirement for simulated fuel debris

- ・Simulation and reproducibility
- ・Solubility (insoluble)*1, testability
- ・Manufacturability and homogeneity
- ・Guaranteed composition*2

(2) Simulated fuel debris not containing Uranium*3

Prepared by mixing [Component B] and [Component C]

＊1 Sample containing two phases of a cubic solid solution with relatively high solubility and a tetragonal solid solution with poor solubility is fabricated by heat-treating the mixed press powder of UO_2, ZrO_2, and Gd_2O_3 at high temperature.

＊2 It is necessary to keep the composition at sample preparation to guarantee the composition by a method other than analysis.

＊3 It is prepared and used to verify the handling of radioactive substances in the hot cell due to no license for unirradiated U in hot cell（NFD）.

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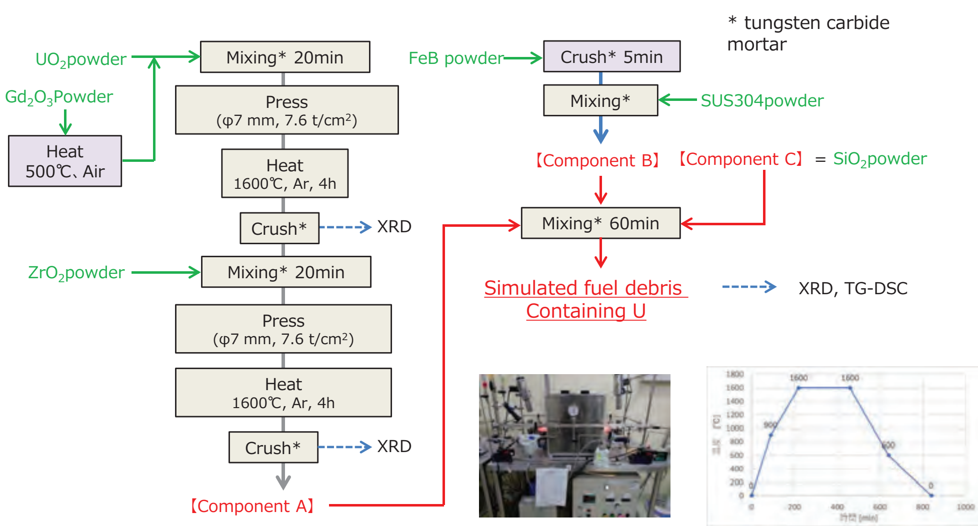
8

Preparation of simulated fuel debris sample

Flow of simulated fuel debris containing U



Electric combustion furnaces

[Component A] Heating program during sintering

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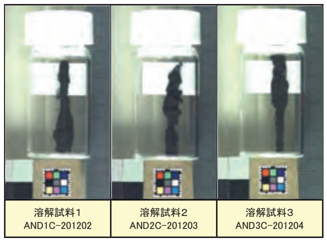
9

Observation of received samples at each lab.

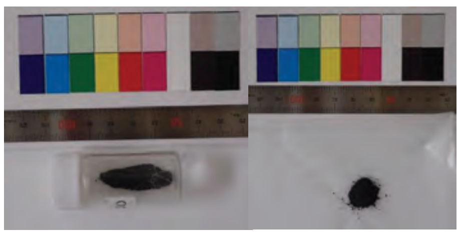
<23/30>

Observation (Simulated fuel debris containing Uranium)



(NDC)

(NFD)



(JAEA Oarai)

(JAEA NSRI)

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10

Dissolution and observation of insoluble residues

<24/30>

Comparison of solubility

Solubility and insoluble residue (Simulated debris fuel containing U)

Organization (Dissolution method)	Sample No.	Sample weight (g)	Weight of insoluble residue (g)	dissolved weight (g)	Solubility		Remark
NDC (HNO_3)	AND1C	0.1004	0.0407	0.0597	0.5946	**0.60**	
	AND2C	0.1007	0.0380	0.0627	0.6226		
	AND3C	0.1000	0.0409	0.0591	0.5910		
JAEA Orai (HNO_3)	AJO1C	0.0998	0.0057	0.0941	0.943	**0.91**	After dissolving HNO_3, add 1 drop of HF
	AJO2C	0.0995	0.0158	0.0837	0.841		
	AJO3C	0.0975	0.0065	0.0910	0.934		
NFD (Aqua regia +HF)	20ANF0C2	0.0925	0.002	0.0905	**0.98**		Dissolution n=1
JAEA NSRI (HNO_3+Alkali fusion)	AJN1C	0.1007	0	0.1007	**1.00**		〃
JAEA NSRI (Alkali fusion)	AJN2C	0.1440	0	0.1440	**1.00**		〃

- In the nitric acid dissolution method, relatively high solubility was observed in applying JAEA condition (heating at 100 ° C x 6h in 8M nitric acid, adding 1 drop of hydrofluoric acid, and then heating again at 100 ° C x 6h) compared with the NDC condition (heating for 60 minutes while boiling 8M nitric acid).
- Almost completely dissolution was observed by the aqua regia -hydrofluoric acid dissolution method or the alkali fusion - nitric acid dissolution method.

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11

Sample dissolution（HNO_3 : JAEA Oarai）

Simulated fuel debris containing Uranium

Condition

① 8M HNO_3 20mL, 100℃×6h Heat

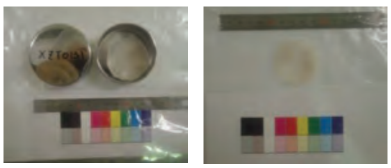
② Add HF (1 drop)

③ 100℃×6h Heat

Sample dissolution

Sample injection　　Add HNO_3　　After heating

Dissolved solution



After filtering the insoluble residue

Results of dissolution test

Sample №.	Sample weight (g)	Weight of insoluble residue (g)	Dissolved weight (g)	**Solubility**
AJO1C	0.0998	0.0057	0.0941	**0.943**
AJO2C	0.0995	0.0158	0.0837	**0.841**
AJO3C	0.0975	0.0065	0.0910	**0.934**

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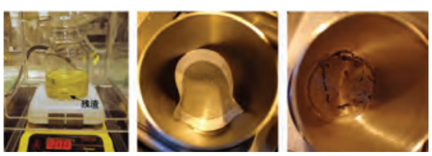
Sample dissolution（Alkali fusion, HNO_3 : JAEA NSRI）

Simulated fuel debris containing Uranium



HNO_3 dissolution　　Dry up filter　　Residue after incineration

filter incineration

Condition

① 13M HNO_3 100mL×4h Heat

② Dry up

③ 300℃ Incineration

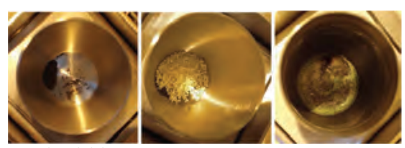
④ Add Na_2O_2, 650℃×1h Heat

⑤ Wash H_2O, 1M HNO_3

⑥ Add 6M HNO_3 Heat below b.p.

⑦ Wash 13M HNO_3

Step from HNO_3 to Alkali fusion



Simulated debris powder in Ni crucible　　Add alkali flux　　Product after fusion

Add HNO_3　　Heating　　After cooling　　Collect to GB

Alkali fusion

Results

Condition	Condition	Sample weight (g)	Weight of insoluble residue (g)	Dissolved weight (g)	**Solubility**
①～⑦	HNO_3+Alkali fusion	0.1007	0	0.1007	**1.00**
④～⑦	Alkali fusion	0.1440	0	0.1440	**1.00**

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Analysis results (quantities of element) by each lab.

Analytical results element composition*

*Evaluated value including quantitative analysis of insoluble residue by SEM

element	content(mg/100mg)					
	Tohoku university	NDC	JAEA Oarai	NFD	JAEA NSRI 1	JAEA NSRI 2
	original	Nitric acid dissolution	Nitric acid dissolution	Aqua regia and hydrofluoric acid dissolution	Nitric acid + Alkali fusion of residue	Alkali fusion
U	20.5 ±0.1	20.0 ±0.6	21.9 ±1.0	17.5 ±0.2	16.5 ±0.8	17.3 ±2.0
Gd	1.7	1.3 ±0.2	1.9 ±0.1	1.5 ±0.2	1.56 ±0.11	1.59 ±0.08
Zr	10.8	11 ±1.4	12 ±1.7	11.3 ±1.1	8.48 ±0.71	10.2 ±0.5
B	4.9	5.6 ±0.5	4.7 ±0.2	4.3 ±0.2	4.63 ±0.50	4.44 ±0.20
Fe	39.0	22.0 ±4.8	34.6 ±3.9	34.7 ±0.4	33.8 ±1.6	35.6 ±1.6
Cr	3.7	0.7 ±0.9	2.8 ±0.6	3.5 ±0.1	2.75 ±0.10	3.57 ±0.15
Ni	2.2	0.3 ±0.4	1.6 ±0.3	2.0 ±0.1	180 ±10	31.0 ±1.4
Si	4.9	12.0 ±2.4	3.6 ±4.5	5.4 ±0.4	3.40 ±0.17	3.22 ±0.16
O	12.2 ±0.1	-	-	-	-	-
Solubility	-	60%	91%	98%	100%	100%

repetition number	Tohoku Univ.	NDC	JAEA Oarai	NFD	JAEA NSRI	
Dissolution	-	3	3	1	1	←
Collection	-	1	1	3	1	←
Measurement	-	Multiple	3	Multiple	3	←

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14

Findings

〈Scope of analytical techniques: for homogeneous samples〉

○HNO_3 dissolution

- The phase that are expected to dissolve are limited to U-rich (U, Zr) O_2 and boron compound.
- However, when only the U isotope ratio is evaluated or when the presence of a specific element is analyzed, it can be a quick and simple dissolution method.
- It is necessary to improve the dissolution rate to improve the quantitativeness. And it was found that the addition of a small amount of hydrofluoric acid is effective in improving the solubility.

○Aqua regia + HF dissolution

- Since total dissolution is expected, and therefore, it is considered that quantitative analysis of nuclides and elements is possible with high accuracy.
- However, rare earth element in the sample may co-precipitate with fluoride. It is necessary to consider re-dissolution of the precipitate in this case.

○Alkali fusion

- All element can be dissolved, and therefore, is suitable for quantitative analysis with high accuracy.
- However, since the Na from the alkaline reagent and the Ni from the crucible are contained, it is necessary to consider the material selection depending on the element to be measured.

〈Applicability to multi-element systems and heterogeneous systems〉

- A homogeneous sample of about 0.1 g was examined, However it is assumed that the actual fuel debris has a larger size and many elements are heterogeneously distributed in one sample.
- In that case, the composition and distribution are analyzed by XRF, SEM, and XRD, and then the site to be investigated in detail is selected. In addition, homogenization by mechanical crushing can be proposed.
- It contains various elements, there is concern about the influence of interfering elements / nuclides, such as superimposition of emission intensity (ICP-AES) and isobaric interference(ICP-MS).

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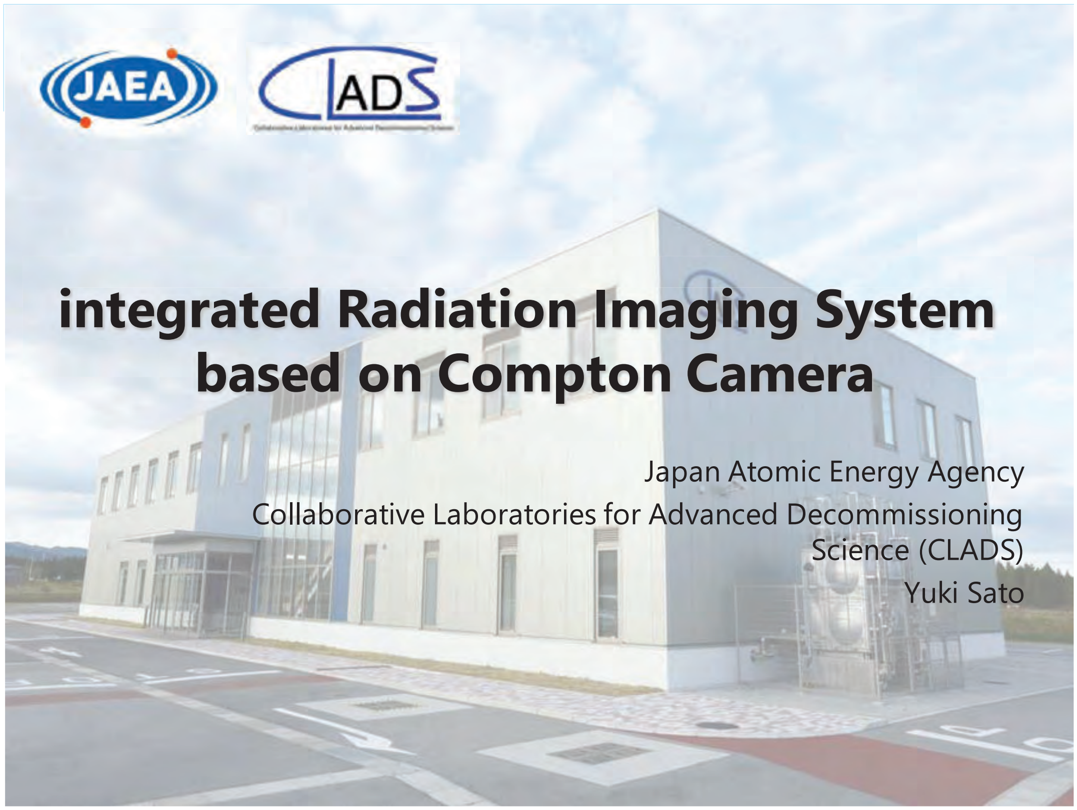
15

Conclusion

- Analysis and evaluation related to fuel debris will be performed consistency with the mid and long term roadmap.
- We are updating the JAEA report, which summarizes JAEA recommendations for fuel debris analysis, reflecting the comments from USA and recent findings. The update is done at the Task Force, which is a gathering of people involved in debris of all JAEA.
- Preparation for the operation of the Okuma Analysis and Research Center, preparation of analysis in the Ibaraki area, and effort of standardization for analytical technique and evaluation method are promoted.

Future perspectives

- To construct and standardize fuel debris analysis techniques and evaluation method, which need to be feasible and reliable (sharing findings from experimental results).
- An international round-robin analysis activity is planned to be performed in new OECD/NEA/FACE project.
- JAEA would like to continuously analyze samples obtained from inside and outside the PCV on 1F (reported from H. IKEUCHI) and fuel debris taken out on a trial basis, verify the analysis technology, provide property data based on the evaluation of the analysis results, and create a database of analysis data.
- Through the analytical technology development, cooperation between domestic and foreign hot laboratories and human resource development will be implemented, and stable operation of the Okuma Analysis and Research Center for fuel debris analysis and will be realized by technical and human support.



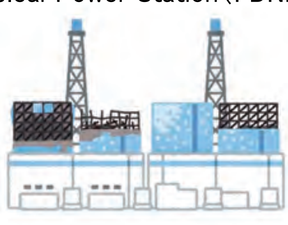
integrated Radiation Imaging System based on Compton Camera

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

Introduction

1

Fukushima Daiichi Nuclear Power Station (FDNPS)

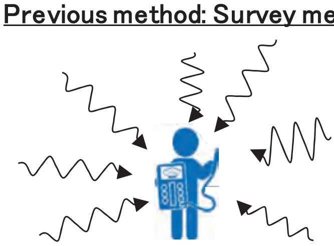


Work environment is contaminated with a radioactive substance.

Information of radiation distribution is necessary

- Decreasing the exposed dose for workers
- Planning the decontamination

Previous method: Survey meter



Measured at "point"

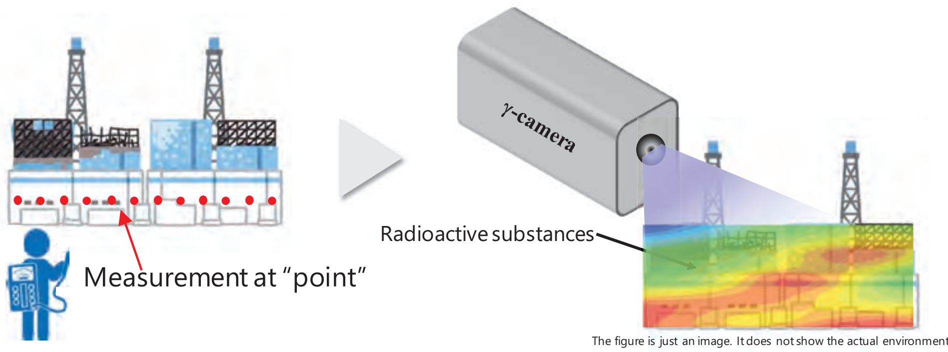
Matter of concern

- Flying direction of gamma rays can not be measured, and it is difficult to locate the radioactive substance
- Overlooking hotspots when measuring wide area
- Increase in worker exposure as measurement time increases

Gamma-ray imager

2

Visually "see" radioactive substance



Visualizing the distribution of radioactive substance using **color information**.

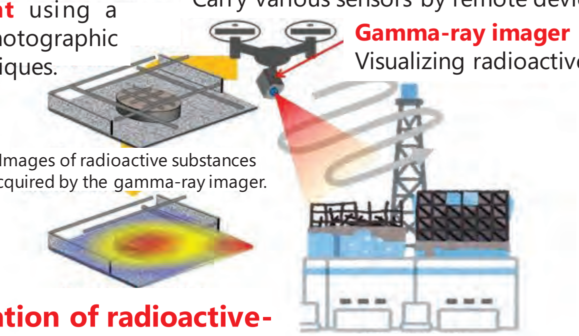
integrated Radiation Imaging System (iRIS)

3

Generate a 3-D model of the work environment using a laser scanner and photographic reconstruction techniques.

Carry various sensors by remote devices

Gamma-ray imager
Visualizing radioactive substances



3-D visualization of radioactive-substances distribution

FDNPS

integrated Radiation Imaging System (iRIS)

integrated Radiation Imaging System (iRIS)

4

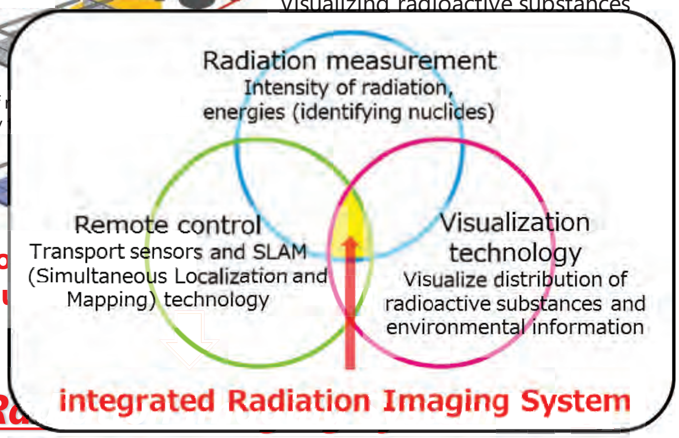
Generate a 3-D model of the work environment using a laser scanner and photographic reconstruction techniques.

Carry various sensors by remote devices

Gamma-ray imager

Visualizing radioactive substances

Images of
acquired by



3-D visualization o
substances distribu

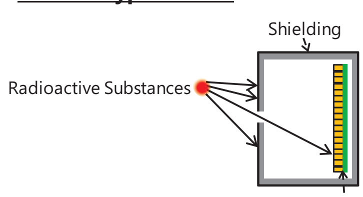
integrated Ra

Gamma-ray imager

5

There are two main types.

Pinhole-type camera



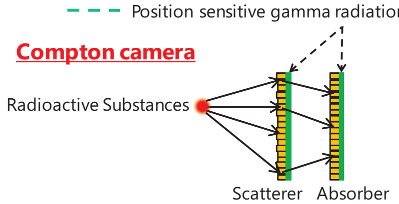
Narrow the field of view with a pinhole
⇒ Shielding is required except for the pinhole area.

Contamination in the 1F building
^{137}Cs (662 keV-γ), ^{134}Cs (604 keV, 795 keV-γ) Mainly
- To reduce gamma-ray intensity to 1/10, shielding of ~cm order with lead is required.

Weighs more than several tens of kilograms

Position sensitive gamma radiation detector

Compton camera



Estimate the direction of gamma rays by solving the kinematics of Compton scattering (scattering of gamma rays)

⇒ No shielding is required in principle.

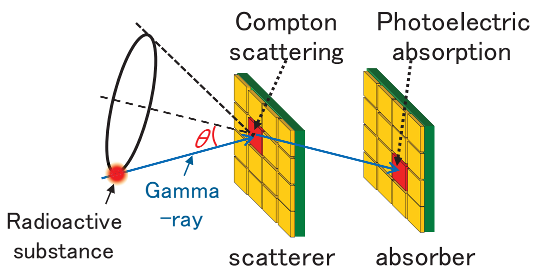
⇒ Advantageous for miniaturization and weight reduction

⇒ Wide field of view

Compton cameras are advantageous in achieving small size and light weight.

Compton camera

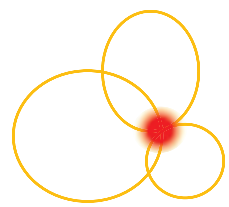
6



Measure "deposited energy" and "interaction position" on scatterer and absorber, respectively

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

⇒ Estimate the **incidence angle of gamma rays and draw a cone** (Compton cone).

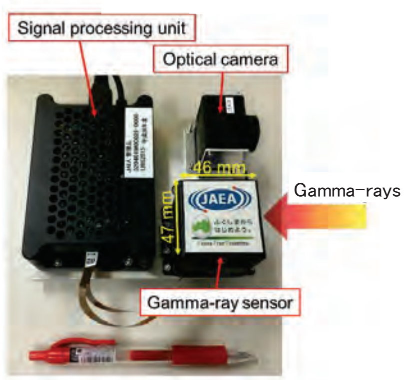


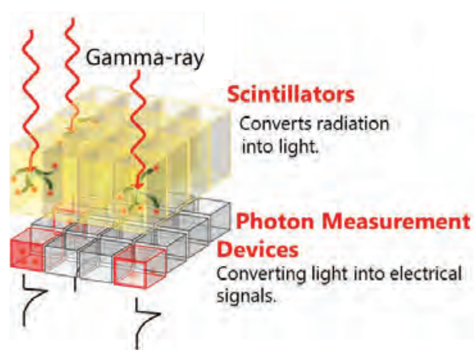
Radioactive substances can be found at the intersection of Compton cones by observing multiple gamma rays.

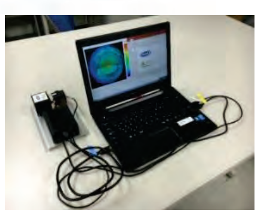
Visualize radioactive substances by superimposing optical images

compact Compton camera for FDNPS

7







- Total weight: less than 1 kg (Without shielding)
- Power consumption < 5 W
  ⇒ Operated by USB bus power

Compact size !!

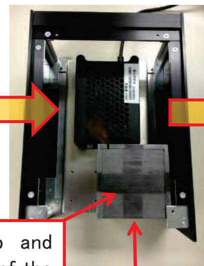
Y. Sato, et. al., Journal of Nuclear Science and Technology, 56 (9-10), pp. 801-808, (2019) Supplemental material
The Compton camera was fabricated based on handheld Compton camera technology jointly developed by Waseda University and Hamamatsu Photonics K. K.

Imaging demonstration inside the FDNPS

8

Experimental condition

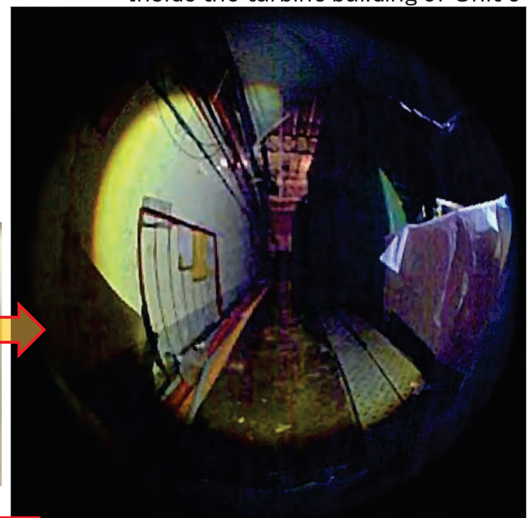
Compton camera



Pb plate installed on the top and bottom, left and right, and back of the gamma ray sensor (1 cm-thickness)

Gamma-ray sensor

Inside the turbine building of Unit 3



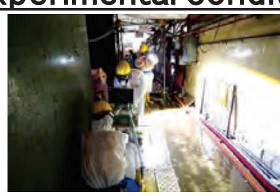
Hotspots are clearly visualized inside the FDNPS

Y. Sato, et. al., Journal of Nuclear Science and Technology, 55(9), pp. 965-970, (2018)

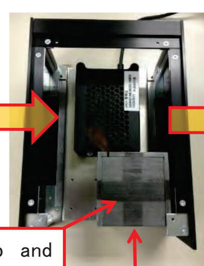
Imaging demonstration inside the FDNPS

9

Experimental condition



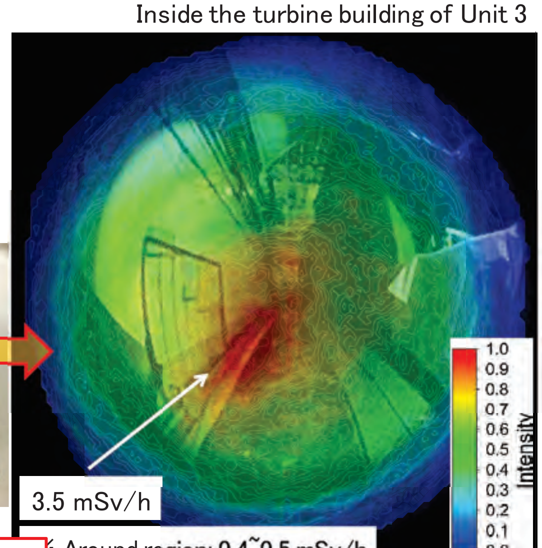
Compton camera



Pb plate installed on the top and bottom, left and right, and back of the gamma ray sensor (1 cm-thickness)

Gamma-ray sensor

Inside the turbine building of Unit 3

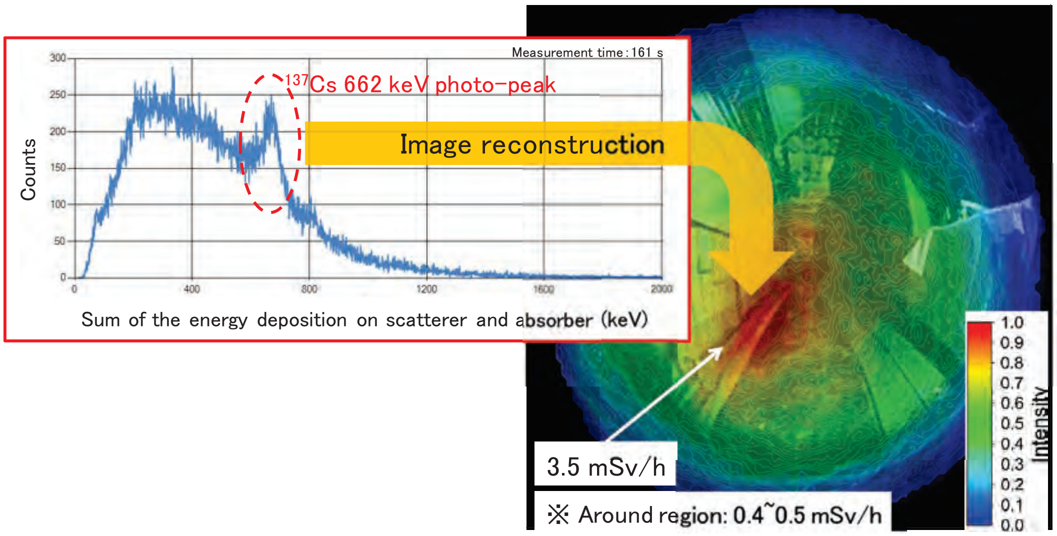


Hotspots are clearly visualized inside the FDNPS

Y. Sato, et. al., Journal of Nuclear Science and Technology, 55(9), pp. 965-970, (2018)

Imaging demonstration inside the FDNPS

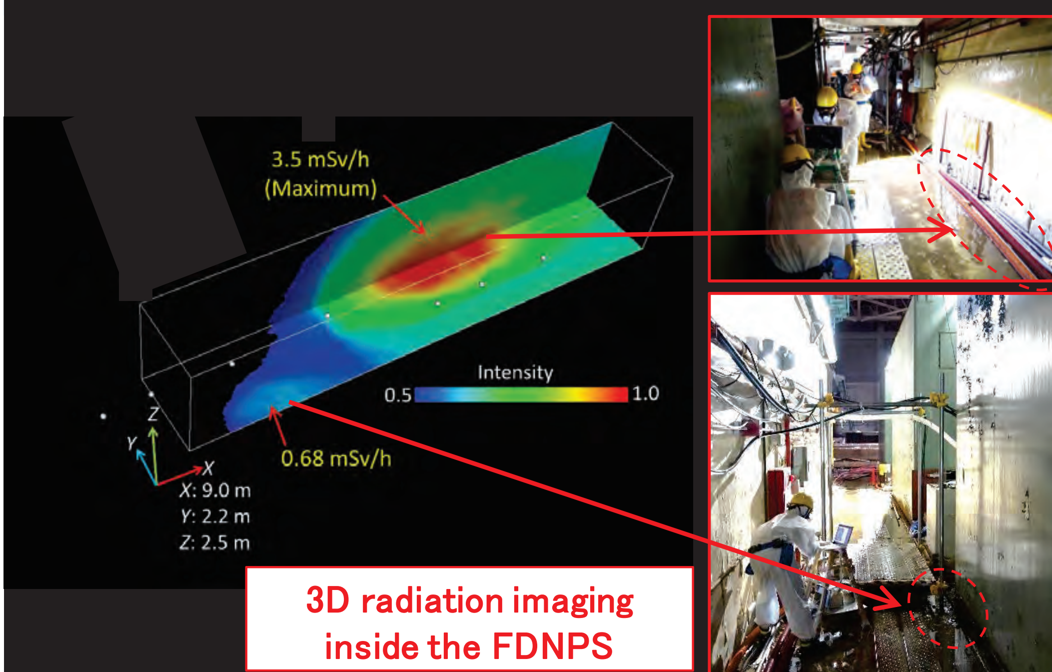
10



Hotspots are clearly visualized inside the FDNPS

Y. Sato, et. al., Journal of Nuclear Science and Technology, 55(9), pp. 965-970, (2018)

11



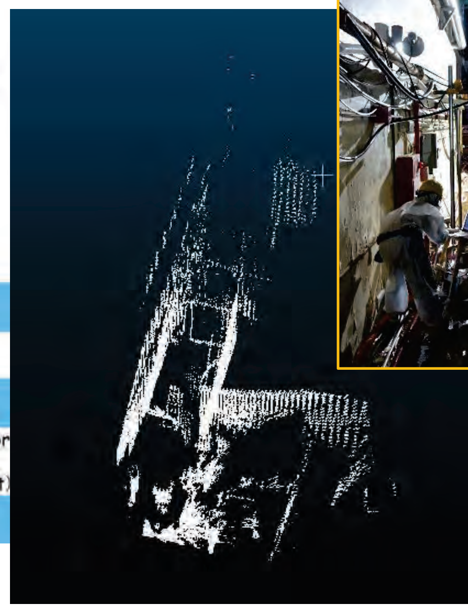
Y. Sato, et. al., Journal of Nuclear Science and Technology, 55(9), pp. 965-970, (2018)

Measurement of a 3D-model data of the building

12

Hokuyo YVT-X002(3D)

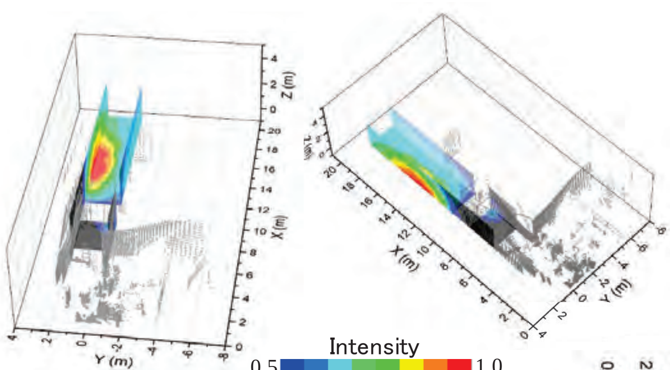
Laser Scanner	YVT-X002
horizontal detection angle	210°
perpendicular detection angle	40° (−5～35°)
optics	Laser Class 1 (Wavelength:905nm)
detection capability	0.3～8m{Black-Reflector sheet(10.0%)} 0.3～25m (White sheet)
size & weight	70mm×106mm×95mm (W×D×H) / 750g



Y. Sato, et. al., Journal of Nuclear Science and Technology, 55(9), pp. 965-970, (2018)

3-D Radiation distribution map inside the FDNPS

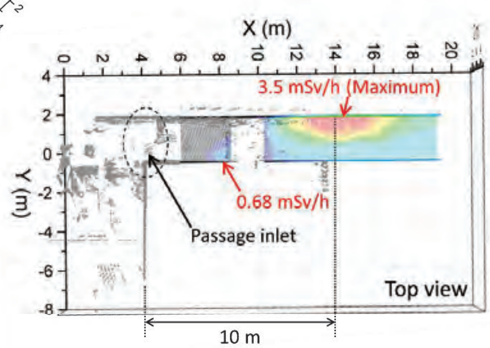
13



Radiation image
+
3D-model data

● It is possible to observe the contamination distribution from an arbitrary viewpoint.

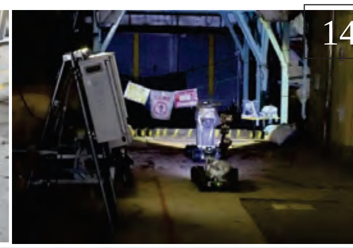
●It can be visually understood the position of the radioactive contaminations. easily.



Y. Sato, et. al., Journal of Nuclear Science and Technology, 55(9), pp. 965-970, (2018)

14

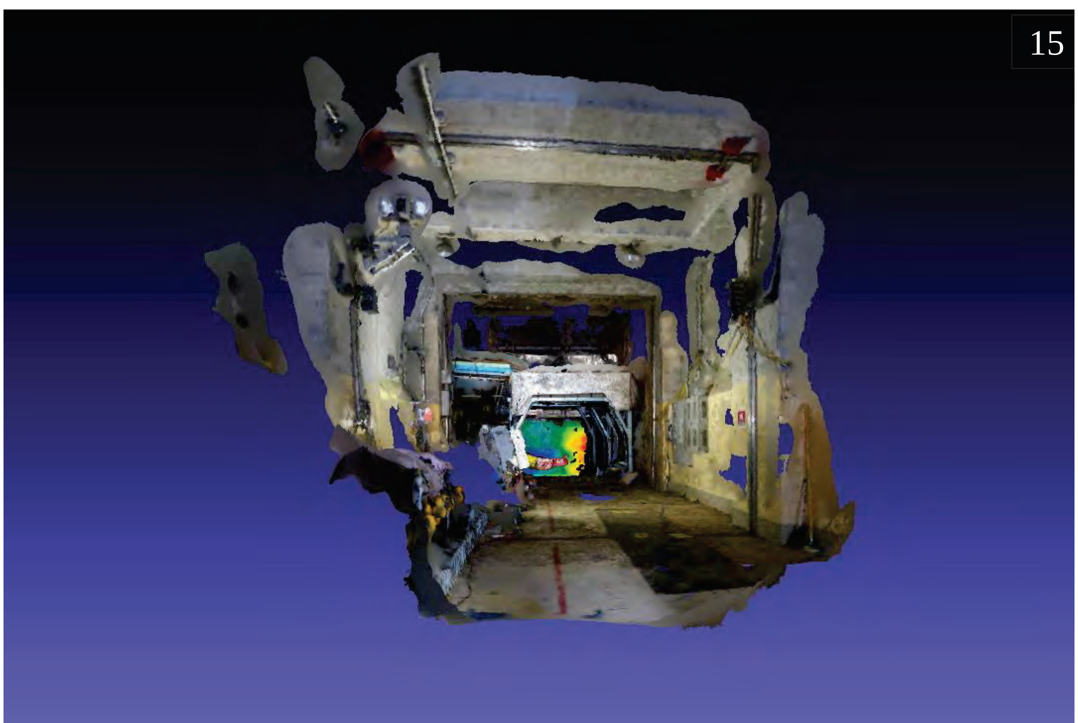
Hotspot search inside the reactor building of Unit 1



- Compton camera was installed on the crawler robot.
- Pb plate installed around the gamma-ray sensor (27 mm-thickness)
- Dose rate: 5 mSv/h >
- Measurement was remotely conducted from Main Anti-Earthquake Building.

Reconstruction of a 3D model of the work environment from multiple photographs

15



Y. Sato, et. al., Journal of Nuclear Science and Technology, 56(9-10), pp. 801-808, (2019)
JAEA Press Releases, August 28, 2018

Visualization of hot spots inside the FDNPS

16

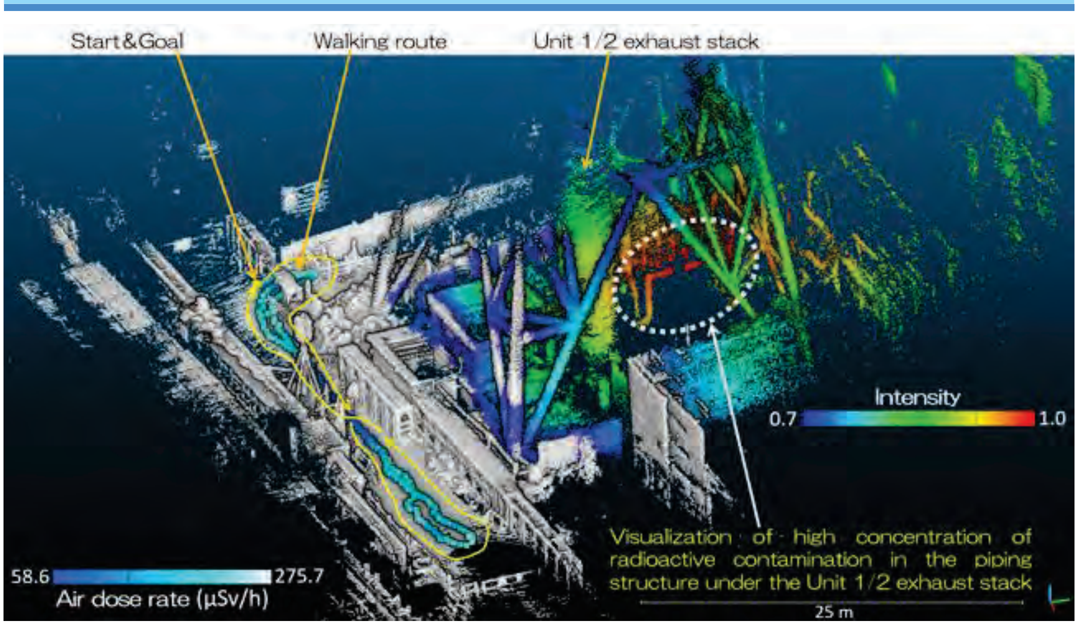
Unit 1/2 exhaust stack at FDNPS

Sato, Y., and Terasaka, Y., J. Nucl. Sci. Technol. in press, DOI: 10.1080/00223131.2021.2001391

JAEA, press release, 14th May, 2021

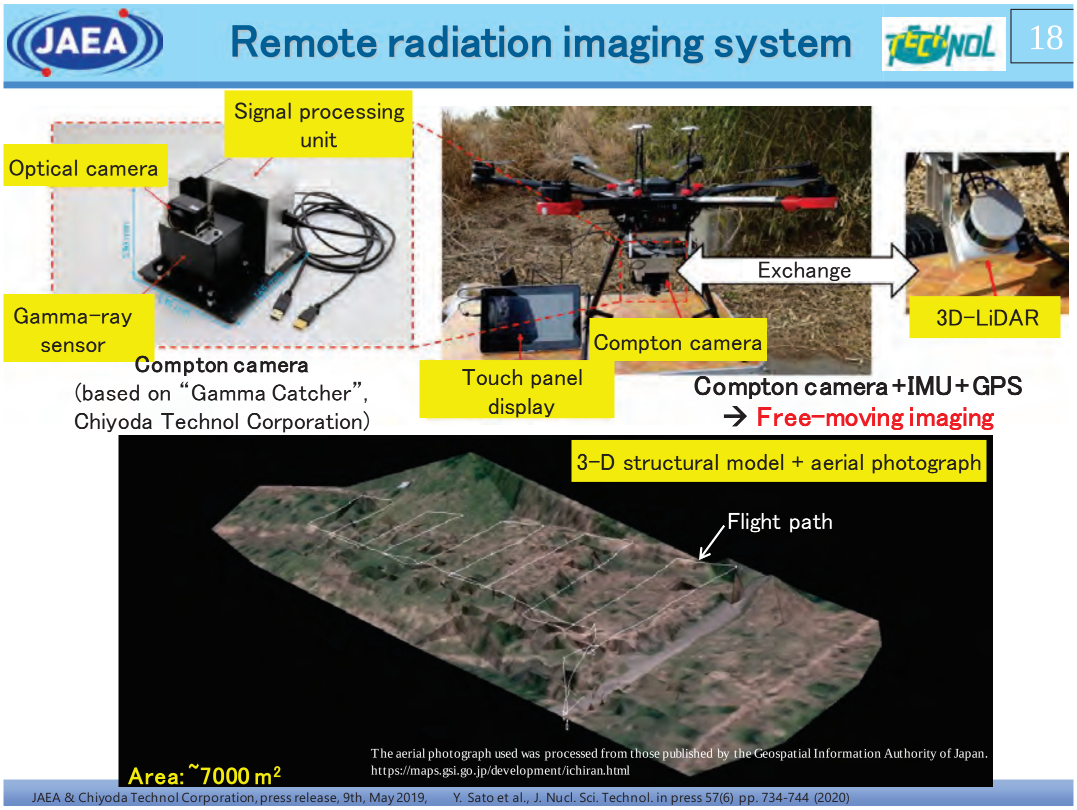
Generating a 3D map with visualized hotspots

17

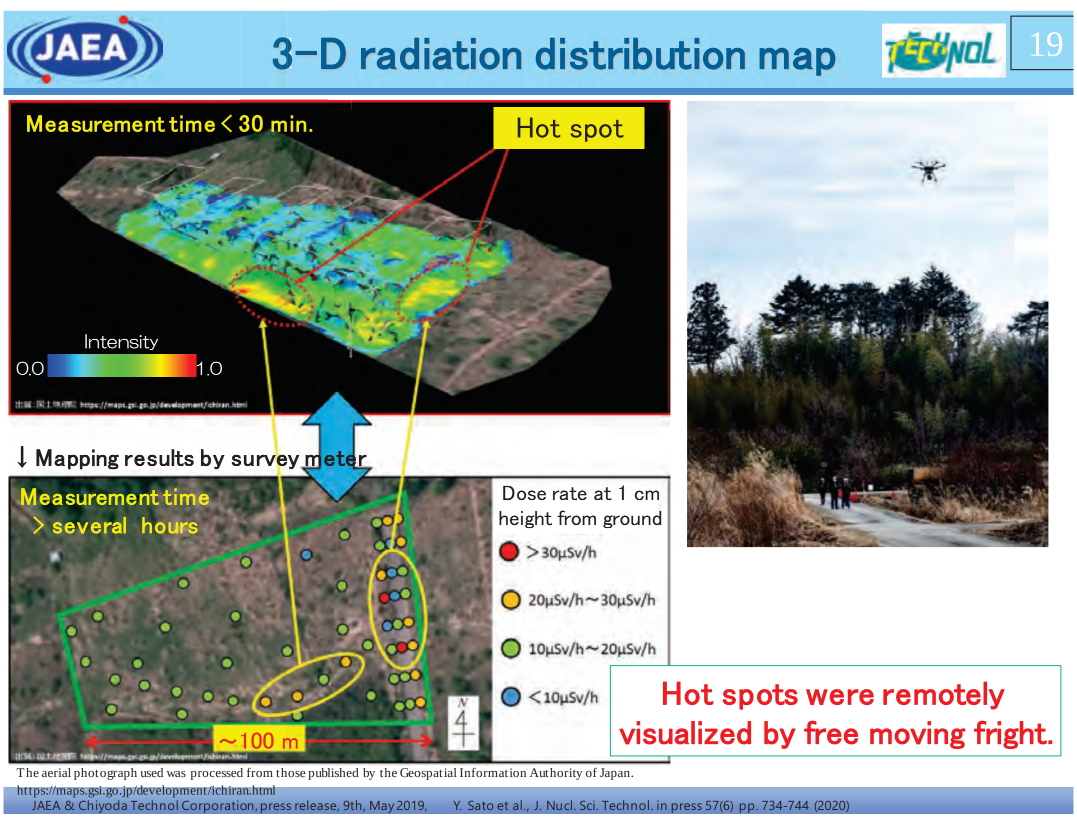


Sato, Y., and Terasaka, Y., J. Nucl. Sci. Technol. in press, DOI: 10.1080/00223131.2021.2001391

JAEA, press release, 14th May, 2021



JAEA
Remote radiation imaging system
TECHNOL
18
Signal processing unit
Optical camera
Gamma-ray sensor
Compton camera
(based on “Gamma Catcher”, Chiyoda Technol Corporation)
Exchange
3D-LiDAR
Compton camera
Touch panel display
Compton camera + IMU + GPS
→ Free-moving imaging
3-D structural model + aerial photograph
Flight path
Area: ~7000 m²
The aerial photograph used was processed from those published by the Geospatial Information Authority of Japan.
https://maps.gsi.go.jp/development/ichiran.html
JAEA & Chiyoda Technol Corporation, press release, 9th, May 2019, Y. Sato et al., J. Nucl. Sci. Technol. in press 57(6) pp. 734-744 (2020)



JAEA
3-D radiation distribution map
TECHNOL
19
Measurement time < 30 min.
Hot spot
Intensity
0.0
1.0
↓ Mapping results by survey meter
Measurement time
> several hours
Dose rate at 1 cm height from ground
> 30μSv/h
20μSv/h～30μSv/h
10μSv/h～20μSv/h
< 10μSv/h
～100 m
Hot spots were remotely visualized by free moving fright.
The aerial photograph used was processed from those published by the Geospatial Information Authority of Japan.
https://maps.gsi.go.jp/development/ichiran.html
JAEA & Chiyoda Technol Corporation, press release, 9th, May 2019, Y. Sato et al., J. Nucl. Sci. Technol. in press 57(6) pp. 734-744 (2020)

Conclusion

- We developed a 3D radiation imaging system based on a compact Compton camera.
- Succeeded in 3D visualization of high dose regions in FDNPS building.
- By combining with a robot, it is possible to measure the distribution of radioactive substances remotely.
- It is a radiation imaging system integrating radiation measurement, remote equipment and environment recognition technology. We would like to contribute to the survey of the deep part inside the reactor buildings in the future.

C.2. Introductory Presentations

C.2.1. U.S. Forensics Program Overview

U.S. DEPARTMENT OF ENERGY | Nuclear Energy

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

Damian Peko
DOE Program Manager, US Department of Energy

November 28-30, 2021

Program Overview

U.S. DEPARTMENT OF ENERGY
Nuclear Energy

Forensics Efforts Offer US Perspective to Fukushima Daiichi Examination Activities

Objectives:

- Develop consensus US input for ***high priority examination tasks and supporting research*** that can be completed with ***minimal disruption of TEPCO D&D activities***.
- **Evaluate obtained information to:**
  - **Gain a better understanding of events that occurred in each unit at Daiichi**
  - **Gain insights to reduce uncertainties in predicting phenomena and equipment performance during severe accidents**
  - **Provide insights beneficial to TEPCO Phase 2 Fuel Debris Retrieval Evaluations**
  - **Confirm/improve guidance for severe accident prevention, mitigation, and emergency planning**
  - **Update/refine original information requests**
- **Facilitate implementation of Japan-led international research efforts to support D&D.**

Graphic courtesy ANS

Motivations:

- **Provides Japan access to US expertise in plant operations, severe accident modeling & testing, and defueling & cleanup.**
- **Provides US access to full-scale, prototypic data from multiple units with distinct accident signatures.**

2

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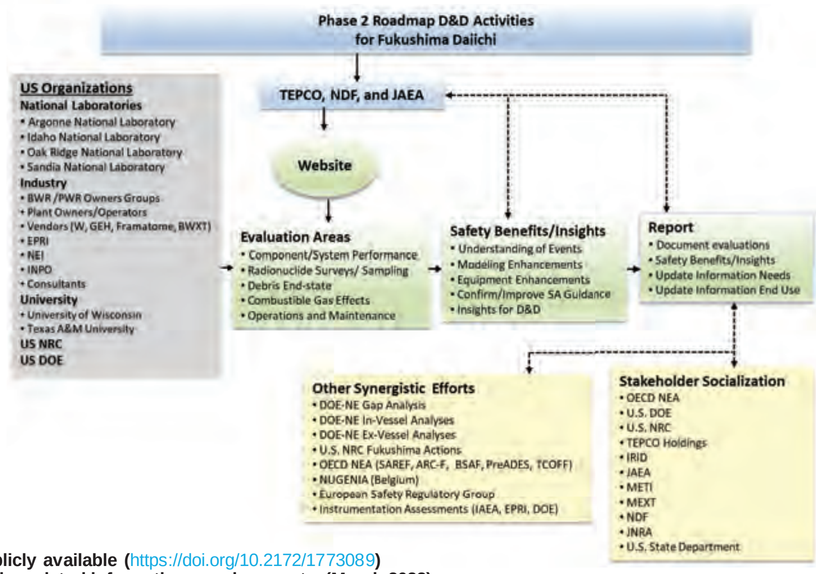
U.S. Efforts Coordinated with Phase 2 Roadmap D&D Activities and Other Programs



FY2021 report publicly available (https://doi.org/10.2172/1773089)
FY2022 report with updated information need requests (March 2022).

U.S. DEPARTMENT OF **ENERGY** Nuclear Energy

U.S Benefits from Forensics Program

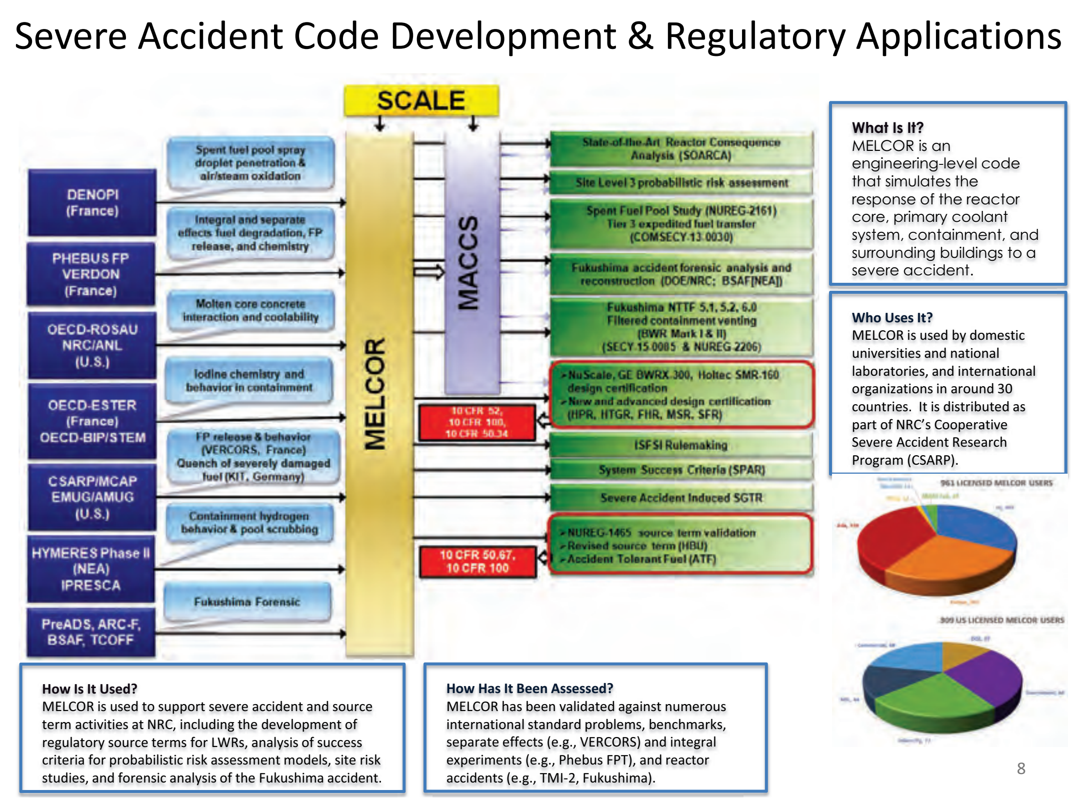
- **Continued safe and economic operation of the existing fleet is essential aspect for public acceptance of advanced SMRs (LWR and non-LWRs)**
  - **Post-Fukushima actions (FLEX, updated guidance to rely on existing instrumentation with water addition and early venting strategies) as well as severe 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, etc.)**
- **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 FY21 recommendation, DOE funded development of letter report documenting new technologies of potential interest.**

REMINDER

INFORMATION RELEASE

5

C.2.2. U.S. NRC Severe Accident Program Overview



Severe Accident Activities (1/3)

- MELCOR Development
  - MELCOR 2.2.18019 (12/2020) – ATF, non-LWR
  - MELCOR 2.2.xxxxx (December 2021 planned)
  - Code modernization (2020-2024)
    - Thermal-hydraulic package is done – focus is on COR modeling for the next two years
  - Modify code and perform analysis for Accident Tolerant Fuel (ATF), High Burn-Up Fuel (HBU), and non-light-water reactors (non-LWRs)
  - International technical meetings including Cooperative Severe Accident Research Program (CSARP), MELCOR Code Assessment Program (MCAP), Asian MELCOR User Group (AMUG), and European MELCOR User Group (EMUG)

3

Severe Accident Activities (2/3)

- OECD/NEA projects participation (examples)
  - PreADES: Preparatory Study on Analysis of Fuel Debris (2018-2021) [Assist in Fukushima decontamination & decommissioning]
  - ARC-F: Analysis of Information from Reactor Buildings and Containment Vessels of Fukushima Daiichi Nuclear Power Station (2019-2021) [Continued engagement of international severe accident and source term experts]
  - ROSAU: <u>R</u>eduction <u>o</u>f <u>S</u>evere <u>A</u>ccident <u>U</u>ncertainties (2020-2024) [Molten core-concrete Interaction experiments]
  - ESTER: <u>E</u>xperiments on <u>S</u>ource <u>Te</u>rm for delayed <u>R</u>eleases (2020-2024) [Data to support improvement and validation of source term predictive models]
- DENOPI: NRC participation in the IRSN project to investigate fuel coolability during a loss of coolant accident in SFP
- MUSA: NRC participation in Management and Uncertainties of Severe Accidents under the European Commission Horizon 2020 Project

4

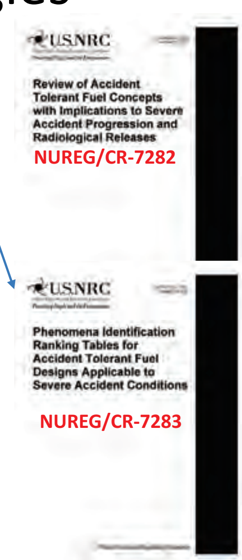
Severe Accident Activities (3/3)

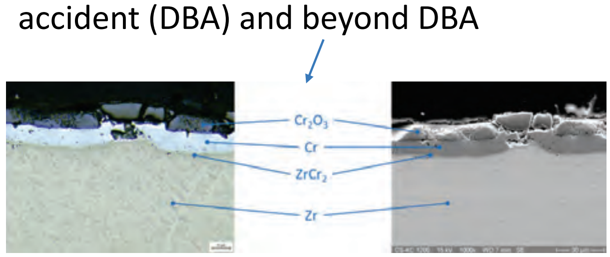
- Site Level 3 Probabilistic Risk Assessment
  - In-house MELCOR severe accident and source term analysis for reactor and spent fuel pool
- Small Modular Reactor (SMR)
  - NRC issued final safety evaluation report for NuScale with evaluation of severe accidents (August 2020)
  - Evaluation of containment performance and severe accident for Holtec SMR-160 and GE-BWRX-300
- Revision of Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactor"
  - Fuel handling accident in-house analyses to formulate the technical basis to revise these accident scenarios
  - Re-evaluation of settling velocity distribution and the multi-group method
  - Evaluation of the Impact of Fuel Fragmentation, Relocation and Dispersal for the Radiological Design-Basis Accidents

5

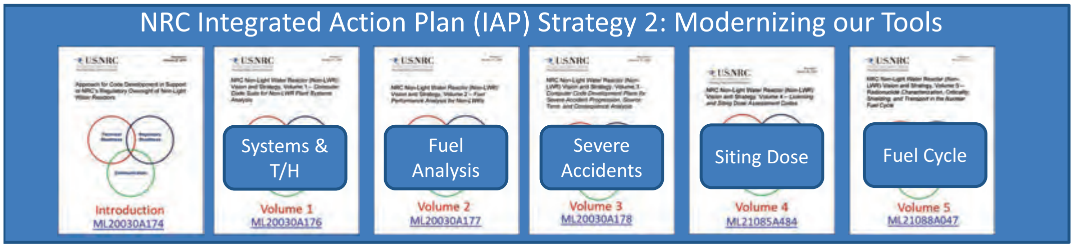
Advanced Fuel Technologies

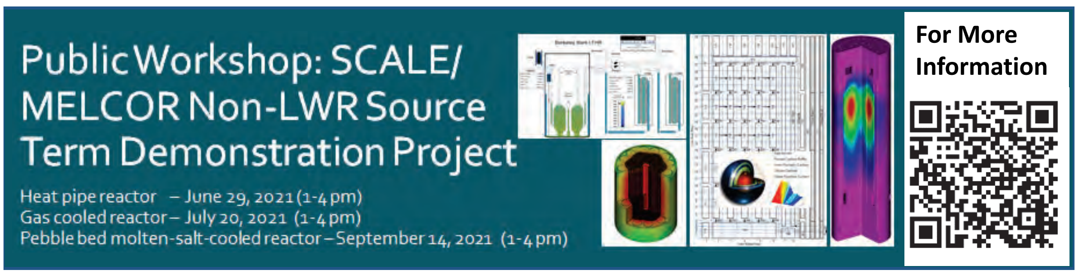
- Panel of international severe accident experts Phenomena Identification and Ranking Tables (PIRT) that addressed significant phenomenological issues to improve MELCOR
- Source term calculations for HBU/HALEU fuel
- QUENCH-ATF: Experiments for ATF cladding materials in the QUENCH facility at Karlsruhe Institute of Technology (KIT) – Near term chromium-coated cladding under design basis accident (DBA) and beyond DBA





Advanced Nuclear Technology Research





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*, soon to be issued, 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

C.2.3. FY2021 Program Results and International Lessons Learned Overview

U.S. DEPARTMENT OF ENERGY | Nuclear Energy

FY21 Findings and Recommendations and International Lessons Learned Overview

Joy Rempe
Technical Lead, Rempe and Associates, LLC

November 28 - 30, 2021

FY21 Overview

U.S. DEPARTMENT OF ENERGY
Nuclear Energy

FY21 Findings and Recommendations

- **Finding 1 -** Available Fukushima-related information from Japan and discussions of this information at DOE Forensics Expert Panel meetings continue to benefit the U.S. operating fleet.
  - ***Associated Recommendation*** - U.S. organizations should continue to monitor and evaluate information obtained from the affected reactors at Daiichi. Important insights continue to come from examinations at Daiichi that affect accident management strategies and reduce uncertainties in systems analysis codes.
  - ***Associated Recommendation*** - In future years, U.S. Forensics Expert Panel Meetings should include options for in-person and virtual participation.
- **Finding 2 -** U.S. evaluations of information from Fukushima are of interest to several organizations within Japan.
  - ***Associated Recommendation*** - Additional consideration should be given to the proposal of using coolant suspension test data as a basis for reducing water injection rates, and possibly terminating water injection, to the affected units at Daiichi. However, results from these scoping calculations should first be confirmed using more detailed models available in systems analysis codes.
  - ***Associated Recommendation*** **-** Representatives from the NRAJ and the U.S. Forensics Effort should continue to communicate as new information related to combustible gas generation and ignition becomes available.
  - ***Associated Recommendation*** **-** Topic Area 4 investigations should remain focused on identifying and reducing, where possible, uncertainties that impact accident management strategies.
- **Finding 3 -** The Terry™ Turbine Expanded Operating Band (TTEXOB) project offers the potential for important reactor safety insights and reduced plant operating costs.

2

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Nuclear Energy

FY21 Findings and Recommendations (cont.)

- **Finding 4 -** Japan D&D efforts include multiple demonstrations of effective use of new technologies, such as unique robots, gamma-ray imaging cameras, drones, three-dimensional visualizations of radiation levels and temperatures, and muon technologies. In addition to providing information, not originally anticipated when U.S. experts originally defined information requests, applications of these new technologies could provide many other important insights in the areas of maintenance requirements, radiation protection methods, reactor design, and siting requirements. These insights apply to the existing fleet, new reactor design and siting, and radiation cleanup activities.
  - ***Associated Recommendation*** - To increase the impact of information from Daiichi, an information bulletin should be prepared regarding radiation protection 'best practices' learned from Daiichi D&D activities. With participation by BWROG, EPRI, and NRC, the DOE forensics program should lead this effort during FY2021.
- **Finding 5 -** Important new insights are expected from on-going and planned near-term investigations.
  - ***Associated Recommendation*** - Further evaluations should be performed to understand the cause for the observed 1F3 RPV pressure drop, considering the effects of RCP leakage, RCIC performance (and impact of operator actions during the 1F3 event to conserve battery power), and code nodalization.
  - ***Associated Recommendation*** - Given the importance of SRVs in EOPs and SAMGs, additional analyses, and possibly testing, should be performed to evaluate the risk impact of SRV behavior due to low N2 accumulator pressure. The performance of SRVs, which may vary with vendor, design, and environmental conditions, should be better understood and communicated to plant operators. Analyses should consider the effects of PCV pressure and temperature, operator actions, core decay heat, RCIC steam requirements, and RCIC cold water injection. Additional radiation measurement data from plant piping, if available, may provide insights about valve operation.
  - ***Associated Recommendation*** - Given the importance of containment venting in the EOPs and SAMGs, additional analyses, and possibly testing, should be performed to evaluate the risk impact of vent valve and vent line rupture disc behavior. These evaluations should consider the height and timing of radiation released from each unit.

3

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FY21 Findings and Recommendations (cont.)

- **Finding 6 -** No new information requests were identified by U.S. experts.
- **Finding 7 -** Capabilities proposed for new CLAD hot cells will provide data required for effective D&D and for gaining insights needed to enhance reactor safety.
  - ***Associated Recommendation*** - Additional consideration should be given to the U.S. request for obtaining examination information that could be used to characterize debris morphology (e.g., porosity, shape distribution, size distribution). Such data are important for future D&D activities and for accident mitigation strategies.
- **Finding 8 -** Participants agreed that systems analysis codes have demonstrated a good ability to capture the main trends of the accident progressions up through core degradation, including nuanced differences between the 1F1, 1F2, and 1F3 sequences, but uncertainties remain in simulating certain aspects of the accidents.
  - ***Associated Recommendation*** - To the extent possible, funding agencies should continue to document insights from the affected reactors at Daiichi in reports and update systems analysis code models to reflect risk-important insights. Implementation of these insights and new models into these codes can significantly improve our ability to predict the progression of future accidents, prevent loss of knowledge, and maintain the code state-of-practice.
  - ***Associated Recommendation*** - New RCIC models in MAAP and MELCOR should be benchmarked against Tennessee Valley Authority data in which the RCIC system ran on April 27, 2011 after a tornado (subsequent to the meeting, this plant data was provided to EPRI and SNL).
  - ***Associated Recommendation*** - An activity should be completed to compare risk-important MELCOR and MAAP input parameters and associated uncertainty distributions.
  - ***Associated Recommendation*** - A water level sensor model, similar to the one implemented in MELCOR, should be implemented in MAAP.

4

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Recent International Activities Regarding Daiichi Findings, Lessons Learned, and Recommendations

5

International Perspective

U.S. DEPARTMENT OF ENERGY
Nuclear Energy

Lessons Learned – An International Perspective

- **Findings, lessons learned, and recommendations from Fukushima (as well as prior accidents) have been emphasized in seminars and reports from several domestic and international organizations, including the following international sources:**
  - OECD, "Fukushima Daiichi Nuclear Power Plant Accident, Ten Years On Progress, Lessons and Challenges", March 2021.
  - IAEA, "Decade of Progress after Fukushima Daiichi Building on the lessons learned to further strengthen nuclear safety", March 2021.
  - IRSN, "Anticipation and Resilience Considerations a Decade after Tthe Fukushima Daiichi Accident", 2021.
  - IAEA, International Conference on a Decade of Progress after Fukushima-Daiichi: Building on the Lessons Learned to Further Strengthen Nuclear Safety" November 8-12, 2021 (Presentations by T. Kikihara, Ambassador of Japan to IAEA, and H. Yamana, President, NDF)
  - NDF, "The 5th International Forum on the Decommissioning of the Fukushima Daiichi Nuclear Power Station", October 31, 2021-November 1, 2021.
- **Selected items summarized on Slides 7 and 8.**
- **Initial thoughts regarding U.S. activities completed and possible next steps provided on Slide 9 (for discussion during our Closing Session on Tuesday).**

6

U.S. DEPARTMENT OF ENERGY
Nuclear Energy

Additional Prevention and Mitigation Measures

- **Holistic systematic approach to safety:**
  - Consider systems, structures, and components (SSCs) as well as Institutional Defense-in-Depth
  - Consider prior nuclear and non-nuclear incidents
  - Increased equipment robustness (diverse means for water addition, power, venting, filtration, monitoring) to address unanticipated challenges
  - Continue R&D to gain consistent understanding of Fukushima accident progressions and SSC performance, reduce uncertainties in systems analysis codes, and improve operator guidance and training
  - Consider cost/benefit of any proposed countermeasures
- **Encourage regulators to adhere to international safety standards (fundamentals, requirements, and guides)**
- **Knowledge management (archival in an easy-to-access system) and transfer (international, younger generation, new plant designs, advanced fuel design)**
- **Improve realism in emergency planning (consider events involving multiple units, multiple sites, increased infrastructure degradation, increased contamination, and longer duration releases, and impact of international input)**

7

International Perspective

U.S. DEPARTMENT OF ENERGY
Nuclear Energy

Longer-term Response Measures to Improve Stakeholder Interface and D&D Processes

- **Recognize and address issues (economic, psychological, and physical) faced by evacuees**
  - Monitor returnee exposure levels to address returnee concerns regarding radiation from other communities, stored waste, etc.
  - Improve stakeholder communication and increase stakeholder involvement in decision-making (provide more data, provide information to interpret data, acknowledge limitations on knowledge)
- **Determine criteria for:**
  - Fair and timely compensation for loss of home, employment, and community
  - "Acceptable" contamination levels and "required" infrastructure (gas, electricity, water, health care, shops, etc.) and economic recovery actions for return
  - Characterization and management of contaminated waste (lack of standards)
- **Take holistic D&D approach that considers risk reduction, cost, schedule, worker exposure, and waste generation**
  - Consider future hazards (external events, aging, etc.) and uncertainties
  - Appropriate quality management
  - Develop and implement advanced D&D technologies and information systems to compile, store, and present data used for decision-making and informing stakeholders.

8

Initial Thoughts Regarding U.S. Actions

- **U.S. plant owners/operators and government agencies have:**
  - Updated external hazard assessments
  - Increased equipment robustness (FLEX, SFP instrumentation, hardened containment vent)
  - Improved operator guidance and training
  - Benchmarked (and revised if warranted) models in systems analysis codes (MELCOR, MAAP)
  - Continued to consider prior nuclear and non-nuclear events
  - Continued R&D efforts to improve understanding of events and SCC performance during these events
  - Increased severe accident knowledge management activities (archiving TMI-2 documents, DOE Forensics Effort, etc.)
- **Any further U.S. actions appropriate?**
  - Continue to consider forensics information from the affected reactors at Daiichi
  - Continue to update suite of emergency response guidelines
  - Learn from plant exercises demonstrating Post-Fukushima actions
  - Maintain active participation in FLEX and SAFER Working Groups
  - Consider impact of climate on plants

9

FY22 Agenda

Reactor Safety Technology Expert Panel Forensics FY22 Meeting Topics*

- **NDF – Strategic Plan 2021 – Ito**
- **NRA – Japan Activities - Yasui**
- **TEPCO Activities Update and Discussion – Mizokami, Owada, and Cibula**
- **JAEA Activities Update and Discussion – Washiya, Koyama, Ikeuchi, and Sato**
- **Introductory Remarks by NEI, DOE, and NRC – Butler, Peko, Esmaili, and Rempe**
- **Related EPRI Activities – Nudi**
- **Recent MELCOR Development Activities – Luxat**
- **Topic Area Lead discussions:**
  - **Area 1 - Components/System Performance – Robb and Gabor**
  - **Area 2 - Radionuclide Surveys/Sampling – Luxat**
  - **Area 3 - Core Debris Location Evaluations – Farmer and Plys**
  - **Area 4 – Combustible Gas Effects - Luangdilok**
  - **Area 5 – Operations and Maintenance – Bunt, Ellison, and Williamson**
- **Updates to Information Requests and Next Steps - All (led by Rempe)**

*See Agenda for Presentation Schedule

Link to FY22 Agenda, Viewgraphs, FY21 Report & Draft FY22 Report (when available): https://1drv.ms/u/s!ApliColj18LBgfgt4Z_T5C_02zTtJQ?e=iOcpvY

10

C.3. System Analysis Code Updates and Other Related Activities

C.3.1. Related EPRI Activities

Related EPRI Activities

Matt Nudi
Risk & Safety Management Program

Reactor Safety Technology Expert Panel Forensics Meeting
November 29, 2021

www.epri.com

Review of Related FY21 Report Recommendations

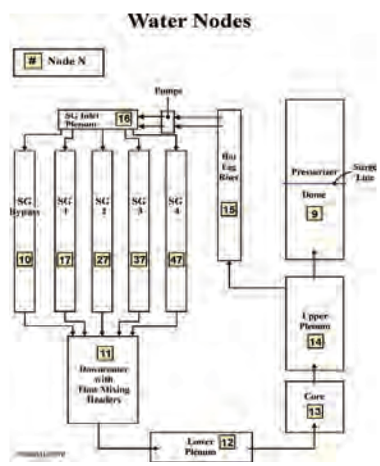
- Alignment of risk-important parameter uncertainties in SA codes
  - 2021 MAAP uncertainty work extended methodology developed for Fukushima assessments
  - Work supported European H2020 MUSA Project with participants using MAAP, MELCOR, AZTEC, etc.
  - EPRI to support meeting planned for 1Q 2022 through MUSA Project to collaborate with other SA code developers on this topic
- Benchmarking of MAAP's RCIC model to TVA operation data
  - Scope was not able to be completed in 2021 but is expected to perform these comparisons 2022
- MAAP Implementation of BWR water level instrumentation model
  - MAAPv5.06 scope set at the time of 2020 meeting, but this enhancement is being tracking for consideration in MAAPv6.00

2 www.epri.com

Ongoing MAAP Development

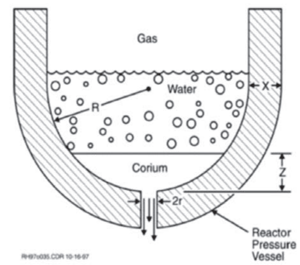
MAAP version 5.06

- Release included variety of code enhancements to meet near-term user needs ahead of modernized MAAP6 code
- Included modifications for integral Korean SMR design and key enhancements:
  - Extend Fukushima BWR Modifications 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



MAAP version 5.06 – Ex-vessel Relocation Model

- Originally implemented in MAAP 5.04 for BWRs
  - Intended to address lessons learned from Fukushima where it was observed that MAAP5 assumptions limited applicability for debris relocation assessments
- Benefits of the ex-vessel relocation model include
  - Enhanced modeling to account for how debris properties in lower head influence debris pour rate
  - Captured the potential for debris relocation from multiple failure points
- MAAP 5.06 extended functionality to PWRs
  - In absence of PWR validation data, model is intended to provide method for exploring sensitivity of lower head relocation dynamics

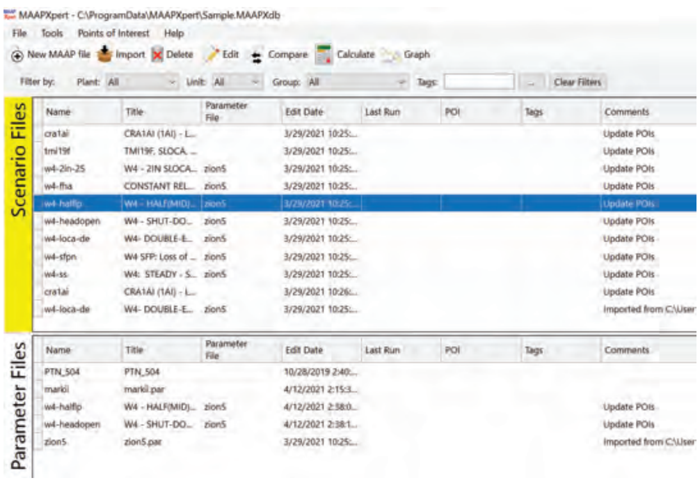


MAAP6 - Code Modernization

- MAAP5 has been ported from Fortran to C++ and tested -> **MAAP6**. From this point, the modernization of the MAAP6 code will:
  - Reorganize the MAAP code base to enhance the testability of the code
  - Improve code base to make a single source for PWR, BWR, and CANDU (and later VVER) builds.
  - Increase the robustness of the MAAP code base
  - Reorganize the existing MAAP code base in a manner to support more cost-effective and rapid enhancement to the MAAP code to support applications for the operating fleet as well as evaluation of <u>advanced nuclear technologies</u>
  - Enhance performance of the code base
- The primary means to achieve these goals is the adoption of modern software development practices, including reorganization of the existing code base to simplify and provide more modularization.

MAAP6 – Advanced User Functionality

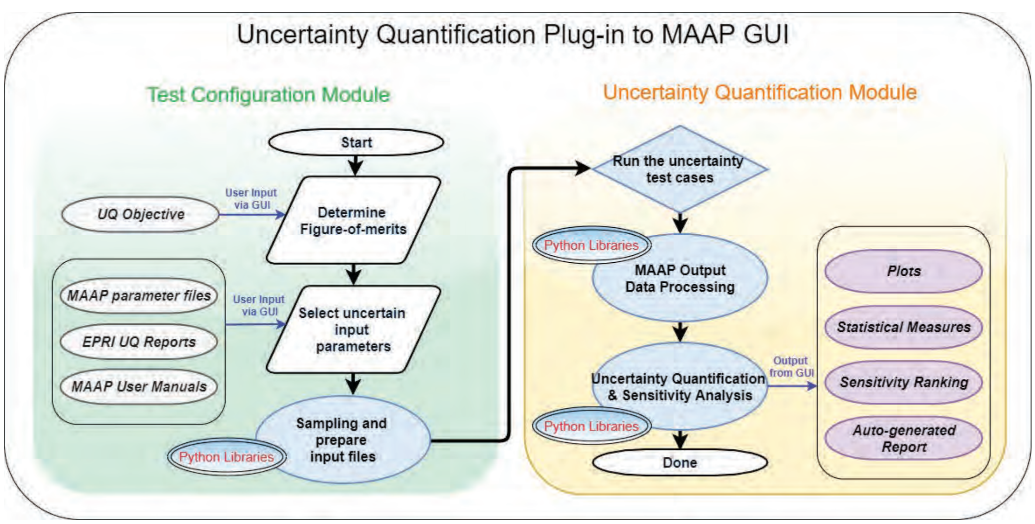
Graphical User Interface



Test Suite

  EPRI

Uncertainty Quantification and Analysis using MAAP



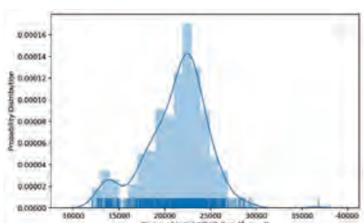
  EPRI

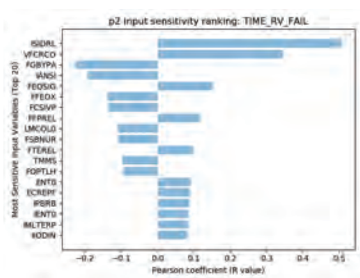
Extension of Fukushima Uncertainty Evaluations

 www.epri.com

Uncertainty Quantification

- Fukushima assessments provided framework for MAAP uncertainty assessments
  - Required extension to PWRs and MkII/MkIII BWRs
- EPRI released updated report in 2021 covering PWR/BWR plant types (3002020762, publicly available)
  - Assessed phenomenological uncertainty across range of mitigated & un-mitigated sequences
- Extension of Fukushima approach demonstrated that sensitive inputs can be both plant and scenario specific
  - Report provides summarized list of sensitive parameters and their distributions which are generally applicable for the MAAP code





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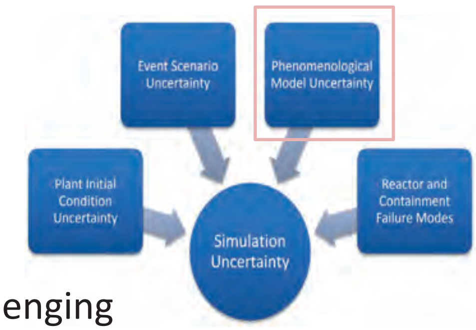
Insights from Uncertainty Analysis Approach Development

- Context of the accident scenario has potential to outweigh phenomenological uncertainty
- Uncertain parameter distributions should reflect the state of knowledge of that parameter
- Additional work is needed to correlate uncertain inputs to better understand impact on conclusions from UQ
- Identification of common approach to representing and communicating uncertainty results continues to be challenging



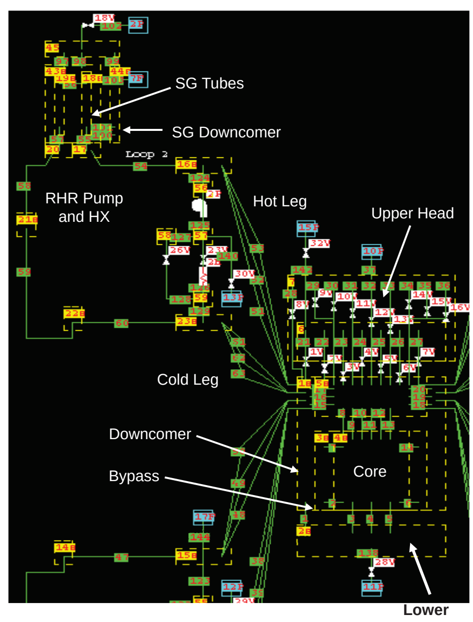
Ongoing Uncertainty Research

- Work in 2021 developed a general design plan for providing uncertainty quantification capabilities within MAAP6 GUI
- Future investigations include
  - Research to further separate impacts of code numerical noise from phenomenon uncertainty
  - New methods for extracting actionable insights from the uncertainty analysis, to be ultimately integrated with MAAP GUI
  - Incorporation of uncertainty into MAAP validation test suite to support prioritization of future development and/or code benchmarking

Applicability of GOTHIC for Fukushima Decommissioning Activities

General Attributes

- GOTHIC simulates:
  - Multiphase compressible flow with heat and mass transfer
  - Transient mass, momentum and energy transport with diffusion and turbulence effects
  - Fission product transport and release in the liquid, vapor, and droplet fields as well as on surfaces and filters
    - Modal representation with multiple drop fields for aerosol modeling
- It provides flexible nodalization
  - Domain decomposition approach
  - Seamlessly integrate 0-D to 3-D representations in a single model
- Integrated software package:
  - Graphical user interface (GUI)
  - Solver
  - Post-processor



Evolution of GOTHIC Aerosol Modeling Capability

GOTHIC 6.0
Implemented tracking for radioactive isotopes in the vapor phase

GOTHIC 7.2b
Solid particle tracking in the liquid phase

GOTHIC 8.0
Added the ability to account for multiple drop/aerosol fields

GOTHIC 8.2
Added capability to account for entrainment of solid particles in drop/aerosol fields

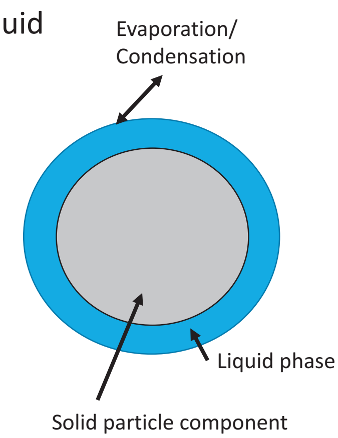
GOTHIC 8.3
Further improvements for modeling fully dry aerosols

GOTHIC 8.4
Pool scrubbing, hygroscopic effects, and improved computational performance

Incremental development of GOTHIC aerosol modeling capability to fully account for aerosol transport phenomena with proven applications along the way.

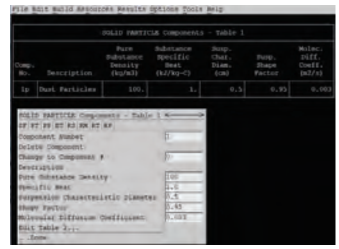
GOTHIC for Modeling Fission Product Transport and Release

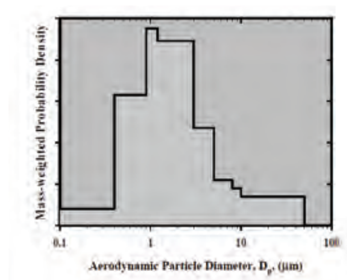
- Tracers
  - Track as part of liquid (both films and drops), vapor or conductor surfaces
  - No Impact on fluid properties and move with the carrier fluid
  - Can include radioactive decay, progeny and heat release
- Components
  - Generalized Filter
  - Charcoal Filter
  - Dryer/Demister
- Aerosols
  - Any number of fields each with their own size distribution
  - Transport and interactions based on aerosol mechanics
  - Can consist of liquid, solid particles or both



Aerosol Transport – Solid Particles

- GOTHIC contains the capability to model “solid particles” with user defined input options for:
  - Density, specific heat, suspension characteristic diameter, shape factor, etc.
- Able to model multiple constituents to account for variability in corresponding aerosol composition
- Can define multiple aerosol fields to account for a wide range of user-defined debris/aerosol compositions
- Defined aerosol fields implemented through use of boundary conditions
  - Ability to define geometry, flow paths, and aerosol composition adds analysis flexibility





Applicability to Fukushima Decommissioning Activities

- Recent GOTHIC development allows for application to variety fission product and solid particle transport analyses
  - Can complement integral SA code results
- As confirmatory SGTS and shield plug contamination measurements become available, GOTHIC could be applied to better understand the FP transport experienced at Fukushima
- Future work includes
  - EPRI support of FACE project which aims to investigate behavior of radioactive particles generated during fuel retrieval operations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19
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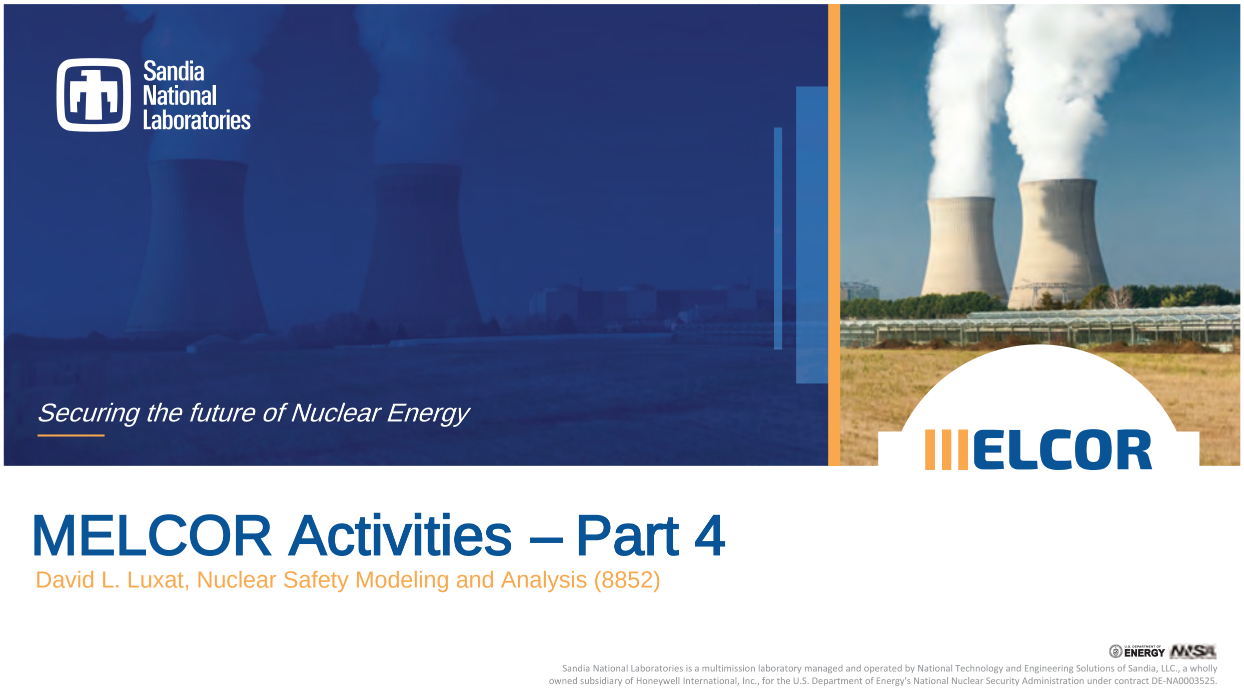
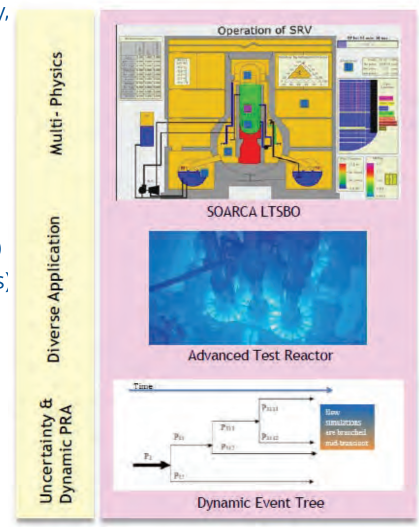
EPRI

C.3.2. MELCOR Updates and Related SNL Activities



MELCOR History and Introduction

- Fully integrated, multi-physics engineering-level code
  - Thermal-hydraulic response in the reactor coolant system, reactor cavity, buildings
  - Core heat-up, degradation, and relocation
  - Core-concrete attack
  - Hydrogen production, transport, and combustion
  - Fission product release and transport behavior
- Diverse application
  - Multiple core designs
  - Models built from basic code constructs
  - Adaptability to new or non-traditional reactor designs (ATR, Naval, VVER)
- Validated physics models (ISP's, benchmarks, experiments, accidents)
- Uncertainty analysis & dynamic PRA (fast-running, reliable, access to
- User convenience
  - Windows/Linux versions
  - User utilities and post-processing/visualization capabilities
  - Extensive code documentation



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MELCOR RCIC System Modeling & Analysis

- FY21 was an extended period-of-performance (no cost extension)
- Milestone 7 work focused on input model development for the ZS-1, GS-2, and generic BWR
  - ZS-1
    - New experimental data available from TAMU
    - New MELCOR code capabilities employed
    - Inputs configured for uncertainty analysis demonstration with the DAKOTA code
  - GS-2
    - New experimental details available
    - Some of the ZS-1 modeling improvements apply
    - Set up for future comparison with experimental observations/data
  - Generic BWR
    - Revolved around the question of self-regulating behavior
    - Three observed modes of self-regulation
    - Identification of particularly influential parameters and/or modeling choices
  - FY21 summary report has been publicly released

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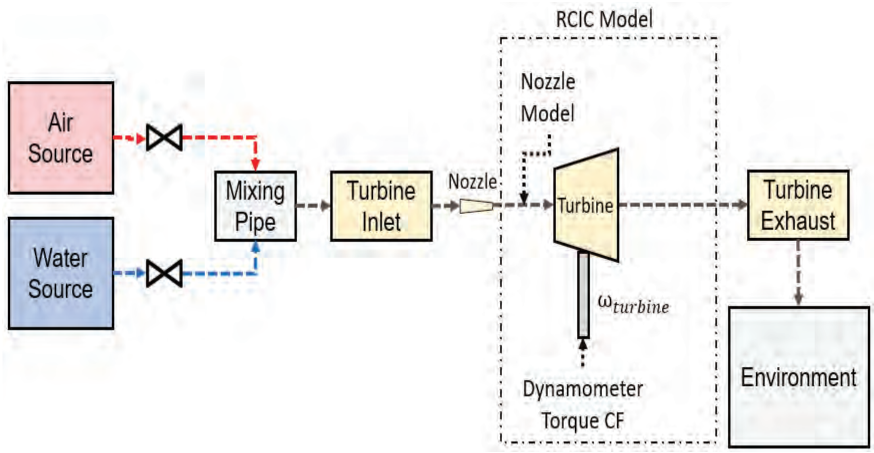
MELCOR RCIC System Modeling & Analysis

- ZS-1 Model



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MELCOR RCIC System Modeling & Analysis

- ZS-1 Modeling
  - Temperature and pressure set while air/water mass flow modulated
  - Experiments suggest turbine losses include a linear term:

$$\tau_{loss} = c_{windage}\omega^2 + c_{linear}\omega + c_{constant}$$

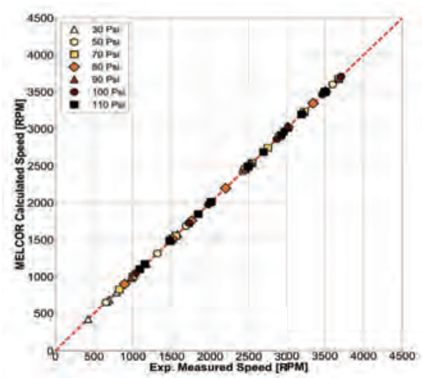
  - Loss coefficients experimentally determined
  - Ideal gas mixture nozzle flow model employed along with new systems-level RCIC mechanistic models
  - c_{torque} found with MELCOR simulations
    - Deterministic calibration
      - Can do this per experiment
      - Can derive a single value considering all experiments → c_{torque}=0.3453
    - Bayesian calibration
      - MELCOR/DAKOTA coupling
      - Uncertain parameter is c_{torque}
      - Mean value compares well with deterministic value → c_{torque}=0.3467

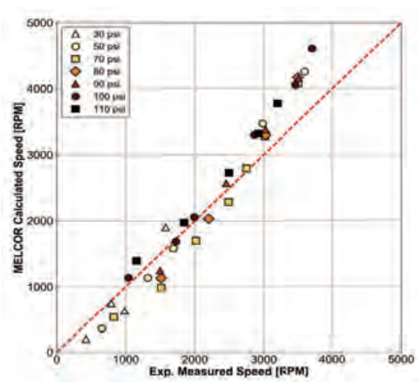
Coefficient/Constant	Value
$c_{windage}$	1.39×10^{-7}
c_{linear}	2.3×10^{-4}
$constant$	3.8×10^{-2}

6

MELCOR RCIC System Modeling & Analysis

- ZS-1 MELCOR results vs experimental data
  - Left: Per-experiment c_{torque} calibrations (proper speed with proper calibration per experiment)
  - Right: c_{torque} calibrations across all experiments





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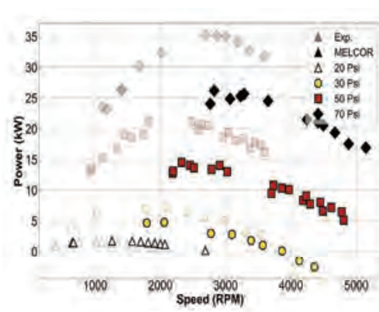
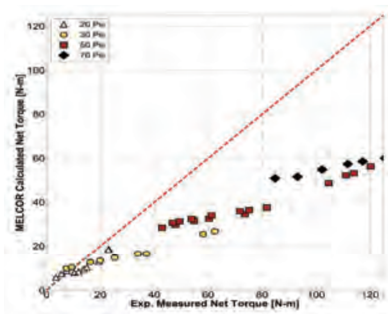
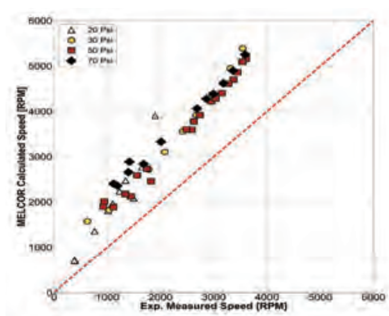
MELCOR RCIC System Modeling & Analysis

- ZS-1 c_{torque} in GS-2 simulations?
  - Loss data not yet available and/or incorporated
  - Using ZS-1 coefficients in GS-2 leads to comparatively poor MELCOR/experiment agreement
    - Repeat ZS-1 analysis on GS-2 with revised loss coefficients to obtain a GS-2 turbine torque multiplier
    - Or…propose some sort of correction to ZS-1 coefficients?







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MELCOR RCIC System Modeling & Analysis

MELCOR

- Generic BWR
  - Geared towards RCIC modeling
  - GS-1 type Terry turbine
  - 5 circumferential nozzles around rotor wheel (high and low)
- No overspeed allowed
- Pump NPSH failure allowed
- SBO w/ DC loss at 2 hr
- MELCOR mechanistic RCIC models configured and active
  - Turbine - pressure stage and velocity stage(s), friction and losses, etc.
  - Pump - homologous curves, friction and losses, etc.
  - Shaft - torque-inertia equation for synchronous speed considering turbine and pump sides

Parameter	Value
RCIC Turbine Specifications	
Turbine Radius	0.3048 m
Turbine CV Volume	0.106 m^3
Turbine Moment of Inertia	10.0 kg-m^2
Turbine Friction Torque	10.0 N-m
Turbine Bucket Exit Angle	30°
Turbine Nozzle Diameter	12.7 mm
Number of Nozzles	5
RCIC Pump Specifications	
Rated Pump Speed	4287.0 RPM
Rated Pump Head	7.59 MPa
Rated Pump Torque	448.8 MPa

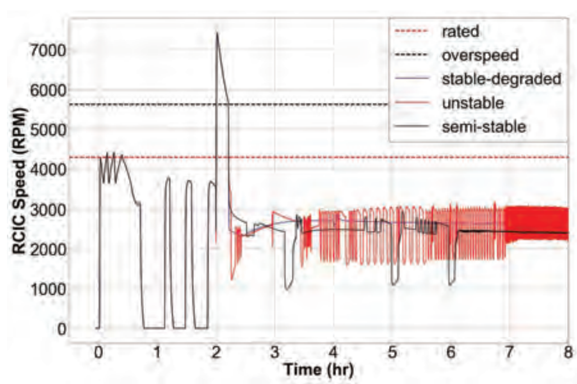
Parameter	Value
Rated Pump Power	2.0147×10^5 W
Rated Pump Injection Rate	0.03886 m^3/s
Pump Moment of Inertia	30.0 kg-m^2
Other RCIC Related Specifications	
RCIC CF Target Injection Rate	38.733 kg/s
RCIC CF Target Relative Downcomer Level	2.54 m

9

MELCOR RCIC System Modeling & Analysis

MELCOR

- Observe three self-regulating modes of operation (pending model input parameters)
  - Stable, degraded – Constant turbine speed and stable (degraded) water injection to RPV
  - Unstable – Oscillations in turbine speed and RPV injection according to steam line flooding
  - Semi-stable, degraded – Stable, degraded with potentially significant speed/injection fluctuations
- Nozzle modeling choices make the difference
  - Single “lumped” flow path
  - Several flow paths at different elevations
  - Steam line and steam chest modeling
  - Experimental insights can help?



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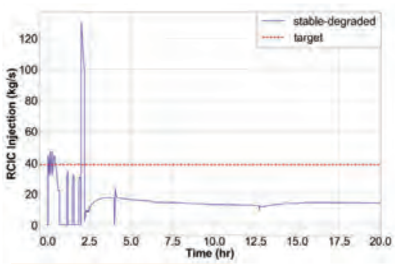
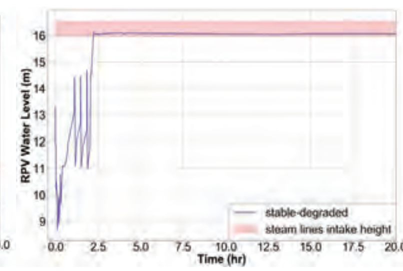
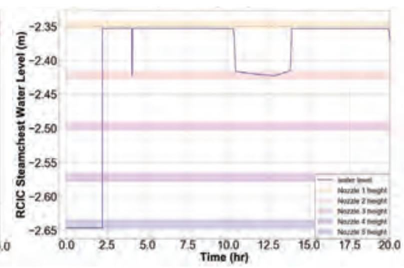
MELCOR RCIC System Modeling & Analysis

- Stable, degraded – Constant turbine speed and stable but degraded water injection to RPV
  - Five nozzles at five distinct locations, circumferentially situated about the rotor wheel
  - Nozzles 3, 4, and 5 (lowest) submerged, nozzle 2 mostly submerged, nozzle 1 always uncovered
  - Top nozzle admits steam, preserves turbine performance, pump flow, and water level







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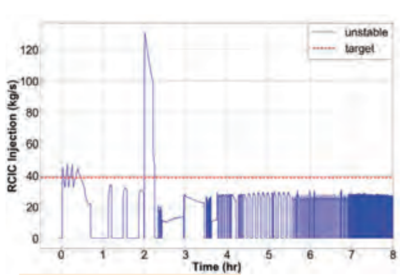
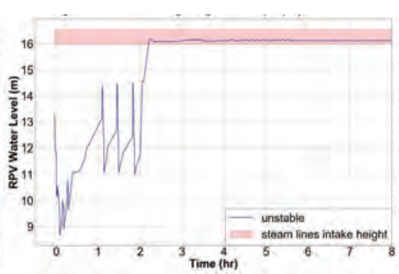
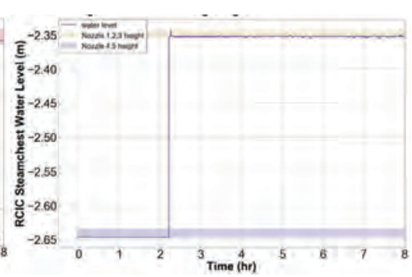
MELCOR RCIC System Modeling & Analysis

- Unstable – Oscillatory fluctuations in speed/injection according to steam line flooding
  - Three high nozzles at common elevation
  - Two low nozzles at common elevation
  - Same nozzle characteristics otherwise
  - Preferential phasic flow as low nozzles flow water only, high nozzles flow two-phase mixture
  - Mixed-phase nozzle void fraction oscillations track with injection oscillations







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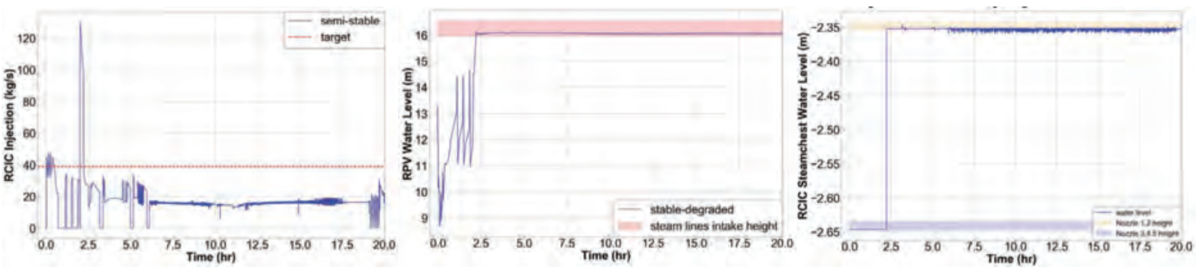
MELCOR RCIC System Modeling & Analysis

- Semi-stable, degraded – Stable, degraded with significant speed/injection fluctuations
  - Two high nozzles at common elevation and three low nozzles at common elevation
  - Fewer mixed-phase high nozzles results in a mitigation of oscillatory behavior
  - Extra pool phase low nozzle results in less turbine impulse and more windage loss



13

MELCOR RCIC System Modeling & Analysis

- Summary
  - FY21 MODSIM work consisted mostly of input development on the ZS-1 and generic BWR input decks
  - ZS-1 loss data and experimental results allows calibration of c_{torque} in MELCOR
  - Demonstrated MELCOR/DAKOTA coupling as an alternative pathway to deterministic calibration
  - GS-2 input model with ZS-1 loss coefficients leads to relatively poor MELCOR/experiment agreement, suggesting an item of future work (characterize GS-2 losses experimentally, repeat c_{torque} calibration)
  - Generic BWR input model can predict Terry turbine self-regulation in one of three modes depending largely on nozzle modeling decisions, e.g. the circumferential orientation of nozzles about the rotor

14

C.4. U.S. Topic Area Presentations

C.4.1. 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 28-30, 2021

Topics

- Key questions
- Current status – November 2021
- Observations from 2021 activities

2

Key Questions

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

3

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

ANL-20/79

March 2021

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

Area	1F1	1F2	1F3
X-100B PCV penetration[b]	Possible molten insulating material [34] No damage observed on outside [35]	NA	NA
X-51 PCV penetration[c]	NA	No damage observed; pressurized water could not penetrate blockage in standby liquid cooling system line [36, 37]	NA
X-53 HPCI steam supply penetration (1F2/1F3)[d]	High dose rate measured [38]	No damage observed [39]	No damage observed [40]
X-6 PCV penetration (CRD hatch)	NA	Melted material [41, 42]	No damage observed from inside [43]
Equipment hatch	NA	NA	Water puddle [44, 45] unknown source
Personnel hatch and nearby penetrations	No damage observed [46]	NA	NA
HPCI pipe penetration[e]	No damage observed, but high dose rates measured; traces of flow and white sediment observed [38, 46,47]	NA	NA
TIP room	No leakage observed from PCV through TIP guide penetrations. Relatively high dose rates measured near other primary system instrumentation penetrations (X-31, X-32, X-33) [38,48]	Dose surveys do not indicate leakage from PCV through TIP guides. High dose levels in samples of materials from TIP indexer [49]	NA
WW vacuum breaker line	Leakage on expansion joint of one line (X-5E) [50]	NA	NA
DW/WW vent bellows	Water leakage attributed to vacuum line above [50]	No leakage observed [51]	
DW sand cushion drain pipe	Leakage [52]	No leakage observed [51]	NA
SC water level	Almost full [24]	Middle [24]	Full [24]
DW Water level	~2 m[24]	~0.2 m[24]	~6 m[24]
Torus room	Partially flooded [53, 54] Rusted handrails/equipment [34] NA	Partially flooded [55] Non-rusted handrails/ equipment [34,56] Some room penetrations tested, no leakage observed [58]	Partially flooded [55] Non-rusted handrails/ equipment [34,57] NA
MSIV room	Limited view obtained [59]	Water leakage cannot be observed [60]	Leakage in Line D near bellows [61]

4

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

Area	1F1	1F2	1F3
DW shield plugs	Reactor well shield plug displaced [62]	Possible leakage [63]	Leakage likely due to radiation measurements at head and presence of H_2 burn [23,64]
DW head/flange	No obvious PCV flange deformations observed; but elastic stretching of bolts during event possible [Appendix C.2.3.2 of FY20 report]	NA	NA
RCIC or other low SC piping	NA	Suspected leak location, not confirmed [34]	NA
RPV upper head	NA	NA	NA
RPV lower head	Ex-vessel debris images, dose surveys, and sample examinations indicate failure [24,65,66]	Ex-vessel debris and images confirm failure [64]	Ex-vessel debris images confirm failure [64]
SGTS vent path	High dose levels in vent path confirms rupture disk (RD) operation [67]	High dose levels in vent path, without RD disk operation, indicates backflow from 1F1 vent piping into 1F2 vent piping [67]	Elevated dose levels downstream of rupture disk confirms operation of RD; HEPA filter dose levels confirms backflow from 1F3 SGTS piping into 1F4 SGTS piping [67]

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

5

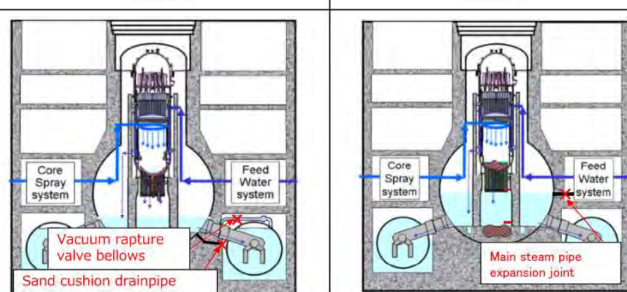
Unit 1 and 3 Water Level Following February 13 Earthquake

3. Presumed cause of the PCV water level drops

TEPCO

- We have found leaks at the following locations at Units 1 and 3

	Unit 1	Unit 3
Concept diagram		



The Unit 3 PCV water level drop has stabilized around the main steam pipe expansion joints. At Unit 1, water level is slowly dropping around the vacuum rupture valve bellows, and the PCV water level drop continues.

Whereas we cannot deny the possibility that a new leak was caused below the PCV water level, it is more probable that the earthquake caused changes to the status of existing leaks.

- Going forward we aim to obtain more information by monitoring changes in parameters, such as water level, through cooling water injection shut off tests.

8

Source: TEPCO Holdings
https://www.tepco.co.jp/en/hd/decommission/information/newsrelease/reference/pdf/2021/reference_20210325_01-e.pdf

6

Unit 1 and 3 Water Level Following February 13 Earthquake

- As a result, increased frequency for water level monitoring implemented
  - Checking plant parameters
  - Video monitoring
  - Water sampling
- Check on various parameters (e.g., temperatures) shows no significant changes

7

Unit 1 and 3 Water Level Following February 13 Earthquake

5 . Current countermeasures

TEPCO

	Current countermeasures
Common to Units 1~3	■ Units 1~3 parameters are stable, and the PCV water levels in Units 1 and 3 are also stable. However, just to be safe, for the time being (until the end of March) enhanced monitoring will continue and plant parameters will be assessed.
Unit 1	The following will be implemented in order to stably monitor PCV water levels and confirm that water levels can be controlled: ■ For the time being the amount of cooling water injected shall be increased as necessary since the water level has fallen below L2, and we shall monitor PCV water level changes between L2 and T2. Furthermore, we shall examine how to address future drops in PCV water level. ■ We shall deliberate methods for continuously monitoring water levels (additional installation of pressure gauge on S/C nitrogen injection line)
Unit 3	■ More data will be obtained by forcing changes in PCV water level through cooling water injection shutoff tests

15

Source: TEPCO Holdings
https://www.tepco.co.jp/en/hd/decommission/information/newsrelease/reference/pdf/2021/reference_20210325_01-e.pdf

8

Unit 1 and 3 Water Level Following February 13 Earthquake

【Reference】 Initiatives to address the PCV water level drops (cooling water injection shut off test)

TEPCO

- Cooling water injection shut off test: To be implemented in the future in order to control the amount of accumulated water. In light of the recent water level drops, the plans for cooling water injection shut off tests at Units 1 and 3 are as shown in the chart below.

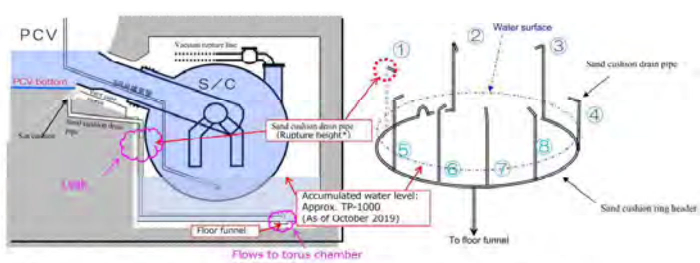
	Unit 3	Unit 1
Prior objective (test period)	To confirm whether water level falls to the lowermost point of the main steam pipe expansion joint (seven days)	To check whether or not water levels fall below T1 at the lowermost part of the PCV seen during cooling water injection shut off tests implemented in November (five days)
Current conditions	Water level has dropped and is remaining stable around the main steam pipe expansion joint	Water level drop trends are being monitored to assess whether or not water level has fallen below T2
Objectives in light of the PCV water level drops	In addition to the previous objectives, we will check to see if water levels stabilizes even if it falls below the lowermost point • We will learn if there are large leaks lower than the main steam pipe expansion joint • This may lead to cooling water injection reductions or even longer cooling water injection shut off tests • If the water level drops further, we will know that there is a leak below the main steam pipe penetration.	Test priority is low • The previous test showed that water level drops to around T1 • Even longer shut off tests will be considered in light of the increase in water level trend data
Other	• The tests are being planned for April • In conjunction with the tests, we are examining the possibility of using a camera to check the conditions inside the MSIV room	• Water level may have to be maintained at a certain height in order to conduct the internal investigation using an ROV

18

Source: TEPCO Holdings
https://www.tepco.co.jp/en/hd/decommission/information/newsrelease/reference/pdf/2021/reference_20210325_01-e.pdf 9

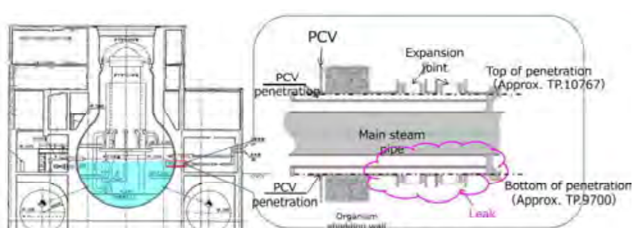
Existing Unit 1 and 3 PCV Leaks

- Unit 1 – Vacuum rupture line bellows



* There are eight sand cushion drainpipes, and one of them has a rupture that is exposed to the air

- Unit 3 – Main steam pipe penetration



Cross-section of Unit 3 reactor building

Enlarged view of main steam pipe penetration

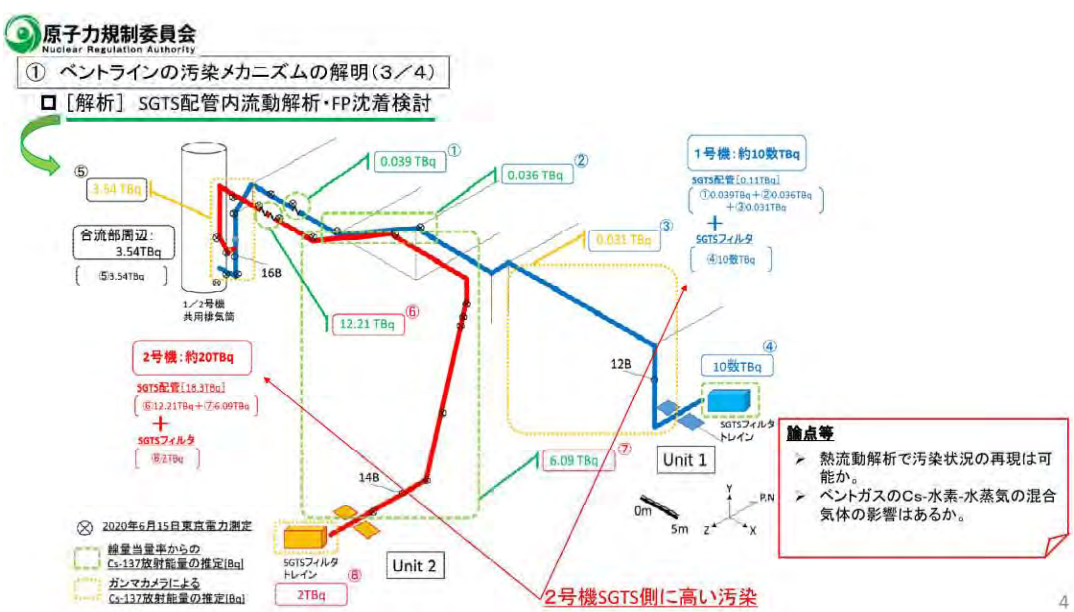
Source: TEPCO Holdings
https://www.tepco.co.jp/en/hd/decommission/information/newsrelease/reference/pdf/2021/reference_20210325_01-e.pdf 10

Unit 1 and 2 Vent Line Investigation

- Unit 1 and 2 vent lines had high degree of fission product deposition
- Deposition higher in Unit 2 (not vented)
- Unit 1 and 2 stack interface shows high degree of deposition

11

Unit 1 and 2 Vent Line Investigation



Source: Nuclear Regulation Authority, Japan
https://www.nsr.go.jp/data/000352410.pdf

12

Insights

- Increased containment water leakage from Feb 2021 earthquake
  - Injection testing can help provide additional details
- Fission product study of vent paths provides insights on backflow
- From 2020 forensics meeting
  - Increased containment pressure impact on SRV opening accounted for in typical Level 1&2 PRA
    - Valves can reclose on elevated containment pressure
    - Input into BWR Primary Containment Pressure Limit
  - Influence of reduced accumulator pressure on intermediate opening of SRV with possible area reduction, not typically addressed in PRA.
- Continued focus on instrumentation readings impacted by plant conditions in simulation codes

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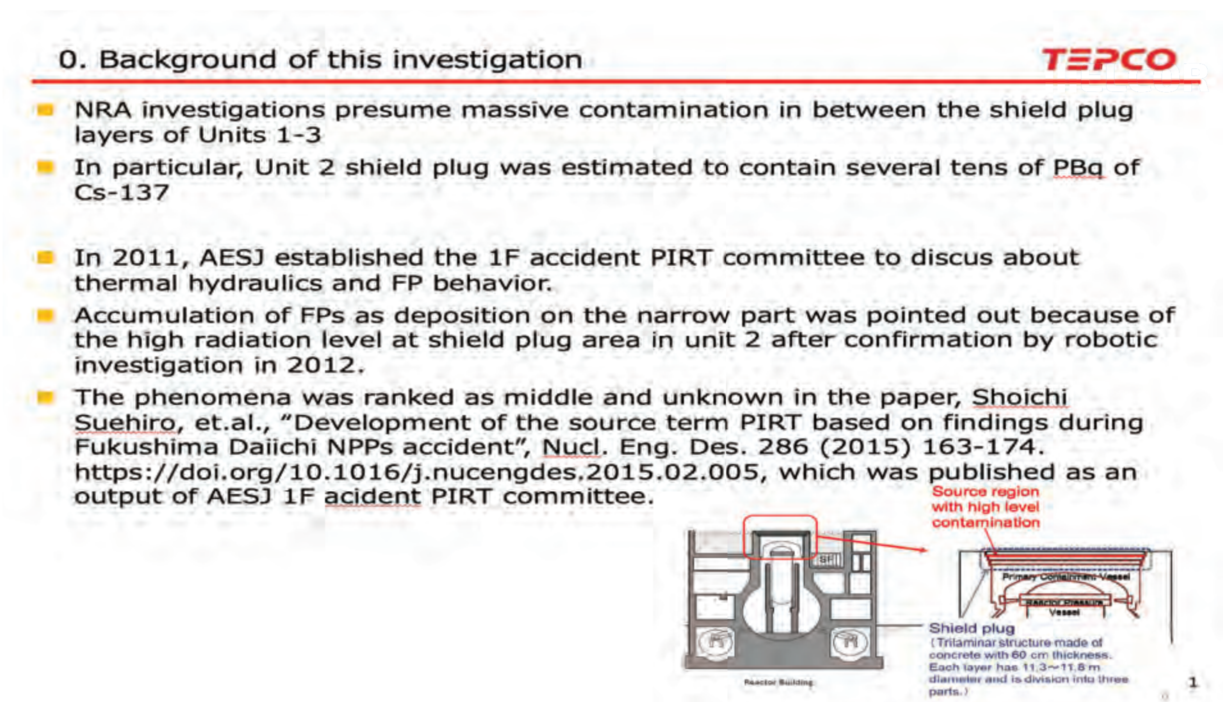
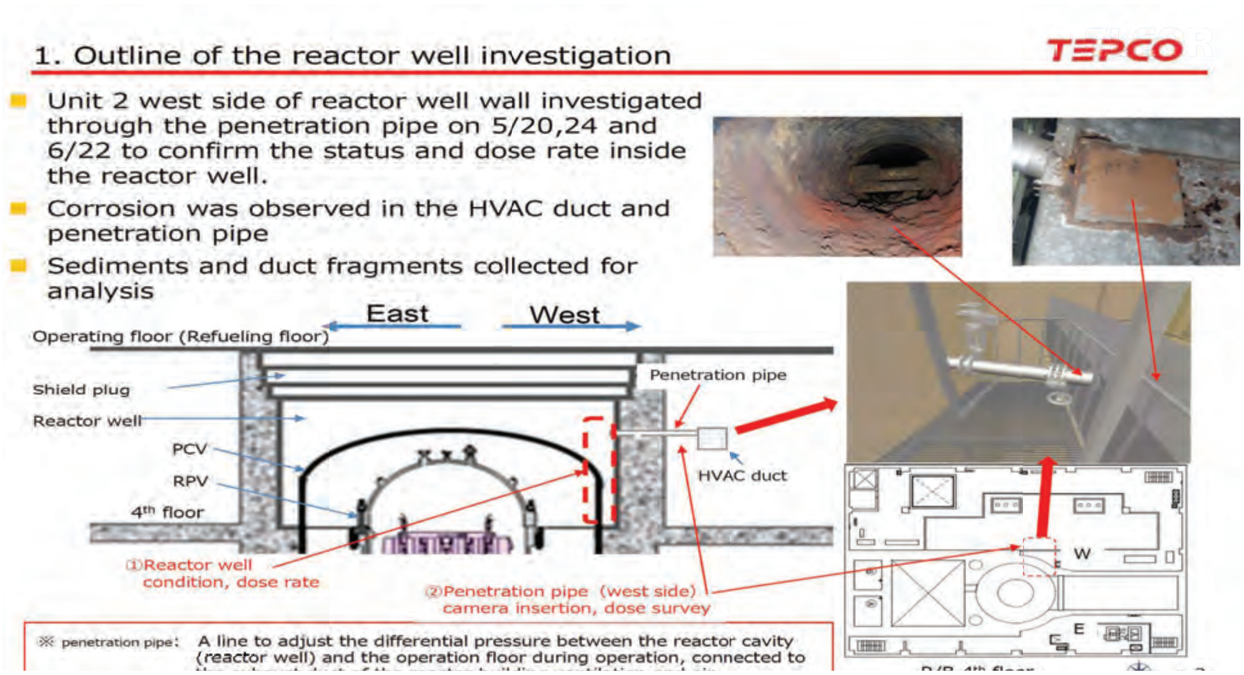
C.4.2. Topic Area 2- Radiation Surveys and Sampling





Study Overview

IIIELCOR

- Guiding Questions:
  - Is water injection ever an unsuitable course of action during a severe accident?
  - What is the impact of water injection on containment and environment source terms?
  - How do the timing of water injection, the quantity of water injected, and the failure pressure of containment interact and affect severe accident evolution?
- Study Parameters
  - Sample size: 600 simulations
  - Simulation time: 72 hours
  - Fukushima-like BWR Mk-I containment SBO

Uncertainty	Range	Distribution	Comment
Injection Onset Time [h]	0.0 – 24.0	Uniform	–
Normalized Injection Multiplier	0.0 – 1.0	Uniform	Injection consists of 283.0 K water source, injected at a rate proportional to the decay heat (e.g., a multiplier of 1.0 injects quantity of 283.0 K water such that the heat of vaporization of the injected water will equal the decay heat).
Containment Leakage Onset Pressure [MPa]	0.5 – 0.9	Uniform	–

*Vessel depressurization only occurs by lower head failure in this study

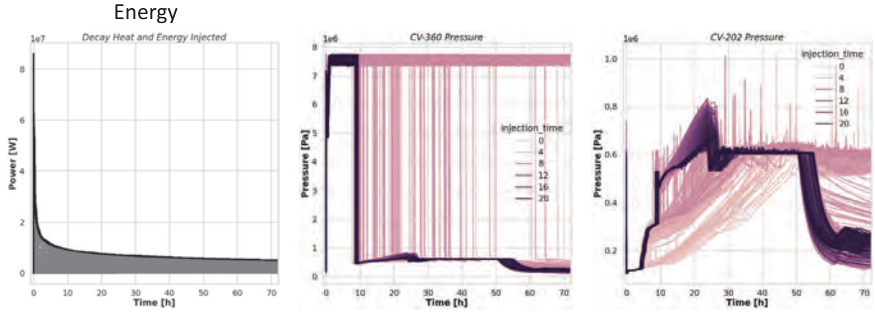
4

Key Observations

- Lower head failure is prevented for
  - Cases exhibiting moderate to high injection rates prior to significant core degradation and
  - High injection rates after the onset of core degradation
- Containment failure is delayed when water injection begins prior to significant core degradation
  - The earliest observed containment failure was at 8.5 hours
  - The latest observed containment failure was at 52 hours
- Early containment failure is observed to occur at the time of lower head failure when the containment leak onset pressure is between ~0.5-0.6 MPa.
  - Failure timings are concentrated near 8.5
- Injection reduces source terms for key radionuclide groups.
  - Early water injection can reduce both containment and environmental source terms
  - Late water injection can reduce environmental source terms
  - Environmental source terms are more sensitive to water injection

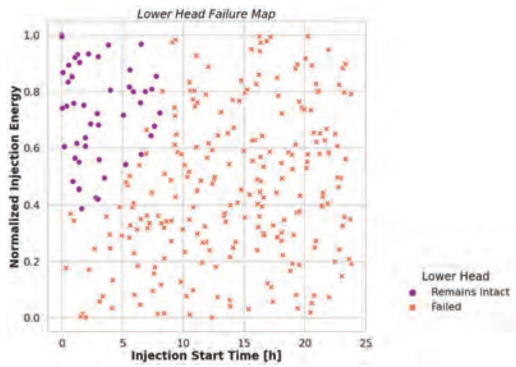
A subset of simulations exhibiting water injection prior to and during lower head failure also exhibit uncharacteristically large source terms. This small set of simulations deviate from the general observations of this study and occur only by coordination of multiple factors including water injection timing, water injection quantity, state of the reactor core, state of containment, quantity of readily transportable radionuclides, etc.

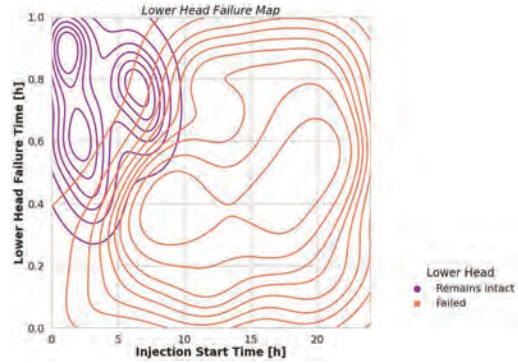
Decay Heat and Injection Energy | Vessel Pressure | Containment Pressure



Lower Head Failure Mapping

MELCOR





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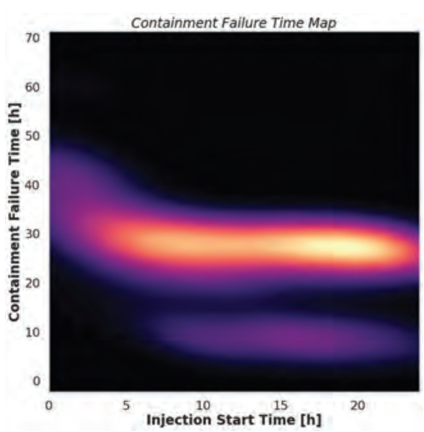
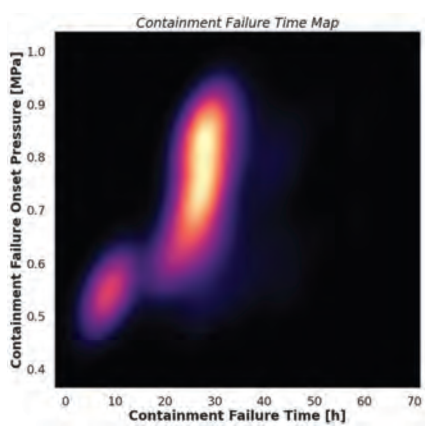
Containment Failure Mapping

MELCOR





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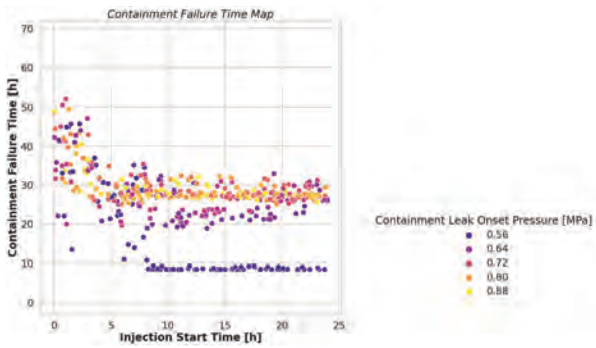
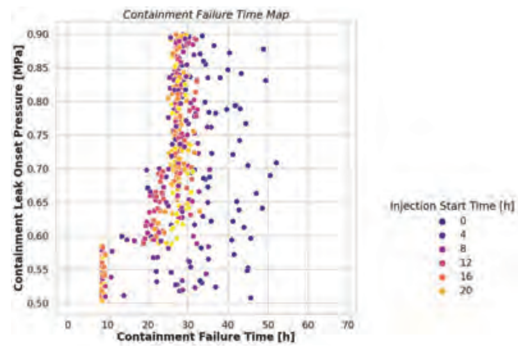
Containment Failure Mapping

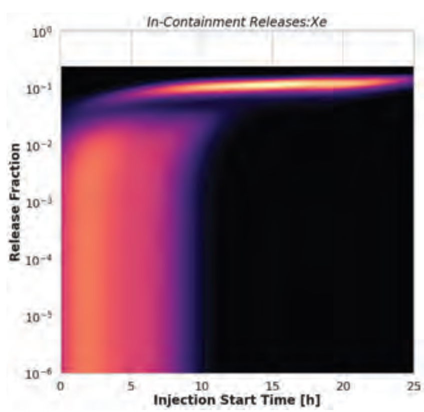
MELCOR

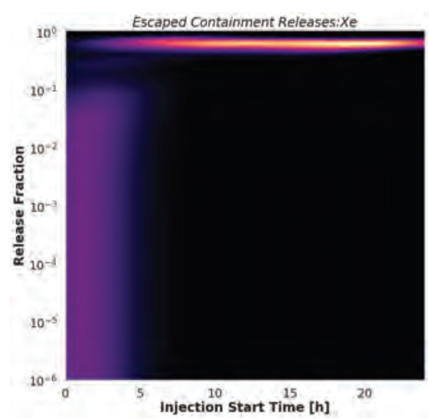




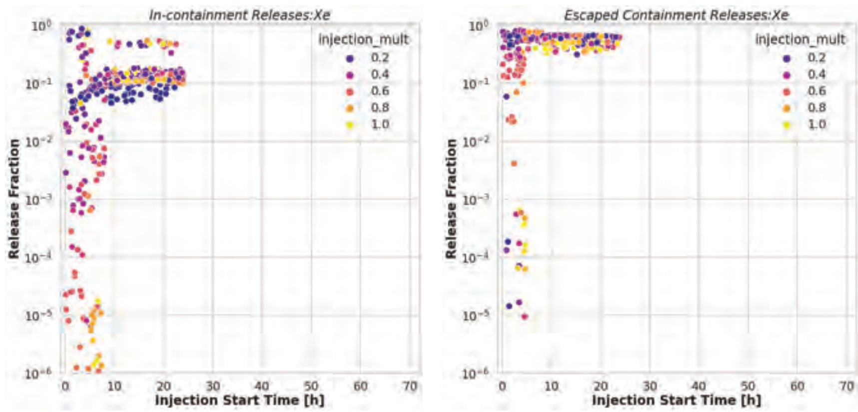
Noble Gases Released After 72 Hours

MELCOR



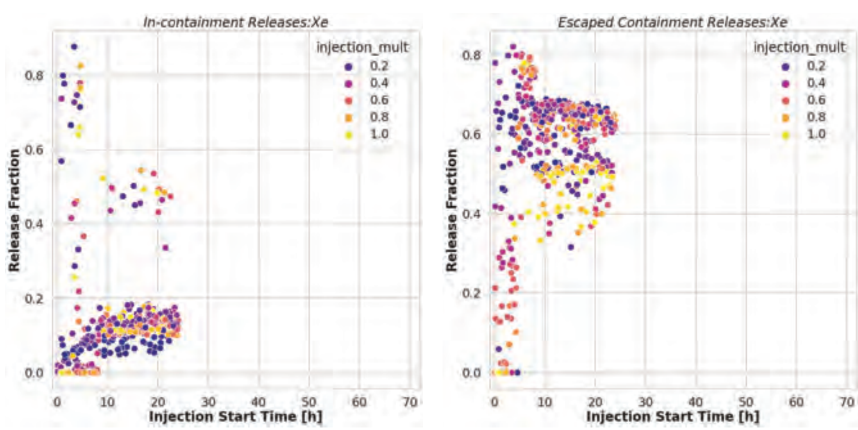


Noble Gases Released After 72 Hours



11

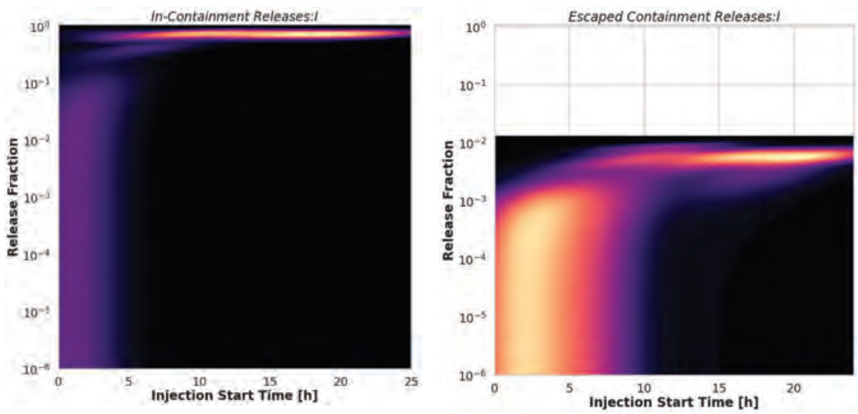
Noble Gases Released After 72 Hours



12

Halogens Released After 72 Hours

MELCOR



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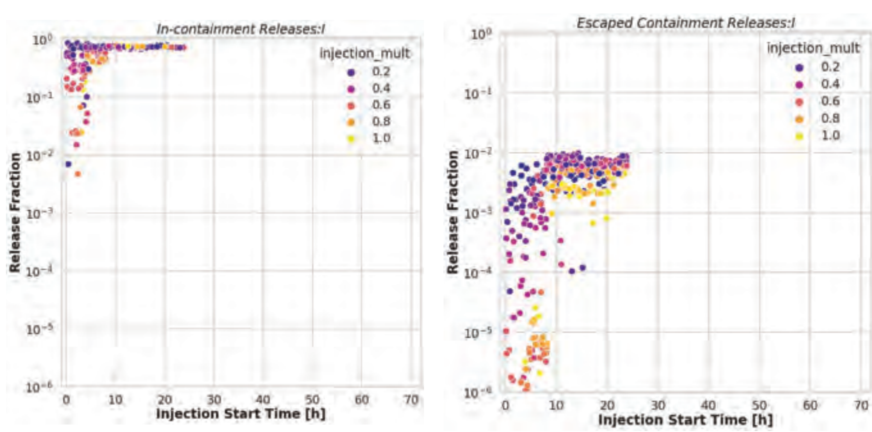
Halogens Released After 72 Hours

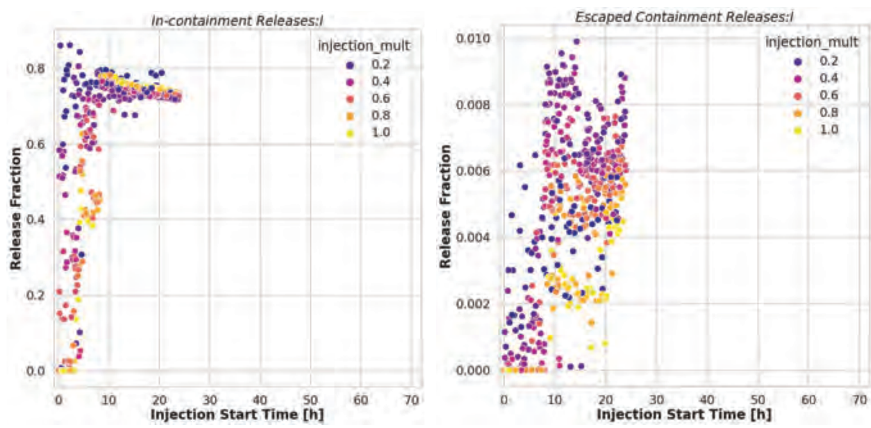
MELCOR



14

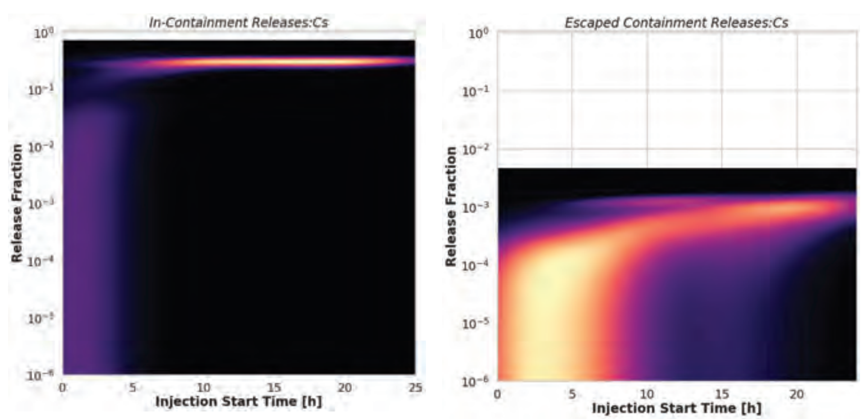
Halogens Released After 72 Hours

MELCOR

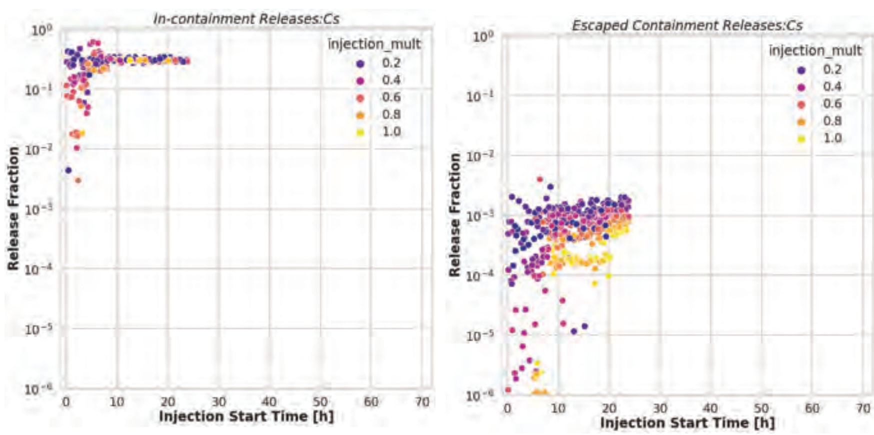


Alkali Metals Released After 72 Hours

MELCOR

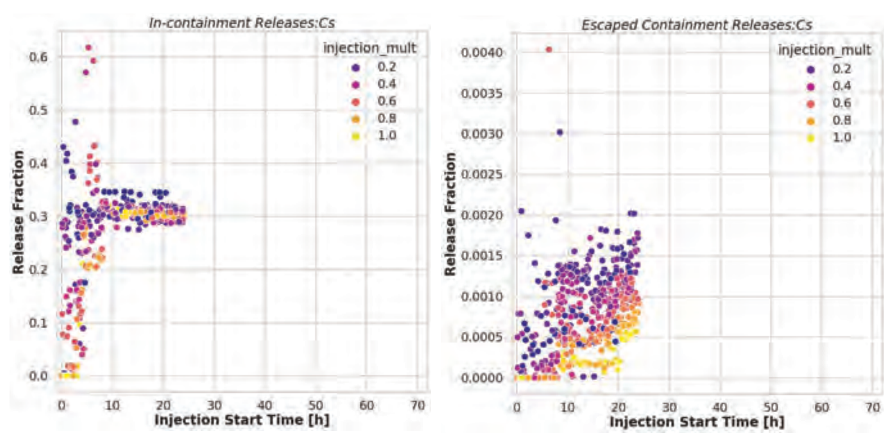


Alkali Metals Released After 72 Hours



17

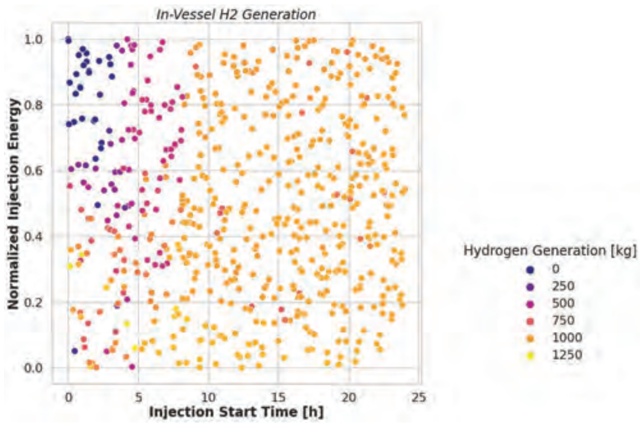
Alkali Metals Released After 72 Hours

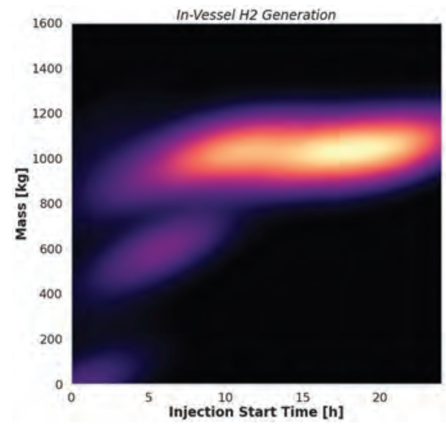


18

In-vessel Hydrogen Generation

MELCOR





C.4.3. Topic Area 3- Debris Endstate

C.4.3.1. Recent Insights regarding Debris Endstate and Coolability

WE START WITH YES.

TOPIC 3 - CORE DEBRIS LOCATION EVALUATIONS

MITCH FARMER
Nuclear Science & Engineering Division
Argonne National Laboratory

Fukushima Forensics Meeting, November 28-30, 2021
Hybrid Virtual (Webex) and In-Person at NEI, 1201 F. Street, NW, Suite 1100, Washington, DC

PRESENTATION OUTLINE

Comments/observations/suggestions on new information gained in the following areas:

- Debris distribution/state within X-6 penetration in 1F2
- New and planned chemical analysis results for 1F1-1F3
- Steps towards reducing water levels in PCV's to mitigate seismic concerns.

Miscellaneous:

- Potential *in-situ* core-debris water ingression measurement for 1F2.

Note: This presentation is based almost exclusively on information gained by using Google Translator for Japanese → English translation from TEPCO/IRID/NRA presentations, so please let me know if something is misinterpreted.

DEBRIS DISTRIBUTION/STATE WITHIN X-6 PENETRATION IN 1F2*

- TEPCO plans to remove deposits/cabling from X-6 penetration in order to insert arm-type equipment into the PCV for internal investigations.
- Activities to characterize materials within the penetration include:
  - Video of deposits (originally obtained in January 2017).
  - Recent sediment contact survey and 3-D scan results (discussed herein).
    - Obtained via an insertable guide pipe with built-in survey equipment.
- Ultimate plan is to clear out this penetration using low/high-pressure water jets and a pusher by relocating materials within the penetration into the PCV.

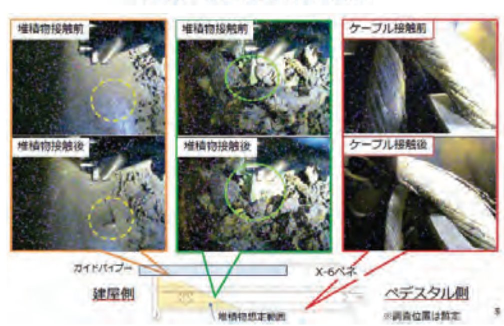
*Information in this section based on information released by TEPCO Holdings (https://www.tepco.co.jp/decommission/information/committee/roadmap_progress/pdf/2020/d201126_08-j.pdf)

Argonne

VIDEO AND CONTACT SURVEY RESULTS

Video Data

Contact Data

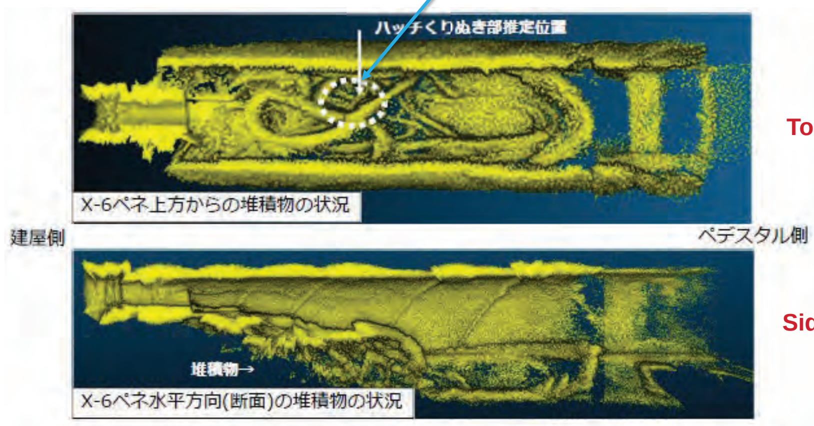


Images courtesy TEPCO Holdings

Argonne

3-D SCAN RESULTS FROM FOR X-6 PENETRATION

Estimated position of 'hatch hollow'



Top View

Side View

Reactor Building

PCV

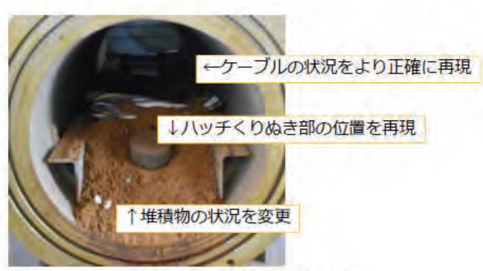
Images courtesy TEPCO Holdings

5

OBSERVATIONS FROM X-6 INVESTIGATIONS

- The sediment height tapers downwards starting from the reactor building side towards the PCV.
- The sediment as well as cabling are loose and move upon contact.
- Confirmed that the X-6 penetration hatch was left in the middle of the penetration.
- The sediment near the center of the penetration is sandy in nature.
- Results will be used to build a mock-up to test equipment for clearing the penetration to facilitate internal investigations.

Mock-up for testing of removal equipment – to be refined



現在のモックアップ設備と改良を検討するポイント

Image courtesy: TEPCO Holdings

6

DISCUSSION ON FINDINGS FROM X-6 PENETRATION INVESTIGATIONS (1/2)

- **No. 1**: There seems to be *a lot* of granular sediment in the penetration.
- What are potential sources for this sediment?
  1. Aerosols from burning of paint/cabling within the PCV as a result of RPV failure and melt relocation into the PCV.
  2. Aerosols from MCCI.
  3. ??
- The amount of sediment seems to be inconsistent with aerosol produced from #1 given the large expanse of PCV (personal opinion).
- Results from reactor material tests with siliceous concrete (e.g., ACE/MCCI tests) indicate a prodigious amount of silica aerosol production during MCCI.
- Chemical analysis of this sediment would provide an indication of the likely source; i.e., large amounts of concrete oxides (SiO_2, Al_2O_3, MgO, CaO) would indicate occurrence of dry MCCI.
  - *Why is this important*? Whether or not the core debris contains concrete slag (particularly that from siliceous concrete) has a strong impact on the mechanical strength/toughness of core debris, which will impact D&D.
  - *Is MCCI a plausible scenario for 1F2 based on other data and code insights*?

DISCUSSION ON FINDINGS FROM X-6 PENETRATION INVESTIGATIONS (2/2)

- **No. 2**: The fact that the sediment height tapers gradually down starting from the reactor building sided to the PCV is intriguing.
- What is this telling us? Possibilities (among others) are suggested below:
- Due to over-pressurization of the containment, this penetration had a large leak rate during the accident.
  - i.e., aerosol-laden gas/steam from the PCV flowed into the penetration, and when encountering the hatch, the gases passed through allowing the aerosols to settle.
  - Due to a likely temperature gradient in the PCV shielding concrete, aerosol-laden gas from the PCV flowed into the penetration towards the top of the hatch, then turned 180°and flowed back out to the containment.
    - During the turn, aerosols would have been deposited near the hatch.
- *Why is this important?* Understanding the potential for PCV penetrations to leak during an accident is important in source term evaluations.

NEW AND PLANNED CHEMICAL ANALYSIS RESULTS FOR 1F1-1F3**

- JAEA has conducted new analyses on previously samples obtained using a modern array of characterization equipment that includes: i) Image Plate (IP), Scanning Electron Microscope (SEM), Energy Dispersive X-ray Spectroscopy (EDS), Wavelength Dispersive X-ray Spectroscopy (WDS), Transmission Electron Microscope (TEM), Inductively Coupled Plasma Mass Spectrometry (ICP/MS), and Focused Ion Beam (FIB)

Summary of samples analyzed

#	Description	Date obtained
1	1F2 and 1F3 Torus room water filters (0.1 μm Filter)	5/2019
2	1F3 PCV (from survey wipe of internal inspection device)	8/2017
3	1F1 X-2 penetration (survey wipe of internal inspection device)	6/2019
4	1F1 operating floor survey smear from well plug	7/2019
5	1F2 survey smear from plastic sheeting laid on operating floor	3/2014

**Information in this section based on information released by TEPCO Holdings (https://www.tepco.co.jp/decommission/information/committee/roadmap_progress/pdf/2020/d201126_08-j.pdf)

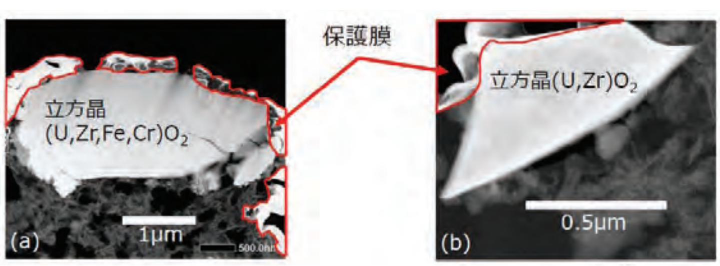
9

SAMPLE # 1 RESULTS

1F2-3 torus room water samples

- > 90 % of α nuclides removed by 0.1 μm filter
  - 1F2 samples contained U particles of 3 to 5 μm size. Most of the α nuclei in the retained water are considered to be stable oxides
  - A small amount of U present as 10 μm particles found for 1F3 sample.
- Confirm that most of U is present in grains in all samples.
- Most α nuclei in the retained water are present as stable oxides.
- *Observations*: i) particle sizes (possibly valuable for source term assessments), ii) no Zr found (fully oxidized), and iii) samples contain elements characteristic of an in-vessel melt composition.

TEM/EDS results for 1F2 torus water sample



Images courtesy TEPCO Holdings

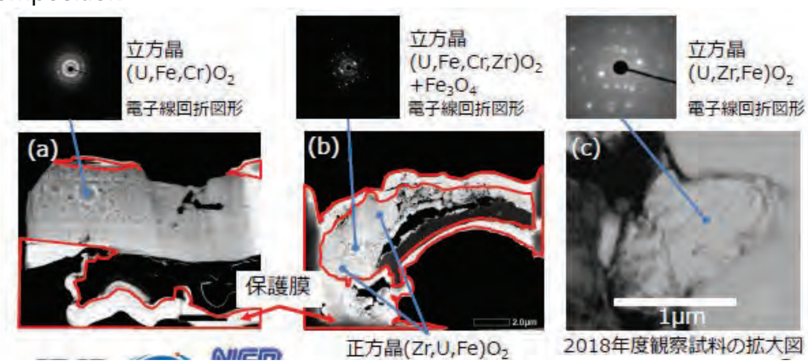
SAMPLE # 2 RESULTS

1F3 PCV

Depending upon where you look:

a) $(U,Fe,Cr)O_2$ grains with no Zr;
b) $(Zr,U,Fe)O_2$ grains and $(U,Fe,Cr,Zr)O_2$ nano-grains are aggregated, which are formed by melt solidification process or a combined process
c) High U region composed of fine crystals present as chemically stable oxide.

- *Observation*: i) samples contain elements characteristic of an in-vessel melt composition.



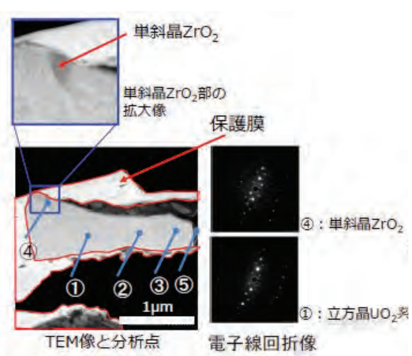
Images courtesy TEPCO Holdings

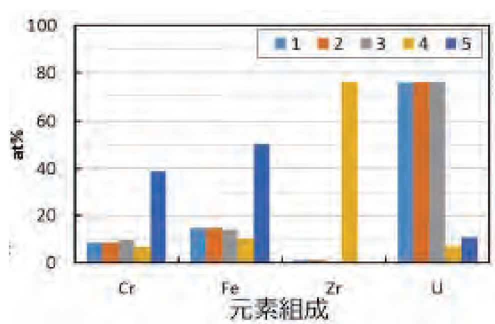
11

SAMPLE # 3 RESULTS

1F1 X-2 Penetration

- Main phase (① ~ ③) is $(U,Fe,Cr)O_2$
- The presence of a monoclinic ZrO_2 (④) indicates that the cooling rate is sufficiently slow for phase separation to occur.
- Insight: probably formed by the melt-solidification process.
- *Observations*: i) sample contains elements characteristic of an in-vessel melt composition, and ii) formation by melt solidification process is consistent with accident interpretation.





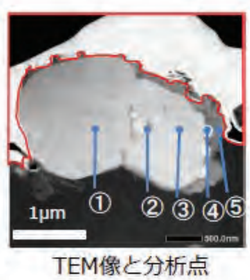
Images courtesy TEPCO Holdings

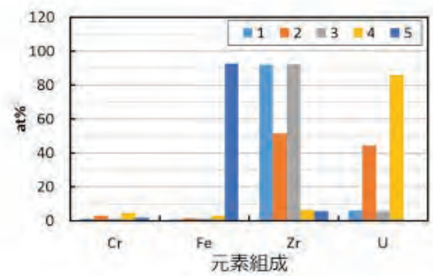
12

SAMPLE # 4 RESULTS (1/2)

1F1 Operating floor survey smear from well plug

- Main phase is aggregated/fused monoclinic ZrO_2 granules
  - Similar to the 1F1 X-2 penetration sample; indicative of a slow cooling rate
  - Presumed to be an agglomeration of multiple particles formed by the fusion and solidification process.
- *Observations*: i) sample contains elements characteristic of an in-vessel melt composition, and ii) formation by slow solidification process is consistent with accident interpretation (i.e., dry scenario).





Images courtesy TEPCO Holdings

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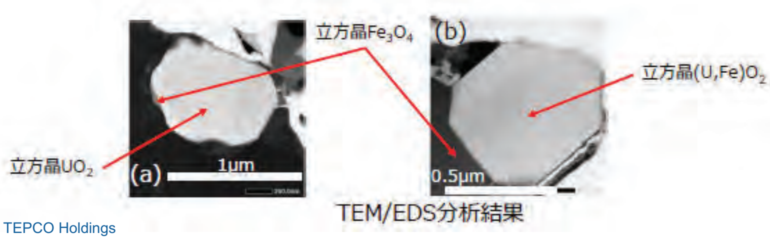
SAMPLE # 4 RESULTS (2/2)

1F1 Operating floor survey smear from well plug

- a) Fe-UO_2 grains containing almost no Zr formed in the process of evaporation and condensation
  - Fe + Cr concentration is ~ 6 at%
- b) ~20 at% Fe (no Cr or Zr) (U,Fe)O_2 granules (with Fe_3O_4 surrounding) likely formed by cooling and solidification of fuel in contact with Fe
  - Lack of Zr could indicate that U-Fe-O particles formed by evaporation - condensation process, or by direct reaction between UO_2 and Fe.
  - Structure consistent with slow cooling and solidification process.
- *Observations*: i) sample contains elements characteristic of an in-vessel melt composition, and ii) formation by evaporation-condensation process is indicative of high melt temperature conditions.



Images courtesy TEPCO Holdings

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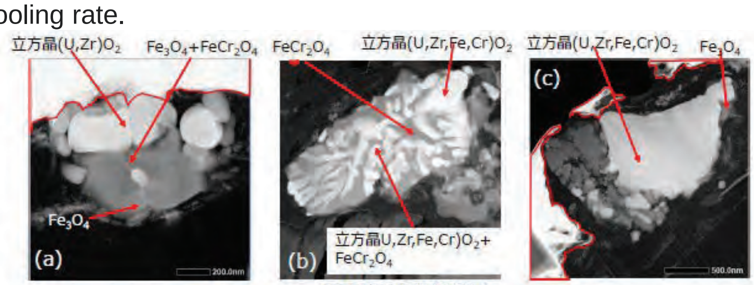
SAMPLE # 5 RESULTS

1F2 survey smear from operating floor

- a) Multiple $(U,Zr)O_2$ crystals containing 6-7 at% Fe+Cr present as Fe_3O_4 - $FeCr_2O_4$ granules. Precipitates aggregated and fused during solidification.
- b) Granules composed of mixed phases of $(U,Zr,Fe,Cr)O_2$, $FeCr_2O_4$, and a mixture of $(U,Zr,Fe,Cr)O_2$ and $FeCr_2O_4$. Size of precipitates may be useful in estimating cooling rate.
- c) ~10 at% Fe in the form of Fe_3O_4 formed during solidification at the periphery of $(U,Zr,Fe,Cr)O_2$ phase.
- *Observations*: i) sample contains elements characteristic of an in-vessel melt composition, and ii) small granular size observed in b) is likely indicative of a rapid cooling rate.



Images courtesy
TEPCO Holdings

CHEMICAL ANALYSIS RESULTS - SUMMARY

1. Based on analysis of water samples from 1F2-3 torus chambers, 99.6% and 92.5% of α-nuclides, respectively, were removed by 0.1 μm filter and present as UO_2 and $(U,Zr,Fe,Cr)O_2$.
2. Based on 1F1 samples (floor well plug and X-2 penetration):
   – Grain structures indicate that aerosols formed via evaporation-condensation and melt solidification processes.
   – Crystals thought to have precipitated from a U-Fe-O-based melt were also detected.
   – U-containing monoclinic ZrO_2 detected, which is indicative of a slow cooling process.
3. Analysis of 1F2 sample from refueling floor indicates that U-containing granules formed by the melt-solidification process.
   – U-Zr-Fe-Cr-O-based melt is thought to have phase-separated during the cooling process, and the composition and structure of the granules may be useful for estimating the cooling rate of the granules.

PERSONAL OBSERVATIONS/QUESTIONS (1/2)

- Overall, these more detailed analyses are highly beneficial in supporting interpretations of the accident sequences.
  - *Caveat*: only a few samples from a limited number of locations have been analyzed to date!
- However, *based on what has been reported to date*, a few points to consider are as follows:
  - The identification of evaporation-condensation as a likely mechanism for aerosol formation in 1F1 is indicative of very high temperatures experienced in the core debris during the accident.
    - Is there a threshold temperature for this mechanism to become active?
- The analyses documented to date seem to only identify principal elements of in-vessel melt compositions (e.g., U, Zr, Cr, Fe + O).
  - Question: Are typical concrete elements (i.e., Si, Al, Mg, Ca) not detected or measured for during the analyses?

17

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PERSONAL OBSERVATIONS/QUESTIONS (2/2)

- MCCI may have occurred in all 3 units
  - Felt to be important from the viewpoint of D&D as the presence of concrete slag in core debris has a strong impact on material properties
    - Based on practical experience during PTE on MCCI tests, this means fracture toughness.
- If concrete erosion occurred, then concrete elements in corium samples can readily be detected using ICP/MS
  - If acid-formed solutions for ICP/MS are still available, it would be useful to re-run the samples specifically looking for concrete elements.
  - If smear pads contain material bearing materials similar to concrete oxides (e.g., silica fiber), then measurement may be skewed. Is this important and has it been factored into the analysis?
- Experience at Argonne indicates that the best solution process for corium is to use a Parr Acid Digestion Bomb with HCl and HNO_3 acids to dissolve samples.
  - A small fraction (few wt%) of samples containing chromium never dissolves.
  - This material is analyzed separately by SEM/EDS and added in with results from ICP/MS to complete the analysis.

18

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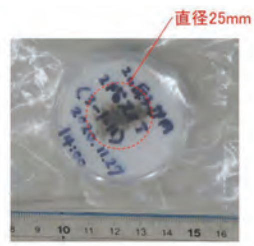
COMMENTS ON ADDITIONAL PLANNED ANALYSIS OF SAMPLES FROM 1F1-2 AND 1F3***

- To obtain information on migration route, chemical form, and composition of released radionuclides, smear samples obtained from surfaces within the reactor buildings, SGTS filters, and SGTS piping will be analyzed.
  - 1 smear obtained from 1F1-2 SGTS piping and 23 obtained from 1F3 SGTS.
- Nuclides to be measured include Cs-134/137, Sr-90, Tc (Mo)-99, I-129, and α nuclides (Th, U, Pu, Am), etc.
- If MCCI occurred, refractory substances such as Sr and α-nuclides may have been released as aerosols.
  - This is one metric to be used in the analysis to determine if MCCI occurred.
- If ICP/MS is to be carried out on samples, then it is recommended to also analyze for elements of concrete oxides (e.g., Si, Ca, Mg, Al) as an additional method for determining if MCCI occurred.
  - A modest amount of concrete ablation will result in > 10 wt% concrete oxides in the samples which is easily detectable.



***Information and image in this section based on information released from the Nuclear Regulation Authority (https://www.nsr.go.jp/data/000352408.pdf) 19

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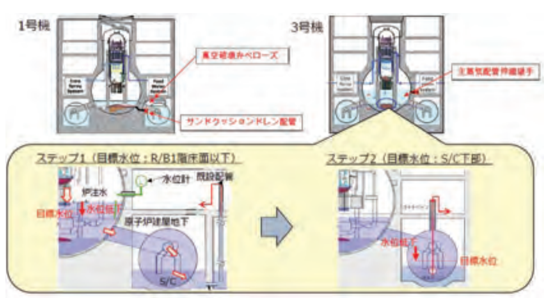
STEPS TOWARDS REDUCING PCV WATER LEVELS IN 1F1 AND 1F3****

- Reduction of water levels in 1F1 and 1F3 PCVs is currently being considered in order to increase plant resiliance to seismic events.
  - Water level in 1F2 is already low.
- As part of this process, it is important to ensure that the debris is adequately cooled and radionuclides are retained within containment when water level is lowered.
- Preparatory steps for 1F1 have recently been reported and are discussed below.



****Information and images in this section based on information released from the Nuclear Regulation Authority (https://www.nsr.go.jp/data/000349465.pdf) 20

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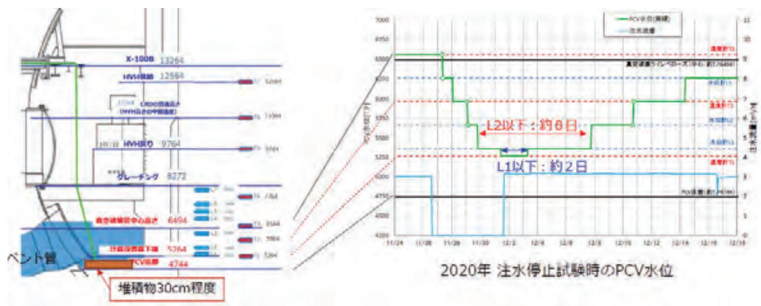
PREVIOUS 1F1 WATER LEVEL TESTING

- In the water injection stop test conducted in 2020, no increase in dust concentration was observed after a period of ~8 days (when water level dropped below ‘L2’ level at ~90 cm above pedestal floor), and after ~2 more days (when water level fell below ‘L1’ level at ~60 cm above pedestal floor).
- Note that debris/sediment depth in the annulus is ~ 30 cm, and the inlet to the downcomers to the torus are at ~ 50 cm.
  – Lowest existing instrument for measuring water level (a TC denoted TC1) is at ~ 50 cm PCV floor.



Source: Nuclear Regulation Authority (https://www.nsr.go.jp/data/000349465.pdf)

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WATER LEVEL REDUCTION/CONTROL IN 1F1: CURRENT STATUS

- Current plan is to maintain water level between L2 (~90 cm) and T2 (at ~120 cm) by monitoring.
  – TEPCO is considering adding a level head (pressure) transducer for continuous level monitoring.
- After confirming that continuous water level monitoring is possible, would change to maintenance between L1 (~60 cm) and L2 (~90 cm).
- When conducting internal surveys using an underwater inspection device, would elevate the water level, and then return to the original level after completion.
- Even if continuous water level monitoring with a pressure gauge is not possible, consider lowering the water level with an existing instruments while confirming that there is no abnormal rise in temperature or dust.
- In the author’s opinion, this is a sound move to keep debris covered by water and adequately cooled while minimizing water depth (and, correspondingly, leakage rates) in the PCV.

22

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ESTIMATES OF 1F1 WATER LEAKAGE RATE

- Leakage rate estimates developed by TEPCO based on water level changes knowing the PCV floor area; results shown in graph.
  - Leakage rates range from 0.5 m^3/ hour (2.2 gpm) at highest water level elevations down to 0.1 m^3/ hour (0.44 gpm) at lowest point (i.e., between T2 and L2).
- Valuable information:
  - e.g., (total) decay heat level in 1F1 debris is currently ~13 kW.
  - Average leak rate between T2 and L2 is ~ ½*(0.3+0.1)~0.2 m^3/hr (~0.9 gpm).
- Thus, if water is injected at average leak rate (to maintain water level constant), then peak (adiabatic) water temperature rise within PCV would be ~55 C due to decay heat.
  - This neglects PCV heat losses to ambient (at this time, relatively large).
  - Approach would: i) keep debris covered with water, ii) minimize leakage, and iii) conservatively maintain acceptable temperature rise in the PCV (IMO).

23

Source: Nuclear Regulation Authority (https://www.nsr.go.jp/data/000349465.pdf)

MISCELLANEOUS: 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 highly 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 can be used to estimate debris permeability and dryout limit for an actual prototypic core debris accumulation, which is invaluable for reactor safety evaluations.

1F2 in-pedestal debris distribution

24

REQUESTS FOR ADDITIONAL INFORMATION

- Various organizations with Japan have there hands full working on D&D at Daiichi, *so there are no additional information requests at this time*.
- Only a few suggestions in this presentation have been made, which are as follows:

1. During chemical analysis, search for concrete oxides in samples from all three units as this will provide information on whether MCCI occurred. This is important for informing D&D operations since the presence of slag in core debris significantly affects mechanical properties.
2. If additional camera entries are made into 1F2 pedestal region, it would be extremely valuable to take an hour of time and increase water flowrate until water pools on the top of the debris and spills over into the drywell. This will provide prototypic data on the permeability and dryout limit for a large accumulation of core debris.

25

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.

26

C.4.3.2. Passive Interim Storage of Fukushima Core Debris

Passive Interim Storage of Fukushima Core Debris

Presented to: Reactor Safety Technology Expert Panel Forensics Meeting
Hosted by the Nuclear Energy Institute, Washington DC
November 29/30, 2021

Dr . Martin G. Plys
Vice President and Chief Technology Officer
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Outline

- FAI Efforts for TEPCO
- Why passive vented interim storage is a far superior option to any others
- How to demonstrate feasibility of passive vented interim storage
- FAI experimental proof that passive vented interim storage can be accomplished
- Data needs to support passive vented interim storage

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FAI Efforts for TEPCO

- FAI is working directly for TEPCO on technical issues for debris retrieval steps
- FAI pertinent experience (experiments and custom modeling):
  - We created most of the technical basis for vacuum drying and sealed interim storage of 2000 tons of highly damaged metallic spent nuclear fuel (Hanford)
  - We created the technical basis for passive vented interim storage of 700 tons of highly damaged metallic spent nuclear fuel (Sellafield)
- We have used our experience to suggest important design features for safe removal of hydrogen and water from Fukushima fuel debris packages
- We have proposed passive vented interim storage for Fukushima fuel debris because this is the preferred modern option – Sellafield vs Hanford experience
- This FY, we have obtained experimental data that proves the feasibility and safety of passive vented interim storage of Fukushima fuel debris

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Superiority of Passive Interim Storage vs Other Options

- Exhaustive studies of options for disposition of Sellafield ponds solids (highly damaged spent nuclear fuel, sludge, ion exchange materials, and mixed beta-gamma waste) has favored passive, vented interim storage over any other options
- Other options have included vacuum drying, encapsulation (grouting), and vitrification
- Passive vented interim storage has also been adopted by Sellafield for wastes retrieved from the Magnox Swarf Storage Silo (MSSS)
- The basic reasons are:
  - Feasibility of passive options
  - Intrinsic safety of passive options
  - Low cost of passive options vs active processing options
  - Relative safety of passive options vs active processing options
- The cost savings is between 1billion and 2 billion GBP – 1 billion GBP for MSSS alone
- These reasons apply to Fukushima core debris equally as well

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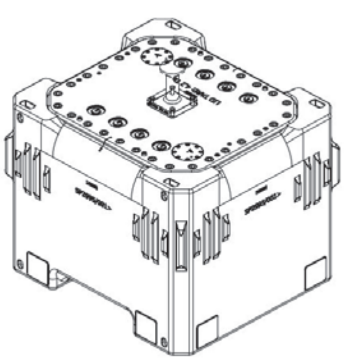
How to Demonstrate Feasibility of Passive Interim Storage

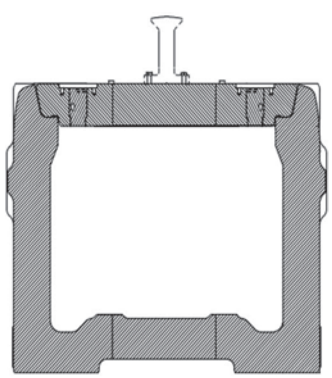
- Key requirements for feasibility of passive interim storage are:
  - Radiological containment and appropriate dose reduction by packaging
  - Passive hydrogen removal – prevent flammable atmospheres in the packages
  - Passive water removal – guarantee eventual package dryness and minimize any potential for package corrosion
  - Passive decay power removal – acceptable package temperature (corrosion, strength) and acceptable debris temperature (minimize activity release)
- FAI has created models for hydrogen removal, water removal, and decay power removal for Sellafield (both pond fuel debris solids and MSSS sludge)
- FAI has experimentally validated the models for hydrogen removal for Sellafield
- The FAI design is now being manufactured: The Self-Shielded Box (SSB) for the First Generation Magnox Storage Pond (FGMSP)

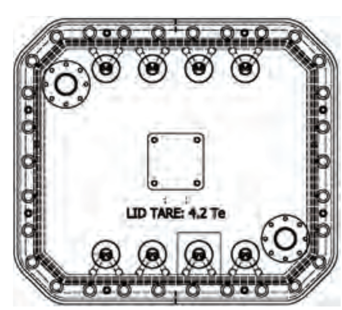
5

How to Demonstrate Feasibility of Passive Interim Storage

- The Self-Shielded Box (SSB) for the First Generation Magnox Storage Pond (FGMSP) at Sellafield has 8 filter vented ports for H_2 and H_2O removal – each port has two bore holes







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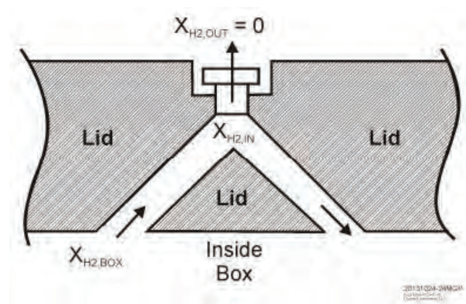
How to Demonstrate Feasibility of Passive Interim Storage

- Principle for hydrogen removal through the thick shielding: Minimize the resistance to hydrogen removal.
- A single bore hole connecting the filter to the inside of the box (cask) creates an infeasibly large resistance
- Using two bore holes induces a sufficient recirculation flow to reduce the bore hole resistance to much less than the filter resistance
- FAI has a patent application on the design



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How to Demonstrate Feasibility of Passive Interim Storage

- Fukushima core debris is anticipated (IRID designs) to include three layers of packaging:
  (1) Small unit cans holding retrieved debris, shown here in green
  (2) Filter vented canisters holding unit cans that are intended to be stacked and have shock absorbing bottom buffers and upper lid features for handling and gas sampling
  (3) A cask or similar overpack holding multiple canisters (not shown)
- Therefore, we must demonstrate that hydrogen removal from all layers leads to less than 4% hydrogen at the worst location

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FAI Experimental Proof of Feasibility of Passive Interim Storage

- FAI has experimental proof of hydrogen removal from proposed Fukushima waste packages, obtained for TEPCO this FY.
- We cannot show the data at this time, but we can state we have achieved complete success: Individual increments for changes in the hydrogen concentration are small enough that there cannot be flammable gas mixtures in a package even with the bounding hydrogen source
- Our experimental results agree with a priori model predictions:
  - We can predict hydrogen removal through filters: The dominant resistance is mass transfer to and from the filter surface, NOT diffusion
  - We can predict hydrogen removal through our modified canister lid design which maximizes filter area and provides a recirculation path
  - We can predict hydrogen removal through the cask lid using a design based upon the Sellafield Self-Shielded Box

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Data Needs to Support Passive Interim Storage

- Regarding passive hydrogen removal:
  - Currently a rather pessimistic bounding high hydrogen source rate is assumed for design, which guarantees a conservative design. Better data or less conservative calculations for radiolysis would potentially allow design simplifications but would also quantify design conservatism.
  - The FAI hydrogen removal data are properly scaled for both the single canister scale and the cask scale. As designs evolve, new experimental demonstration may be required, and perhaps the regulator will require a full-scale mockup.

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10

Data Needs to Support Passive Interim Storage

- Key properties regarding water removal:
  - Macroscopic porosity of the debris determines the water inventory and affects the resistance for evaporation. This affects short-term (a few years) water loss and has been quantified by FAI modeling for TEPCO.
  - Debris particles themselves will have some internal porosity which adds to the water inventory. This affects long-term dryout (more than a few years, requires further study).
  - Hydrate water (chemically bound, such as $UO_3.2H_2O$, $Fe(OH)_3$, etc will not be passively removed. Studies for radiolysis of these hydrates will facilitate eventually moving from interim storage to sealed final storage.
  - The particle size distribution of as-retrieved debris will influence permeability and the resistance for evaporation.

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Data Needs to Support Passive Interim Storage

- Key properties regarding decay heat removal:
  - Thermal conductivity of the fuel debris is the dominant parameter
  - Wet debris should have a thermal conductivity close to that of water and this was the technical basis for both Hanford and Sellafield debris
  - Debris thermal conductivity varies strongly with water content
  - Debris surrogates can be used to obtain data – we have previously recommended that leftover material from molten core concrete reaction experiments could be used
  - Density and specific heat only affect transient temperature profiles, so focus on thermal conductivity
  - We assume packaging properties will be relatively well-known

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12

13

C.4.4. Topic Area 4 - Combustible Gas Effects

C.4.4.1. Comments on Recent NRAJ Investigations and Experimental Investigations

Fukushima Forensics: Combustible Gas Effects
Comments on Recent NRAJ Investigations
Comments on Proposed and Prior Experimental Investigations

Wison Luangdilok
H2Technology LLC

DOE Reactor Safety Technology Expert Panel Forensics Meeting
Nuclear Energy Institute, 1202 F Street NW, Suite 1100, Washington, DC
November 28-30, 2021

H2TECHNOLOGY LLC

Outline

- Motivation behind NRAJ's proposed experiments on thermal decomposition of polymeric materials
  - When is hydrogen flame visible?
  - What was the orange flame observed briefly during the 1F3 explosion?
- Review of recent experiments relevant to the mechanism of the 1F1 and 1F3 explosions
- Dissecting the multi-mode 1F3 explosion dynamics for improved understanding of the combustion modes involved.
- Comments on NRAJ proposed combustion experiments
- Suggestion of experiments

2 H2TECHNOLOGY LLC

Is hydrogen flame visible?

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PNNL, 2017, Hydrogen propane flame comparison, YouTube.com video, Pacific Northwest National Laboratory,

Pure hydrogen diffusion flame is visible in a dark background light.

H2TECHNOLOGY LLC

The Color of Hydrogen Flames on a Gas Stove

Natural gas

Hydrogen

Copyrighted images removed

A recent study by UC Irvine has concluded that the yellow-reddish flame is caused by fine particles in the hydrogen gas

One of the most common reactions is between hydrogen and carbon in the steel: Fe_3C+2H_3 —> $3Fe+CH_4$

In a flame, Fe^{3+} ion can result in orange color.

Y. Zhao, et al. 2019, Investigation of visible light emission from hydrogen-air research flames, Int. J. Hydrogen Energy, 44, 22347-22354

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Hydrogen Flame during a Hydrogen Tank Explosion set off by a Propane Fire.

Copyrighted images removed

N. Weyandt, 2007, Intentional Failure of a 5000 psig hydrogen Cylinder installed in an SUV without Standard required Safety Devices, SAE Technical Paper Series 2007-01-0431

What was this orange standing flame appearing immediately before the rise of a mushroom-shaped fireball during the 1F3 explosion ?

Copyrighted images removed

NRAJ is investigating an unaccounted source of combustible gases from a thermal decomposition of hydrocarbon materials present inside the PCV such as cable insulators, paint, etc. by laboratory experiments.

https://www.nsr.go.jp/data/000325406.pdf

A Fire Event at 1F3 on March 21, 2011

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Yasui, DOE Reactor Safety Technology Expert Panel Forensics Meeting, Nov. 29, 2021

NRAJ is investigating an unaccounted source of combustible gases from a thermal decomposition of hydrocarbon-based polymeric compounds (epoxy, silicone rubber, chloroprene, polyurethane,..) present inside the PCV cable insulators, paint, etc. by laboratory experiments.

Could this be the missing piece?

- From the overall mass balance perspective, combustible gases must be generated fast enough at the right amount and at the right time to support the explosions at the recorded times. In order for this to happen, vessel lower head failure would also need to happen early enough for MCCI to generate hydrogen. So far code calculations could not do this. So this might be the missing piece, especially if MCCI is considered unlikely now based on Mitch's presentation yesterday.
- NRA is investigating an unaccounted source of combustible gases from a thermal decomposition of hydrocarbon materials present in cable insulators and paint by laboratory experiments.
- This is in addition to hydrogen from in-vessel and ex-vessel oxidation of core materials at high temperatures.
- Investigation of radiolytic decomposition of these materials has not been mentioned.
- The experimental research on this is a welcome initiative.
- The research would provide what types of gases and how much of these gases can potentially be generated per each kilogram of materials. Coupling with the plant material inventory, the total potential amount can be estimated.
- If the potential amount is substantial, this could be the missing piece.

Potential Sources of H2 Generation in the 1F3 Accident

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

X= additional source from thermal decomposition of organic materials in electrical cables, insulators, and paints

Estimate of total H2 generation by time of 1F3 explosion **2100 ~ 2400 kg**

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

9

Code Calculations of Hydrogen Generation

BSAF phase 2 as of 2019

Participating organization	SA code used in the analysis	Calculated in-vessel H2 generation (kg)	Calculated ex-vessel H2 generation (kg)	Total calculated H2 generation (kg) up to 1F3 explosion vs expected value from ballpark estimate	Calculated RPV failure time (hr)	RPV failure mode
VTT	MELCOR	1220	1200	2420 vs 2100-2400	43.3	penetration
SNL	MELCOR	1010	700	1710 vs 2100-2400	58	user specified
JAEA	THALES/KICHE	790	875	1665 vs 2100-2400	46.5	vessel melt
IAE	SAMPSON	790	500	1290 vs 2100-2400	55.2	creep
PSI	MELCOR	1180	0	1180 vs 2100-2400	73.1	penetration
IRSN	ASTEC	1150	0	1150 vs 2100-2400	55.4	creep
NRA	MELCOR	910	100	1010 vs 2100-2400	49.4	penetration
CRIEPI	MAAP5	600	0	600 vs 2100-2400	102	penetration

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NRAJ Proposed Combustion Experiments

https://www.nsr.go.jp/data/000325406.pdf

2 Proposed hydrogen combustion experiment

1 Proposed range of hydrogen concentrations for investigation

4wt% H2=37.6 vol% H2
10wt% H2=61.6 vol% H2

3 Proposed mixed gas combustion experiment

These are experiments to study the chemistry aspect of the problem.
The fluid dynamics aspect of the problem also needs to be addressed.

Review of Recent Experiments relevant to the Mechanism of the 1F3 Explosion

- The NRAJ initiative on the study of decomposition of hydrocarbon materials and combustion characteristics of hydrogen gas mixtures with added organic gases is not enough to understand the entire picture of the 1F3 EXPLOSION.
- The dynamics of the explosion need to be studied also.
- (This orange flame might had helped ignited the 5thF gases following the complete failure of the 5thF roof that allowed gases to mix with air.)
- Up to this point, the average 5thF gases are assumed to be too rich in fuel and non-flammable.

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Mushroom-shaped Fireball Evolution Experiments: The $\phi > 1$ Experiments

Guo, et al. 2015, Effect of ignition position on vented hydrogen-air explosions, Int. J. Hydrogen Energy 40 (2015) 15780-15788

Copyrighted images removed

- Experiments were conducted in a stainless cylindrical pressure vessel with two necks at its waist.
- The size of neck's square cross section is 7cm x 7 cm, and the length of necks is 10 cm
- One neck is sealed with a flange, and the other with a diaphragm that brakes open at ~50 kPa internal overpressure.
- Both inner diameter and length of the cylindrical vessel are 25 cm.

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Mushroom-shaped Fireball Evolution Experiments: The $\phi > 1$ Experiments

Guo, et al. 2015, Effect of ignition position on vented hydrogen-air explosions, Int. J. Hydrogen Energy 40 (2015) 15780-15788

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- For each test, one of three ignition positions was used
  - BI: back ignition,
  - CI: central ignition, or
  - FI: front ignition
- The ignition energy is kept about 500 mJ in every test.
- The initial pressure and temperature of hydrogen-air mixtures were 1 atm. and 280 K, respectively. Five hydrogen-air equivalence ratio (ϕ) of 0.6, 1.0, 1.6, 3.0 and 5.0 were tested.

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Mushroom-shaped Fireball Evolution $\phi > 1$ Experiments

Guo, et al. 2015, Effect of ignition position on vented hydrogen-air explosions, Int. J. Hydrogen Energy 40 (2015) 15780-15788

Flame

Unburned gases

Copyrighted images removed

Impact of ignition position on flame movement and combustion completeness when the flame front reaches the vent exit.

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Mushroom-shaped Fireball Evolution $\phi > 1$ Experiments

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Mushroom-shaped Fireball Evolution $\phi > 1$ Experiments

Guo, et al. 2015, Effect of ignition position on vented hydrogen-air explosions, Int. J. Hydrogen Energy 40 (2015) 15780-15788

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The length of the mushroom stem increases with an increasing equivalence ratio.

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Summary and Observations

- The evolution of external flames in a form of mushroom-shaped fireball can be observed only for $\phi \geq 1.0$ in the experiments.
- The initial explosion that occurred on the 4th floor of 1F3 looks like an internal explosion with a back ignition of this experiment with $\phi<1$.
- The follow-on explosion on the 5th floor looks like an external explosion of this experiment with $\phi>1$.

Copyrighted images removed

Luangdilok et al 2015, NURETH-16

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Mushroom-shaped Fireball Evolution $\phi > 1$ Experiments

Guo, et al. 2015, Effect of ignition position on vented hydrogen-air explosions, Int. J. Hydrogen Energy 40 (2015) 15780-15788

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Guo, et al. 2015, Effect of ignition position on vented hydrogen-air explosions, Int. J. Hydrogen Energy 40 (2015) 15780-15788

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Explosion Experiments with an open vent from time zero

Daubech et al. 2013, Hydrogen-Air Vented Explosions: New Experimental Data, 5th Int. Conf. Hydrogen Safety, Brussels, Belgium

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Six pressure gauges were used to measure the overpressure evolution inside and outside the chamber.

The 4 m rectangular experimental chamber (2 m height, 2 m width and 1 m depth) is equipped with transparent walls and is vented (0.25 and 0.5 m square vents).

Vented Explosion Experiments with $\Phi < 0.47$ ($16.5\% H_2$)

The hydrogen-air cloud was seeded with micro particles of ammonium chloride to visualize the propagation of the flame and the movement of the burned and unburned gases inside and outside the chamber.

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The hydrogen-air cloud was seeded with micro particles of ammonium chloride to see the propagation of the flame.

Flame exists the vent at 135 ms

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Daubech et al. 2013, Hydrogen-Air Vented Explosions: New Experimental Data, 5th Int. Conf. Hydrogen Safety, Brussels, Belgium

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The hydrogen-air cloud was seeded with micro particles of ammonium chloride to visualize the propagation of the flame and the movement of the burned and unburned gases inside and outside the chamber.

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The hydrogen-air cloud was seeded with micro particles of ammonium chloride to see the propagation of the flame.

Flame exists the vent at 135 ms

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Summary of Vented Explosion Experiment with Back Ignition

- A fireball with almost no stem can be formed at the vent exit.
- With back ignition, a relatively large amount of unburned cloud can be pushed out and formed a spherical unburned cloud at the vent exit.
- Later on the flame front moves from the ignition point toward the vent until it reaches the vent exit and ignites the external cloud.
- This mode of explosion looks like the 1F1 explosion. The roof and side wall structure of the 5th floor of 1F1 can be blown away with little internal pressure.

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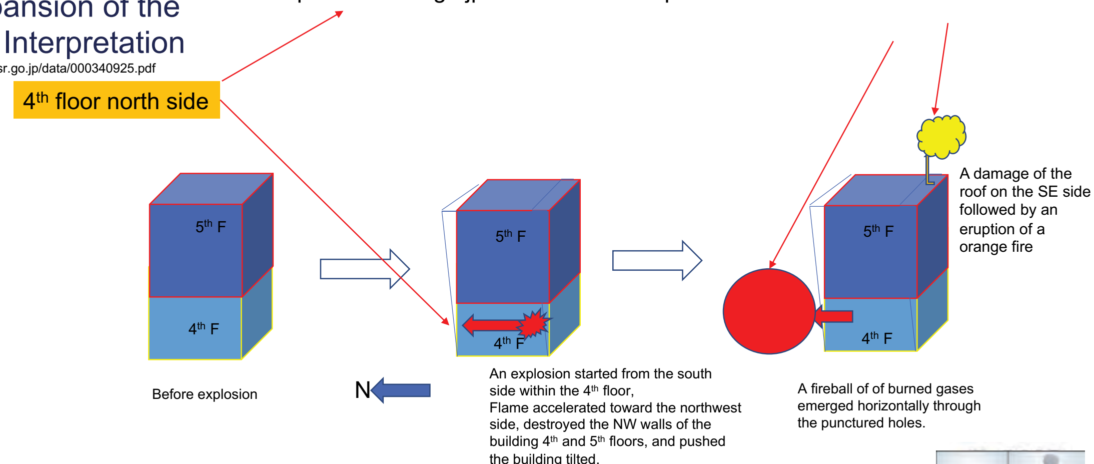
Dissecting the 1F3 Explosion Mechanism :

an expansion of the NRAJ Interpretation

https://www.nsr.go.jp/data/000340925.pdf

https://www.nsr.go.jp/data/000334168.pdf

Copyrighted images removed



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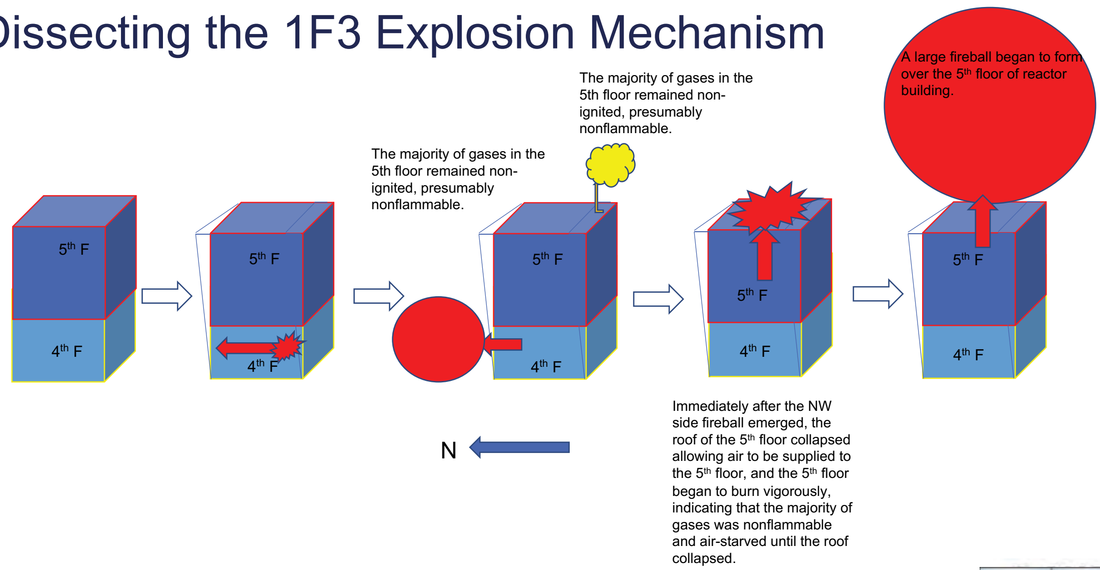
Dissecting the 1F3 Explosion Mechanism



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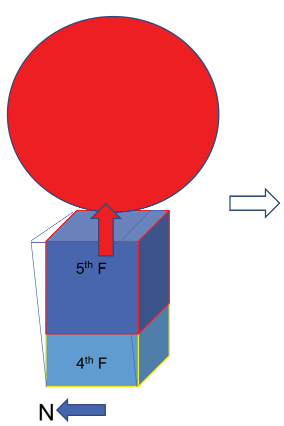
Dissecting the 1F3 Explosion Mechanism

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



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Additional Remarks

- A reasonable explosion mechanism can be constructed based on the insight gained from previous hydrogen explosion experiments with the equivalent ratio ranging from a value less than one to a value of six covering the entire range of flammable hydrogen-air mixture from lean to very rich mixture.
- The explosion mechanism indicates the explosion on the 4th floor of 1F3 is an explosion of Φ<1 (lean fuel), while the 5th floor mushroom fireball explosion is a Φ>1 (rich fuel).
- The experiments discussed here provide the proof that It has been known for a mushroom-shaped long-stem fireball to occur occur the gas mixture has to be very rich in fuel (Φ>1), . The fuel has to be much richer than the stoichiometric ratio so that, following an initial ignition, there is excess unburned fuel to drive the fireball formation.
- Any proposed explosion experiment similar to the ones discussed here but with the combustion vessel designed to have a geometric and failure characteristic of the 4th and 5th floors of 1F3 reactor building would be very helpful.

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Ideal Hydrogen Range for Experiments

Range Φ<1 → provide basic understanding
Range Φ>1 → provide answers to 1F3
Uncertainty → Steam concentration

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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.

https://www.nsr.go.jp/data/000334168.pdf

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C.4.4.2. Previous Work on Organics to Containment

Previous Work on Organics to Containment

Reactor Safety Technology Experts Meeting

November 2021

Some previous work on Organics in Containment

- Some previous studies as formation of Volatile Organic Iodides in Containments
  - Acidification of aqueous volumes
    - Nitric acid from radiolysis of air
    - Beahm has called attention to HCl release from cable insulation
  - Sources of organic reactants
    - Wren has called attention to solvents from paints
    - Cables also be source of organics
- Organic Compounds in Containment - coatings, oils, seals, and plastics
  - Beahm, E.C., Shockley, W.E., et al., 1985. Organic iodides formation following nuclear reactor accidents, NUREG/CR-4327 – ORNL/TM-9627
- Pyrolysis/Radiolysis of Cables – Several tons of cables can be present in containment
  - R. L. Clough and K. L. Gillen, "Investigation of Cable Deterioration Inside Reactor Containment," Nucl. Technol. 59, 344 (1982)
  - Gillen, K. T., R.L. Clough, and L.H. Jones, "Investigation of Cable Deterioration in the Containment Building of the Savannah River Nuclear Reactor," NUREG/CR-2877, SAND81-2613, August 1982.
  - 2012 NUREG Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE), NUREG/CR-2010 Vol. 1 and Vol. 2
- Stainless-steel/B4C reaction and and formation of carbonaceous gases: PHEBUS-FPT3, BECARRE (PHEBUS-ISTP), DF-4,
- Cable also being studied as part of the ESTER project
    - Germans conducting experiments on different cable types

Hypalon

- **Nominal Chemical Formula**

  $C_{85}H_{157}Cl_{13}SO_2$
- **Constituents**

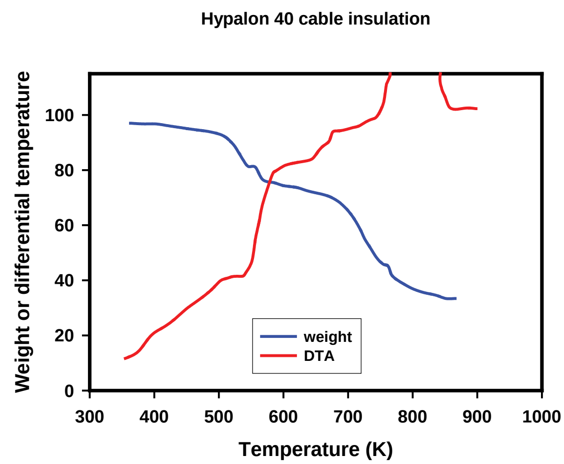
-chlorosulfonated polyethylene	100
- litharge	30
- hard clay	60
- chlorinated parafin	20
- red iron oxide	15
- plastisizers etc	10

Threats to Cables

- **Thermal**
  - very high temperatures in some places
  - > 373 K in most places
- **Radiolytic**

  ~ 1 Mrad/hr

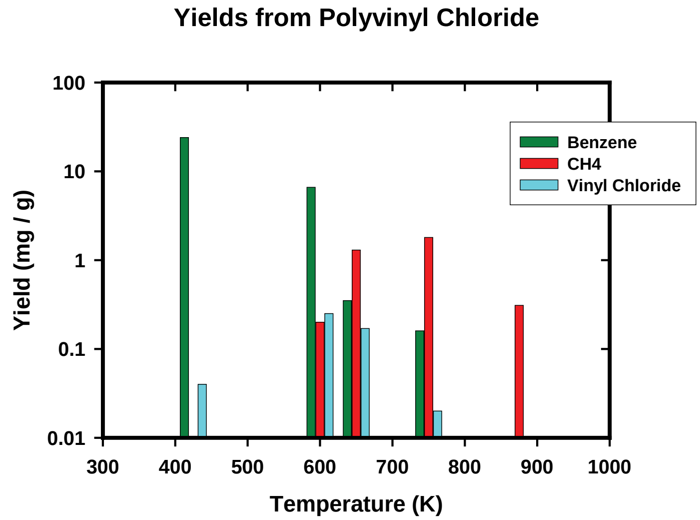
Pyrolysis of Hypalon



Some Yields

HCl	273 mg/g	333 mg/g
CO_2	616	1182
CO	67	90
CH4	6.6	6.8
C2H4	2.3	2.0
C2H6	3.0	2.9
Proplylene	2.0	1.4
Propane	1.7	1.4
Vinyl chloride	3.3	2.6
1-butene	1.1	0.58
Butane	1.1	0.74
1-pentene	0.35	0.15
Cyclopentene	0.14	0.07
Benzene	10.0	11.0
Toluene	0.94	1.0

Pyrolytic Yields



Volatile Organic Products of Pyrolysis

- **Diverse species in moderate amounts with highly variable reactivity**

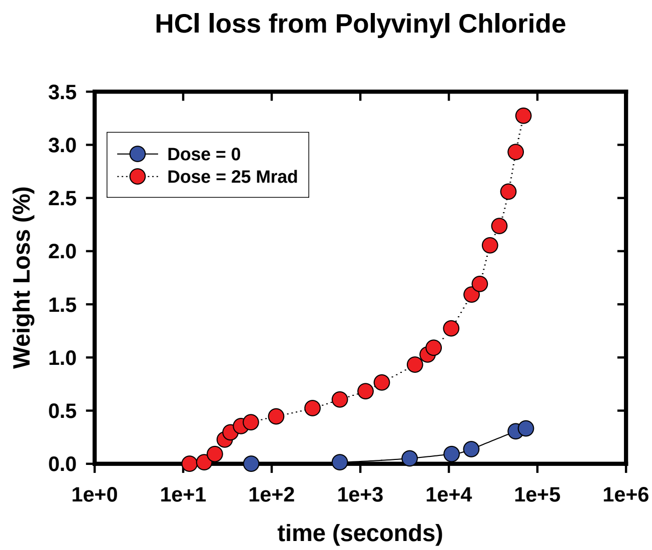
  olefinic > aromatic >aliphatic
- **Yields vary with temperature**
- **Benzene is important at low temperature and decreases with increasing temperature**
- **Olefinic yields such as that of vinyl chloride and propylene pass through a maximum with increasing temperature**
- **Methane becomes more important at higher temperature**

Radiolytic Attack

- **Produces HCl as well as organics**
- **Accentuated in air**
- **HCl release increases with temperature**
- **Suppressed by additives used in forming the cable insulation**

Synergism

- **Cable aging studies by Clough and coworkers show that embrittlement of insulation exhibits a synergism between temperature and dose.**
- **Polyvinyl chloride degradation exhibits synergism between temperature and dose.**
- **Onset of HCl release and organic vapor release occurs at lower temperatures in a radiation field.**

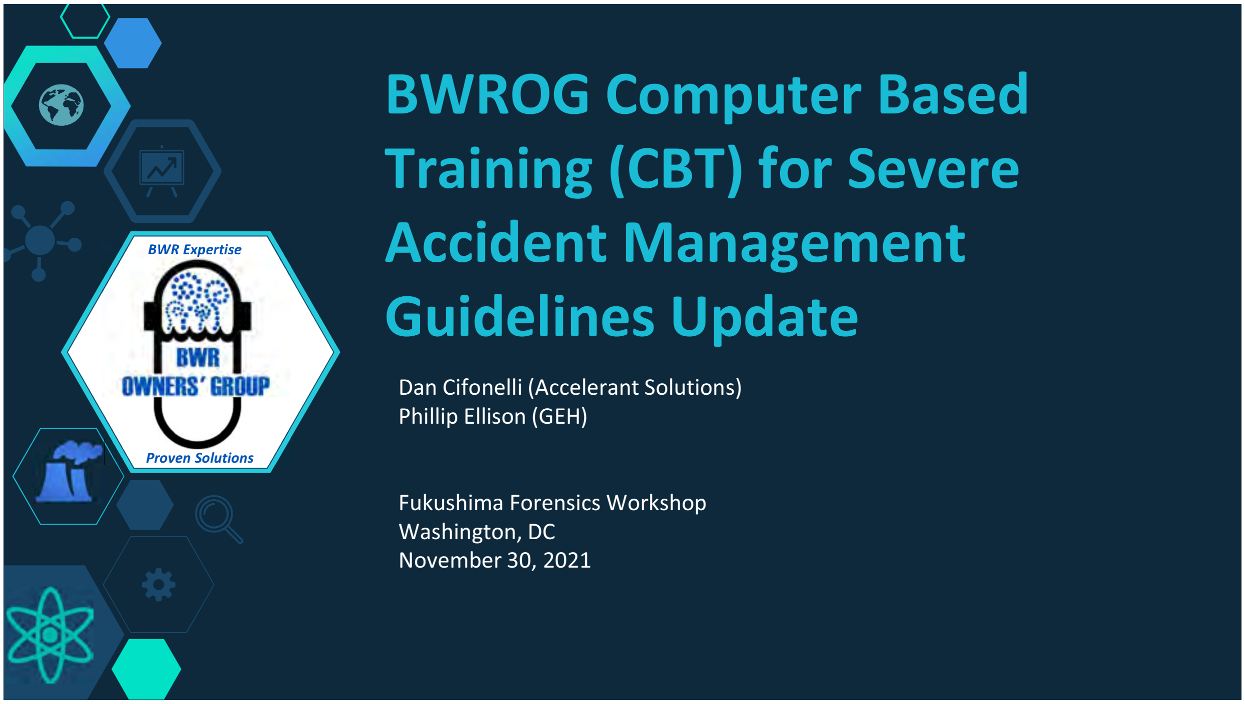


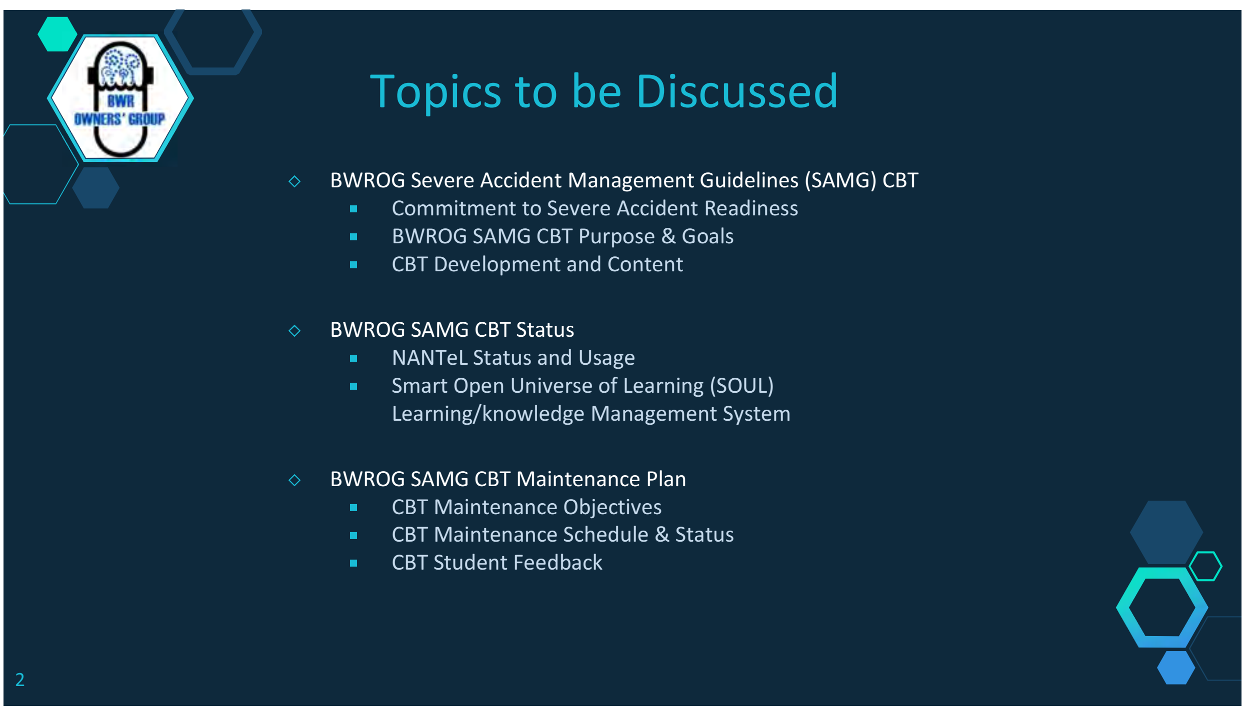
Conclusions

- Substantial previous work on organic compounds in containment during Severe Accidents
  - Original focus on coatings, oils, seals, plastics
- Cable insulation may be an additional source of diverse and often quite reactive organic vapors to containment. Augments volatilization or aqueous extraction of solvents from paint.
- Synergy between temperature and dose
- Ongoing work in ESTER

C.4.5. Topic Area 5 - Operations and Maintenance

C.4.5.1. BWROG Computer Based Training for SAMGs Update





Commitment to Severe Accident Readiness

- NUREG-0737 Item I.C.1, "Guidance for the Evaluation and Development of Procedures for Transients and Accidents" & NRC Generic Letter 88-20 & Supplements –Post-TMI severe accident management Inception
- NEI 91-04, "Severe Accident Issue Closure Guidelines" – Industry initiatives for severe accident management which led to the creation of generic Severe Accident Guidelines
- INPO IER L1-13-10, "Nuclear Accident at the Fukushima Daiichi Nuclear Power Station" - Focuses attention on responsibility
- ACAD 15-10, "Guidelines for the Training and Qualification of Emergency Response Personnel" - Foundation for ERO Programs
- Order EA-13-109, Phase 2, "Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions" – Revises Severe Accident Strategy
- 10CFR50.155, "Mitigation of Beyond-Design-Basis Events (MBDBE)" Rule Making & NSIAC Commitments – Renews industry commitment to Beyond Design and Severe Accident Events & timely plant specific updates to Owner's Group SAGs

3

BWROG SAMG CBT Purpose & Goals

- Using the standard SAT process (ADDIE: Analyze, Design, Develop, Implement and Evaluate) Computer Based Training (CBT) was produced and is being maintained for technical certification of US BWR SAMG Decision Makers, Evaluators & Implementers
  - The BWROG SAMG CBT is the minimum standard for BWR 1-6
  - Focus on Operational fundamentals, such as Parameter Control
  - Emphasize the importance SAMGs to Safety, Preparation and Knowledge
  - Assist utilities in EPG/SAG Rev 4 SAMG Training Program Management
  - Maximize Student Interaction and Engagement thru innovative training
  - Make comprehensive use of Fukushima Case Studies
  - Consider PWROG SAMG CBT Program to assist industry synergies/consistencies
  - Develop certification logic for ERO specific roles per NEI 94-01
  - Provide containment specific Modules and Certifications for
    - Mark I & II containments, and
    - Mark III containments
  - Deliver on NANTeL Platform (US Utilities)
  - Deliver on SOUL Platform (Non-NANTeL Organizations)

4

CBT Development and Usage

- The BWROG SAMG CBT was issued for US utility use on September 21, 2020
- The SAT Process Elements evaluated are described along with background information in BWROG Technical Product: TP20-4-008r1 Development of CBT Program for BWR Severe Accident Guidelines, (Rev 1)
- The Training Program and Content is described in BWROG Technical Product: TP20-1-008r1 EPC CBT Program Description, (Rev 1)
- The CBT is designed for Emergency Response Organization (ERO) roles of Implementers, Evaluators and Decision Makers as committed to in NEI 91-04, Severe Accident Issue Closure Guidelines and advised in ACAD 15-10, "Guidelines for the Training and Qualification of Emergency Response Personnel"
- The CBT is Evaluated ("E" in ADDIE) periodically based on current and developing data. Each Module in the CBT includes a Student evaluative survey. Since initial CBT release, over 8,000 surveys have been completed. There is a current steep learning curve that is assumed to begin to stabilize in 2022, which is driving current maintenance activities with a goal to release a Revision in early 2023

5

CBT Content

- 32 Mods (Unique Courses)
- 4 ERO Positions
- 2 Containment Designs
- Initial & Continuing training
- 16 Certifications

Reference:
BWROG Technical Product: TP20-1-008r1 EPC CBT Program Description, (Rev 1)

Modules for Position Specific Certification & Estimated Times

Training Module	Initial Min	Requal Min	Implementer RO	Implementer EO	Evaluator	Decision Maker
M1 SAMG Introduction	25	20	I-R	I-R	I-R	I-R
M2 Lessons Learned from SA's	15	N/A	I	I	I	I
M3A Phenomenology – Condensed	20	15		I-R		
M3B Phenomenology – Part 1	30	20	I-R		I-R	I-R
M3C Phenomenology – Part 2 #	20	18	I-R		I-R	I-R
M4A SAMG Strategy Overview #	20	N/A		I	I	I
M4B SAMG Overview (Licensed & Requal) #	12	10	I-R	R	R	R
M5 TSG Overview	15	10	I-R		I-R	I-R
M6 TSG Part 1 (Instrumentation and Parameter Control)	20	12	I-R		I-R	I-R
M7 TSG Part 2 (Plant Status Assessment) #	55	45	I-R		I-R	I-R
M10 SAGs Modes 1-4 #	30	20	I-R		I-R	I-R
M11 SAGs Refuel #	15	N/A	I		I	I
M8 TSG Part 3 (Functional Status and Action Assessments) #	35	25	I-R		I-R	I-R
M9 TSG Worksheets #	35	25			I-R	
M12 Fukushima-2 Case Study	40	30	I-R		I-R	I-R
M13 SAMG Performance Demonstration - DM #	30	25				I-R
M14 SAMG Performance Demonstration – E #	30	25			I-R	
M15 SAMG Performance Demonstration - IM (Licensed Operators) #	45	40	I-R			
M16 Plant-Specific Check – DM	5	5				I-R
M17 Plant-Specific Check – Evaluator	5	5			I-R	
M18 Plant-Specific Check – Implementer	5	5	I-R			
Total Minutes / hour Initial Training			362/6.0	80/1.3	405/6.8	370/6.2
Total Minutes / hour 4-Year Requal			255/4.3	45/0.8	280/4.7	255/4.3

6

What the CBT is NOT

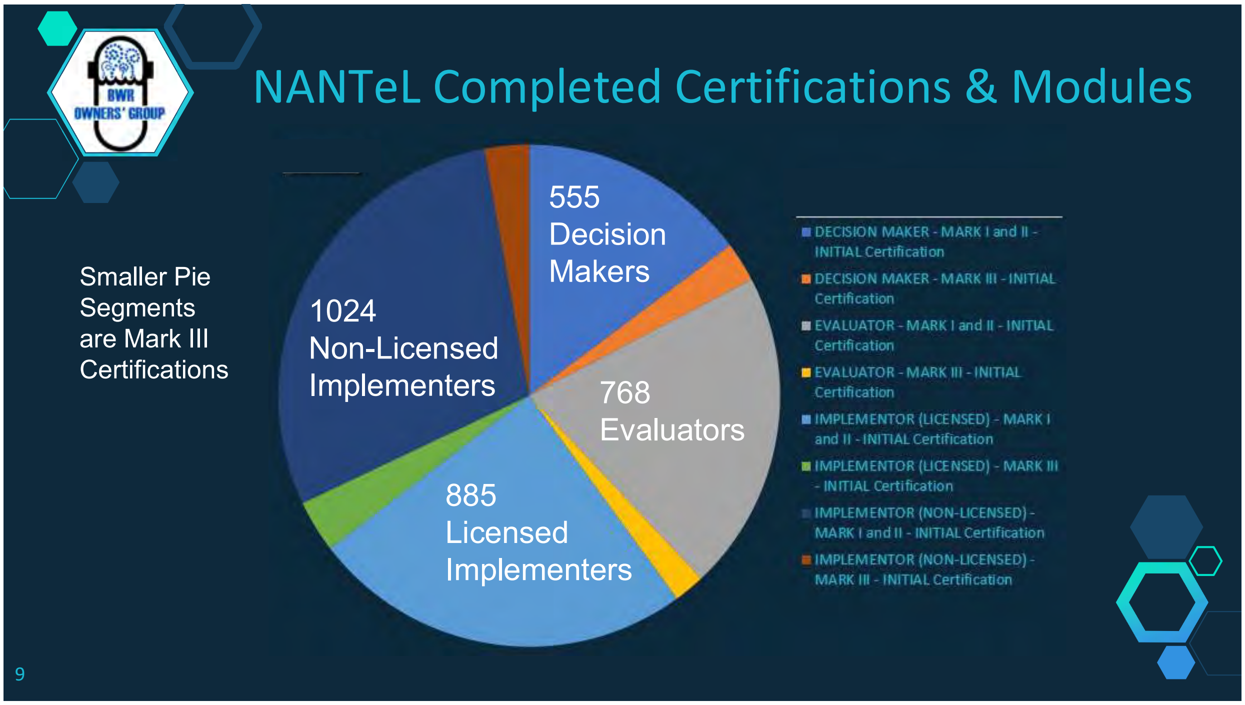
- NOT Plant Specific Training
  - Utilities are required to develop Plant Specific Guidelines
  - The CBT includes a Plant Specific Check Module and emphasizes the generic nature of the training
- NOT ERO Qualification – this is left to Utilities under ERO Program
- NOT FLEX/ELAP/SBO Systems, Controls, Procedures Training – this is generally part of Operator Training
- NOT Hardened Containment Vent Systems, Controls, Procedures Training– this is generally part of Operator Training
- NOT an INPO Accredited Program
- NOT Required Licensed Operator Training, but as ERO Implementors provides Technical Training for Licensed and Non-Licensed Operators

7

NANTeL Completed Certifications & Modules

Certification	Mark I/II Certs Completed (A)	Mark III Certs Completed (B)	Tot Certs Completed (A+B) OR (C)	Modules per Certification (D)	Total Modules per Certifications Credited (C X D)
Decision Makers	555	90	645	14	9030
Evaluators	768	73	841	15	12615
Licensed Implementers	885	116	1001	14	14014
Non-Licensed Implementers	1024	103	1127	4	4508
Totals:	**3232**	**382**	**3614**		
				Total Modules Credited	**40167**

8



NANTeL Completed Certifications & Modules
Smaller Pie Segments are Mark III Certifications
555 Decision Makers
1024 Non-Licensed Implementers
768 Evaluators
885 Licensed Implementers
DECISION MAKER - MARK I and II - INITIAL Certification
DECISION MAKER - MARK III - INITIAL Certification
EVALUATOR - MARK I and II - INITIAL Certification
EVALUATOR - MARK III - INITIAL Certification
IMPLEMENTOR (LICENSED) - MARK I and II - INITIAL Certification
IMPLEMENTOR (LICENSED) - MARK III - INITIAL Certification
IMPLEMENTOR (NON-LICENSED) - MARK I and II - INITIAL Certification
IMPLEMENTOR (NON-LICENSED) - MARK III - INITIAL Certification
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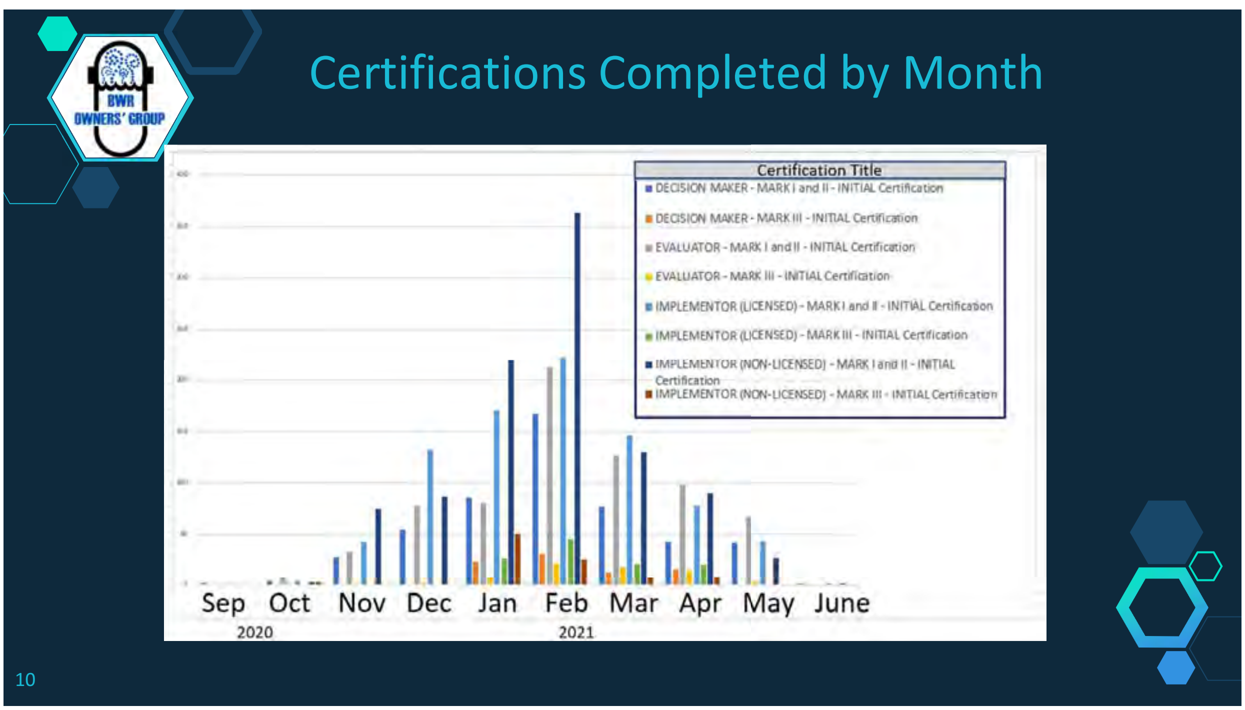


Certifications Completed by Month
Certification Title
DECISION MAKER - MARK I and II - INITIAL Certification
DECISION MAKER - MARK III - INITIAL Certification
EVALUATOR - MARK I and II - INITIAL Certification
EVALUATOR - MARK III - INITIAL Certification
IMPLEMENTOR (LICENSED) - MARK I and II - INITIAL Certification
IMPLEMENTOR (LICENSED) - MARK III - INITIAL Certification
IMPLEMENTOR (NON-LICENSED) - MARK I and II - INITIAL Certification
IMPLEMENTOR (NON-LICENSED) - MARK III - INITIAL Certification
Sep Oct Nov Dec Jan Feb Mar Apr May June
2020
2021
10

SOUL Learning Management System

- The CBT has been requested by the non-NANTeL users. This will allow NRC and International BWROG Members access without the use of NANTeL. To allow access the CBT has been hosted on an independent Learning Management System (LMS)
  - Smart Open Universe of Learning (SOUL)
- Other uses of SOUL LMS
  - Expert Social Media
  - Knowledge Transfer System
  - Secure Platform
  - Documentation Database, Storage & Retrieval
- SOUL Help
  - Phil Ellison: Phillip.Ellison@ge.com or call (910)-508-8772
  - Dan Cifonelli: danielcifonelli@discoveraccelerant.com or call (315)-529-8641
  - Jose Albert: jmalbert@tecnatom.es

11

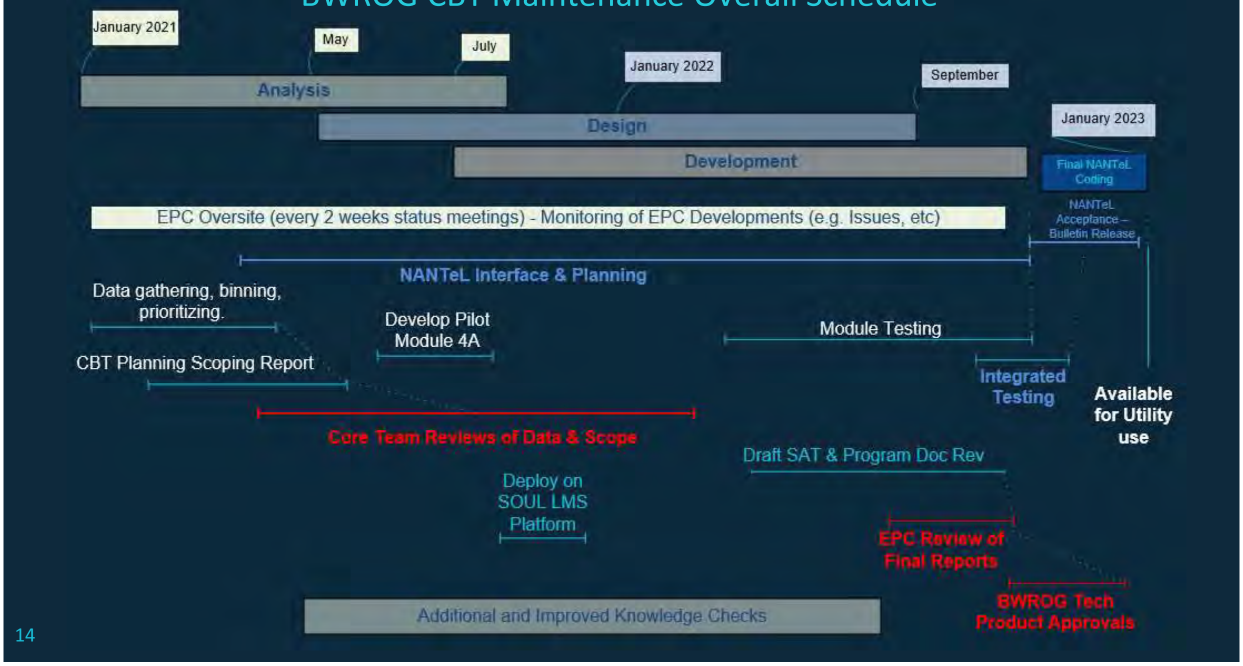
CBT Maintenance Plan

- CBT Maintenance Analysis, Design, Development and Revision work through 2022 with NANTeL release to Utilities scheduled for January 2023
- Evaluation and Minor Maintenance ~ each two years with the first being in 2024.
- Revision every ~ 4-6 years as needed (depending on assessment of SAG technical content change activity, current industry events, BWROG product initiatives and Utility needs)
- Maintain a non-NANTeL LMS, SOUL, for BWROG SAG CBT for BWROG Membership that do NOT have NANTeL access in parallel with the maintenance of the US NANTeL CBT

Note: The first three items above would include coordination with NANTeL and US member utilities. Item 4 would use the developed material for NANTeL and host it on a separate LMS, SOUL, accessible by non-NANTeL members such as NRC or International BWROG Members

13

BWROG CBT Maintenance Overall Schedule



14

BWROG SAMG CBT Maintenance 2021-22

- Analyze (generally complete), Design (in-progress) , Develop (in-progress) SAMG CBT Modules based on:
  - Student Feedback Survey Results
  - BWROG Known Improvement Opportunities
  - NANTeL Data and Incident Reports
  - BWROG EPC SAMG selected Approved Issues
- Scoping Document provided for Long-Term Maintenance, considering International needs, opportunities & strategies for improvements such as
  - Development for Non-NANTeL users (use of SOUL)
  - Considerations of other BWROG CBT
    - Fukushima 1 & 3
    - DAEC Derecho Event
    - TSG Skill Set Training
    - EOP/SAG Development Training (EOP Coordinators)
    - EPGs/SAGs Generic Workshop
- Update SAT Report and Training Program Descriptions (BWROG Technical Products)
- Coordinate Reversioning process/schedules with NANTeL

15

BWROG SAMG CBT Maintenance – 2021-22

- Initial Training Certifications require about 10-12 hours of student time
- Goal: Reduce Initial Training time by ~20% without impacting technical content (8-9 hours goal)
  - Reduce Module 8 soft content, focus on RPV & Containment Water Addition Strategies, for Requalification by ~ 50%, near complete estimated times reduce from ~65 minutes to ~35 minutes with no reduction in technical content
  - Demonstration Modules (Game) – Separate Module to eliminate redundancies
  - Provide more credits for previously seen material for multi-certified individuals
  - Reduce audio and use the written slide alone when deemed appropriate
  - Automation of content appearance to reduce student clicks

16

Overall Student Survey Results

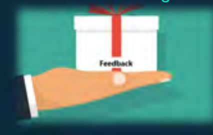
Over 8,300 Student Feedback Surveys completed. Below is an overall score summary by Module.

Module Name	Records	Score
SAMG Introduction	1200	6.7
Lessons Learned from the Nuclear Industry's Three Severe Accidents	993	7.0
Severe Accident Phenomenology (Condensed)	307	7.1
Severe Accident Phenomenology (Part 1)	580	6.7
Severe Accident Phenomenology (Part 2)	425	6.9
SAMG Strategy Overview	554	7.1
SAMG Strategy Overview for Licensed Implementers	176	6.0
SAMG TSG Overview	465	6.8
Technical Support Guidelines (Part 1 - Instrumentation and Parameter Control)	395	6.7
Technical Support Guidelines (Part 2 - Plant Status Assessment)	381	6.7
Technical Support Guidelines (Part 3 - Functional Status and Action Assessments)	372	6.5
TSG Worksheets (Calculation Aids)	187	6.8
SAG-1, RPV Control and SAG-2, Containment and Radiological Release Control (Mode 1 - 4)	350	6.0
SAG-1, RPV Control and SAG-2, Containment and Radiological Release Control (Refuel	326	6.7
Fukushima 1F2 Case Study	354	7.5
Decision Makers - Demonstration	120	7.0
Evaluators - Demonstration	210	7.0
Implementers (Licensed Operators) - Demonstration	153	5.4
Plant Specific Check - Decision Makers	117	7.1
Plant Specific Check - Evaluators	191	7.1
Plant Specific Check - Implementers (Licensed Operators)	140	5.8
Total/Average	**8367**	**6.8**

17

Student Survey Results

Feedback is a gift



- Over 8,300 Student Feedback Surveys completed.
  - Significant written feedback in Open Ended Question
  - Wide opinion from great to adequate to improvement needed
- Some examples:
  - “Excellent coverage of this vital information”
  - “Interactive and challenging”
  - “Scenario at the end really brought together a lot of the concepts”
    - “Three happiest events of my life: Getting married, birth of children, finishing this training”
- Fukushima Daichi Unit 2 Case Study had very positive feedback, example:
  - “Great lesson, best case study that I have seen. Thanks!”
  - “Good use of video capability. Actual footage of Fukushima core conditions is impactful”
  - “Best module yet! I liked the walk through of the U2 actual data....very well done & informative”
- Some Analysis General Results
  - Need to remove some redundancies & unnecessary aesthetic material
  - Need to make navigation more flexible & simplified
  - Question Format change designed for student clarity
  - Survey asks Students for too much detail

18

Thanks! Any Questions

19

C.4.5.2. Terry™ Turbopump Testing 2021 Update

Terry Turbopump Testing 2021 Update

Randy Bunt (Southern Nuclear)
Consortium Chairman
November 2021

BWR OWNERS' GROUP

BWR Expertise – Proven Solutions

Presentation Content

- Overview
- Milestones Update
- Bearing Oil Test, Correlation to Plant Injection and Self-Regulation Results

2

Terry Turbine Testing Overview

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

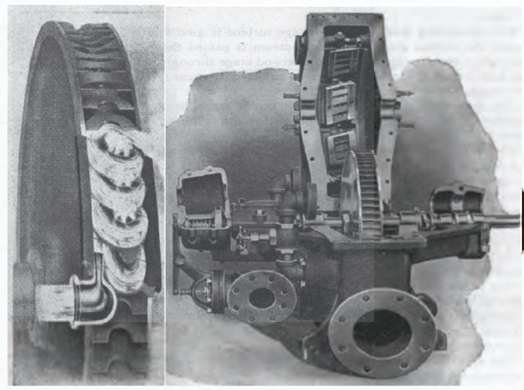


Image courtesy Sandia National Laboratories

3

Terry Turbine Testing Overview (Cont)

- The goal of the International Consortium is to provide long-term oversight of the Terry Turbine Expanded Operating Band Project (TTEXOB). The TTEXOB Project goal is to expand and define the actual operating limitations (margins) of the Terry turbine systems (i.e. RCIC/TDAFW) used in the nuclear industry.
- Milestones 1-7, Plan development, basic science, individual component testing, small scale testing, low pressure testing, self regulation simulation, closeout
- Membership for the project is based on the Turbo-TAG and as identified in the Program Plan (SAND2017-5562). Additional details on *Project* structure and participation are included in the Summary Project Plan (SAND2017-1725). The US Nuclear Industry, USDOE, and IAE (Japan), are the major stakeholders of the TTEXOB Committee and, as such, have leadership roles in the Turbo-TAG as well. Additional members of the *Consortium* would be identified and approved by the Turbo-TAG. The TTEXOB Project Manager will control the membership list and make changes as directed by the Turbo-TAG.
- The overall experimental program support (e.g. equipment, personnel, technical output) will be equitably shared between the major funding stakeholders (Japan, US DOE, and US Industry), but will vary based on milestone content. Cost sharing will vary based on directives of the major funding stakeholders.

4

Milestones – Update

- Milestone 1 – Complete (Plan Development)
- Milestone 2 – Complete (Basic Science)
- Milestone 3 – Complete (Individual Component Testing)
- Milestone 4 – Complete (Small Scale Testing)
- Milestone 5 – Stopped (Low Pressure Testing)
- Milestone 6 – Stopped (Self Regulation Simulation)
- M/S 5/6 Hybrid –Complete (Correlation to Plant Injection)
- Milestone 7 – Complete (Project Closeout)

5

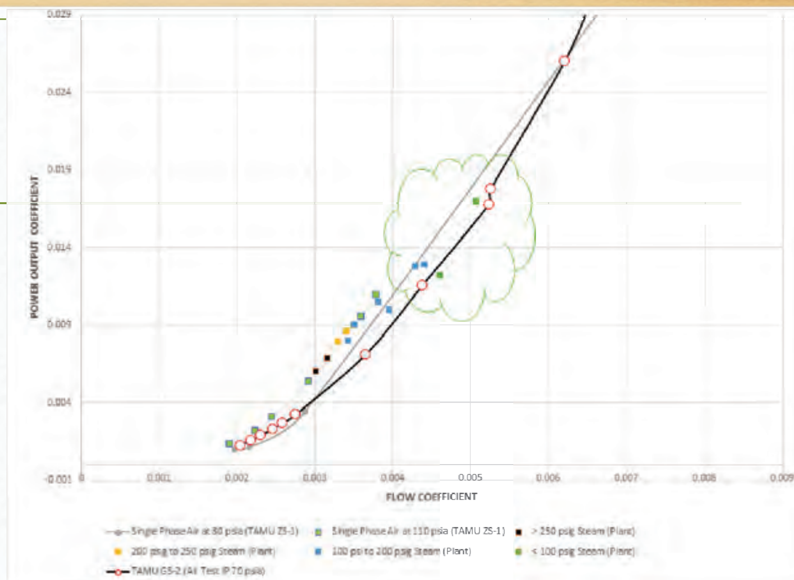
Hybrid 5/6: Brief Description

- Due to reduced funding across the Consortium entities (beginning in the Fall of 2019), there was a reduction in scope and deliverables, which eliminated the full-scale testing planned in Milestone 5 and 6 (MS-5/6). Because of this change in scope, the following gaps remain: full-scale steam test data, full-scale duration test with steam, self-regulation full-scale, and impact of steam quality.
- Milestone 5/6 Hybrid was designed to obtain confidence in scaling factor (MS5) and self-regulation (MS6). This was accomplished through testing at Texas A&M University (TAMU) utilizing previously designed testing stations and utility experience.
- Data was collected from the utilities in which GS Terry turbines were used under a variety of inlet steam pressures during injections for a significant time (~greater than four hours) and compared to steam testing of the ZS-1.
- EPRI Performed data analysis showing good correlation when an appropriate correction factor was applied. There is a specific area when additional data was needed to enhance the confidence of the correction factor.

6

Assessment of Plant data to TAMU Data

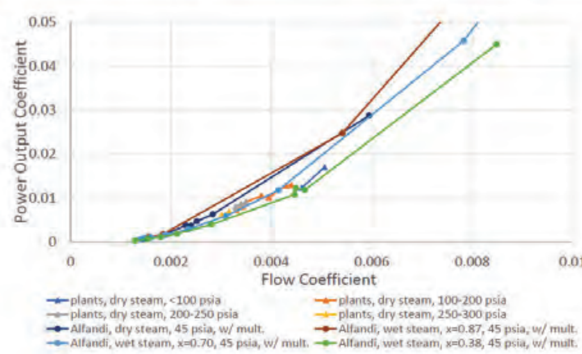
- Good correlation of test data and plant data when using a correction factor based on nozzle differences is applied to the TAMU test data.
- ZS-1 steam-water/air-water testing provided additional data correlation with plant data for the green cloud area.



7

Hybrid Milestone 5

TEXAS A&M ENGINEERING



Power output coefficient vs flow coefficient comparing plant RCIC data (steam) and TAMU ZS-1 data (wet steam) at 45-psia inlet pressure w/ scaling multiplier

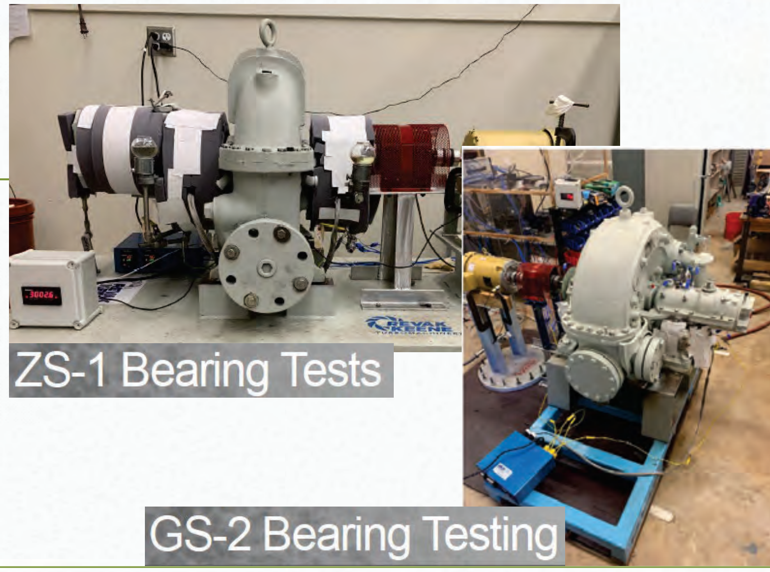
8

TAMU Bearing Testing (GS-2 bearings)

Objective: Develop a body of knowledge regarding the realistic performance of GS-2 Terry turbopump bearings with lubrication oil behavior under BDBE conditions.

- Both ZS-1 and GS-2 tests are in the test plan.

Testing status: Testing has been completed at Laboratory for Nuclear Heat Transfer Systems (Nuclear Engineering).



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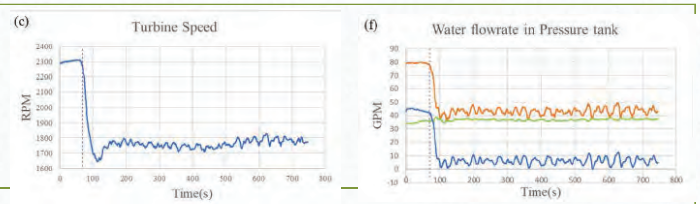
TEXAS A&M ENGINEERING

Self-Regulation Simulation at TAMU



- Turbine speed decays quickly and reaches a steady state value after water enters the air supply line.
- For all air pressures, the system reached similar steady state flowrates. The decay is linearly proportional to the initial flowrate.

10

Questions and Comments

11

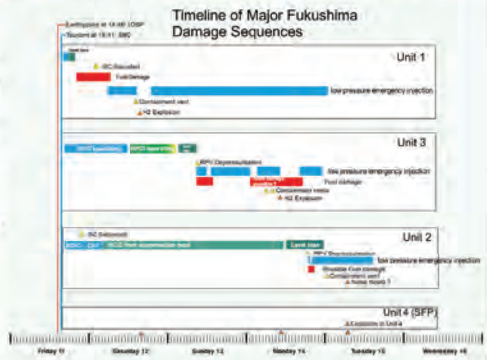
Reference Material

12

Basic Science Modeling

Unexpected RCIC Behavior in Fukushima Daiichi Unit 2

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



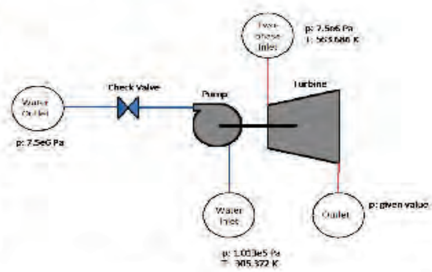
(Credit of R. Gauntt, et al., SAND2012-6173,

13

Basic Science Modeling

Two-Phase RCIC System Test Model

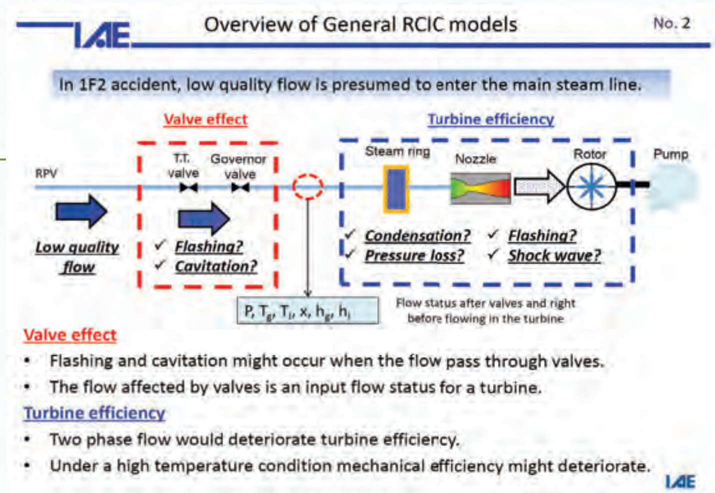
- The goal is to create a Fukushima-accidents-like scenario, with a reasonable self-regulating behavior appearing in the simulation.
- Periodic two-phase condition given at the turbine inlet.
- The turbine inlet pressure is 7.5 MPa and the turbine outlet pressure is 193 kPa.
- The simulation was run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



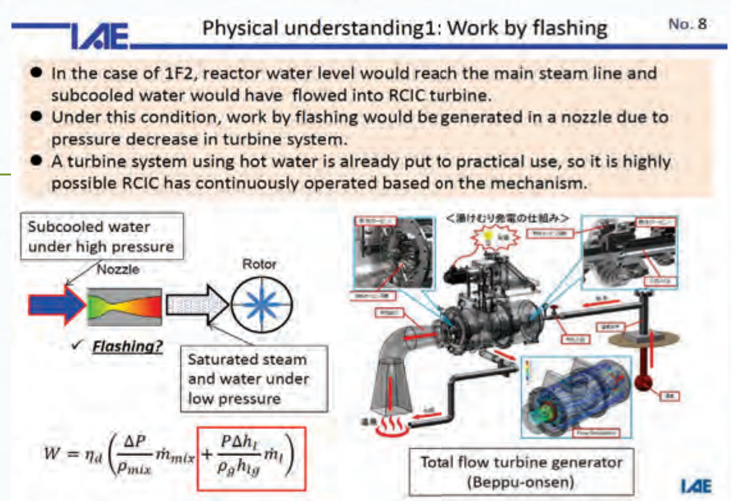
Terry turbine RCIC system test model

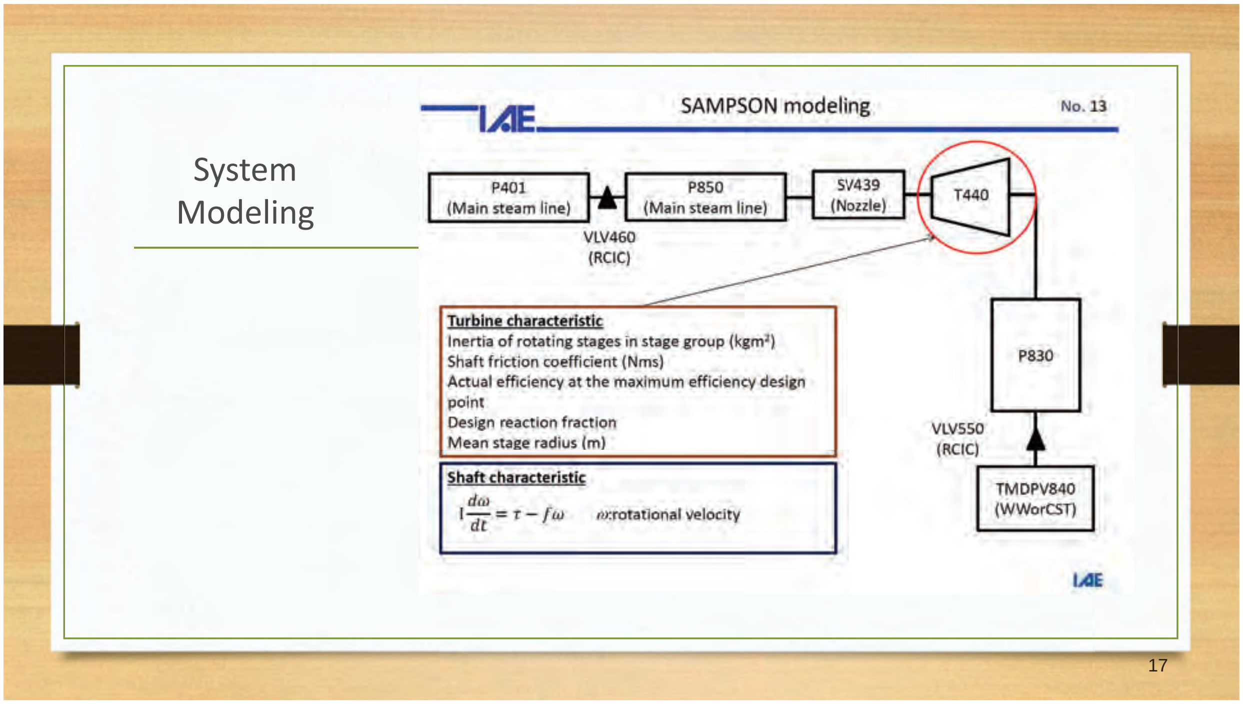
14

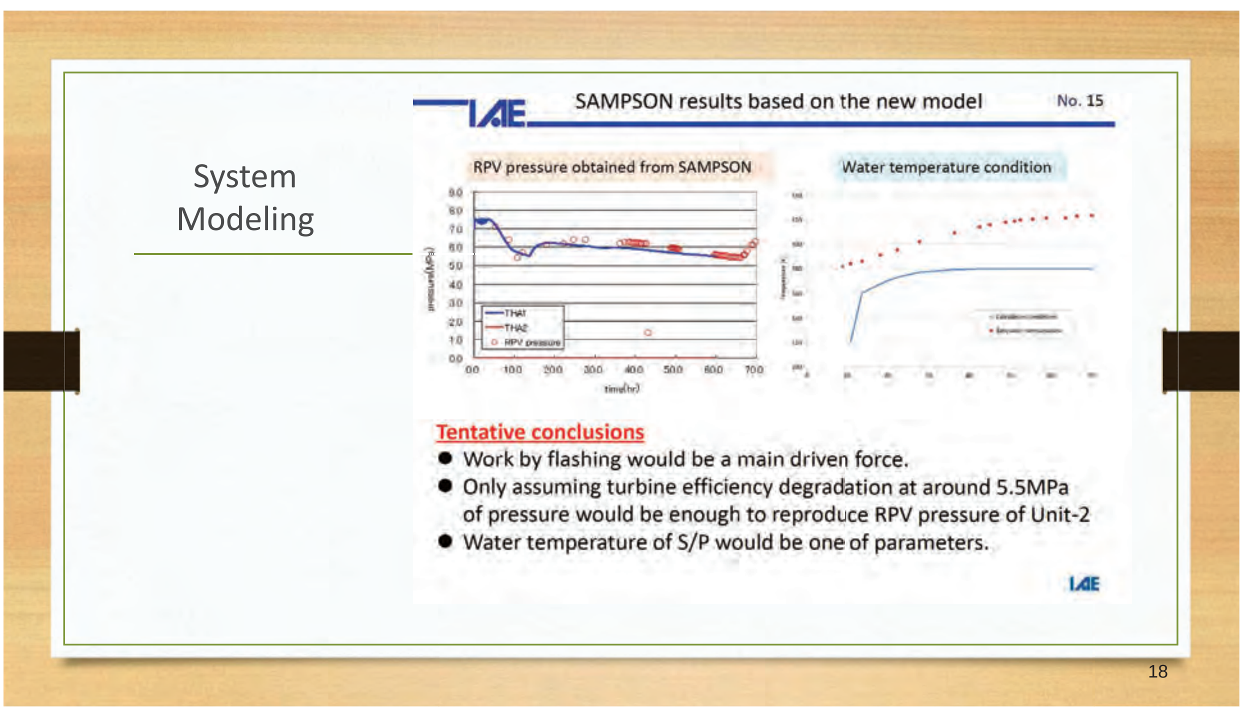
System Modeling

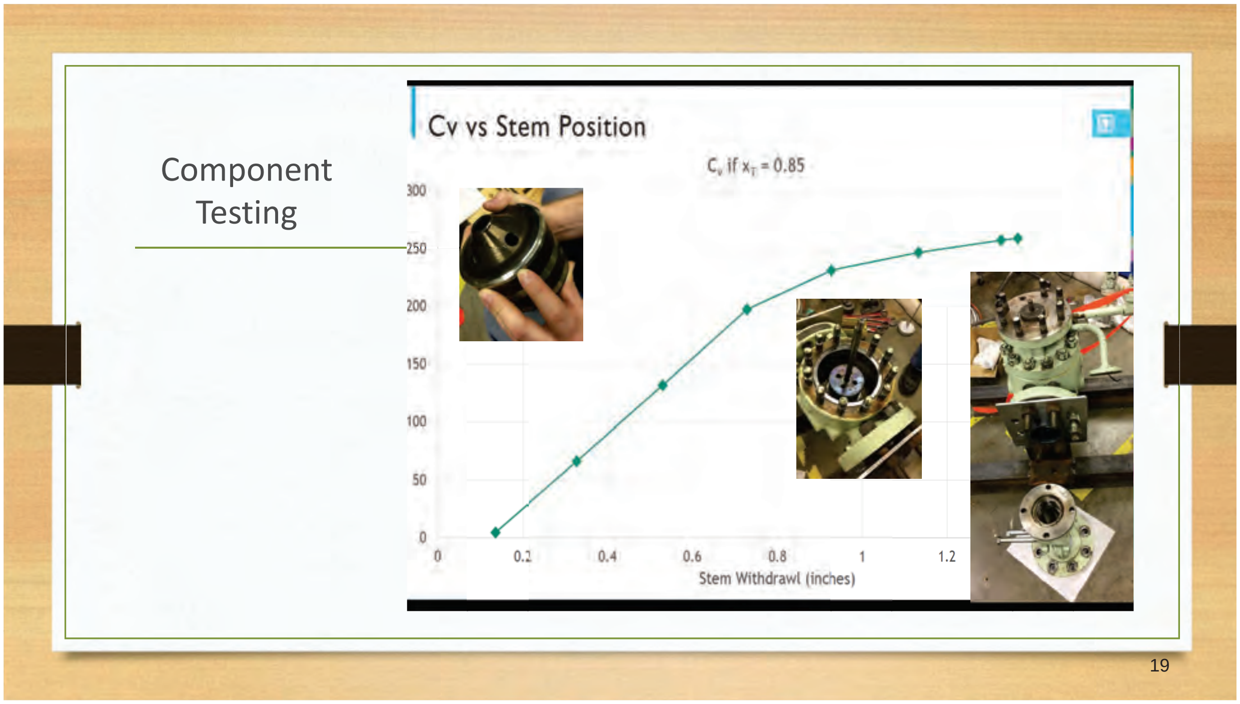


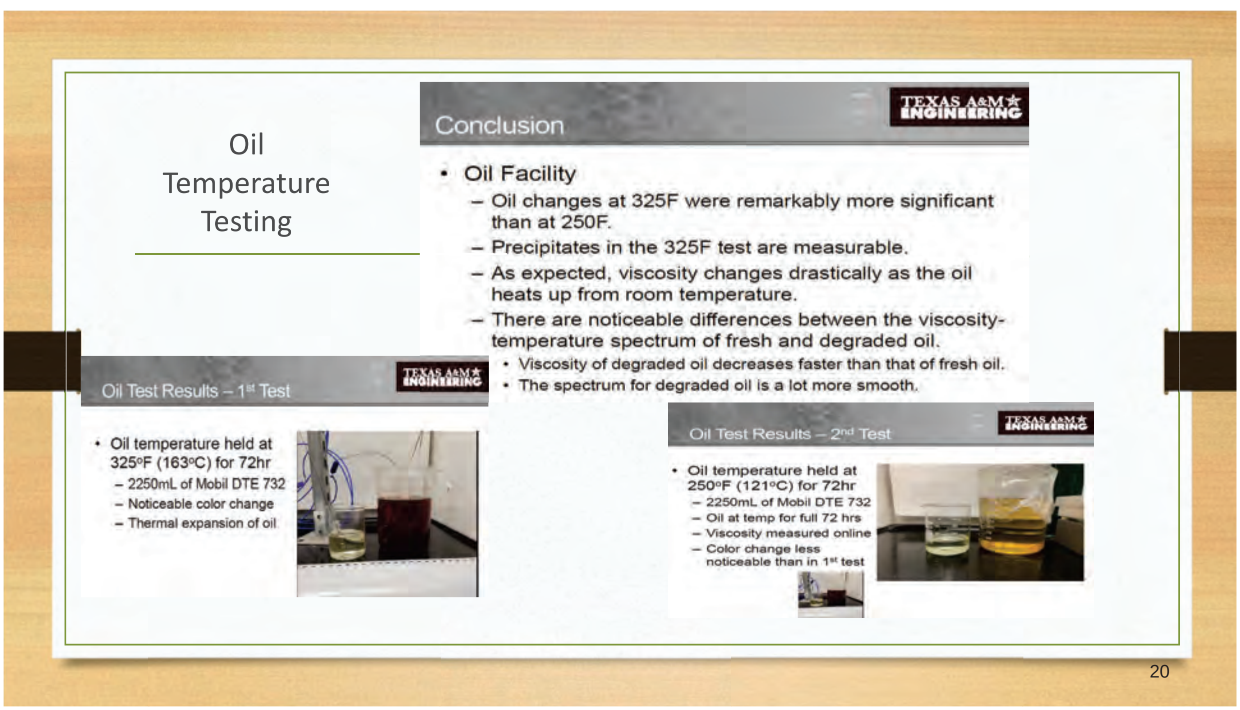
System Modeling

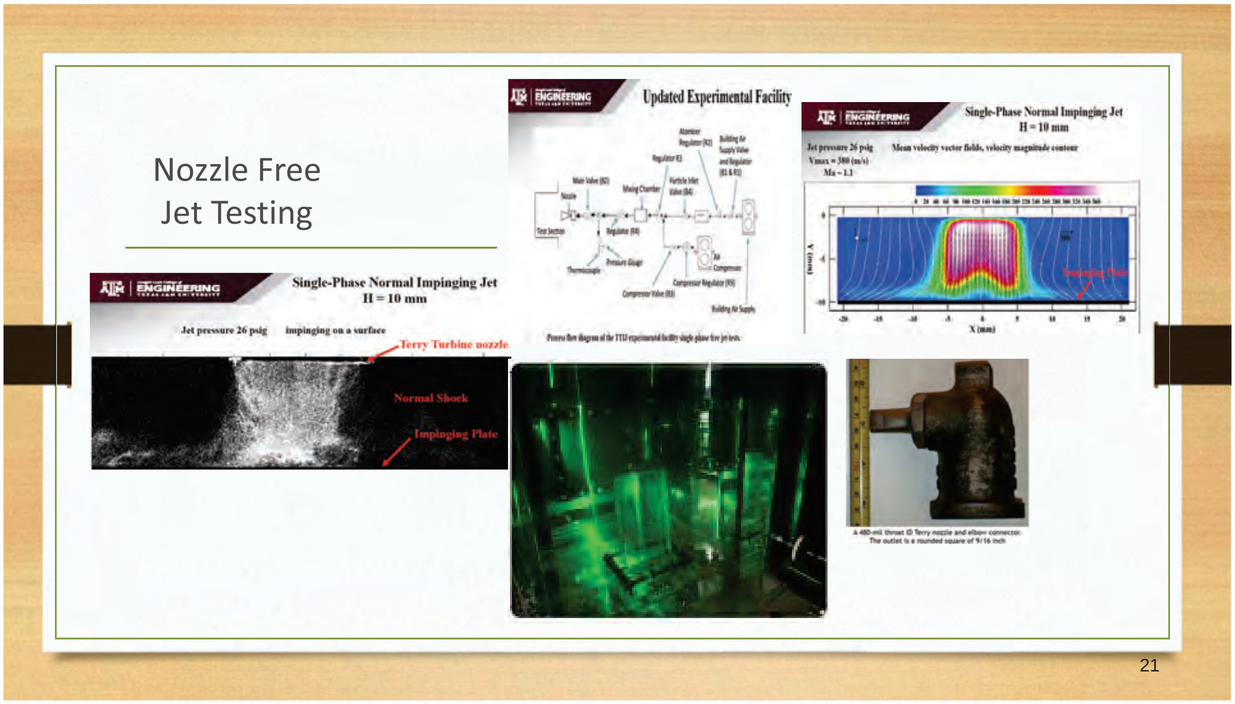



System Modeling
IAE
SAMPSON modeling
No. 13
P401
(Main steam line)
P850
(Main steam line)
SV439
(Nozzle)
T440
VLV460
(RCIC)
Turbine characteristic
Inertia of rotating stages in stage group (kgm²)
Shaft friction coefficient (Nms)
Actual efficiency at the maximum efficiency design point
Design reaction fraction
Mean stage radius (m)
P830
VLV550
(RCIC)
Shaft characteristic
$I\frac{d\omega}{dt} = \tau - f\omega$
ω:rotational velocity
TMDPV840
(WWorCST)
IAE
17

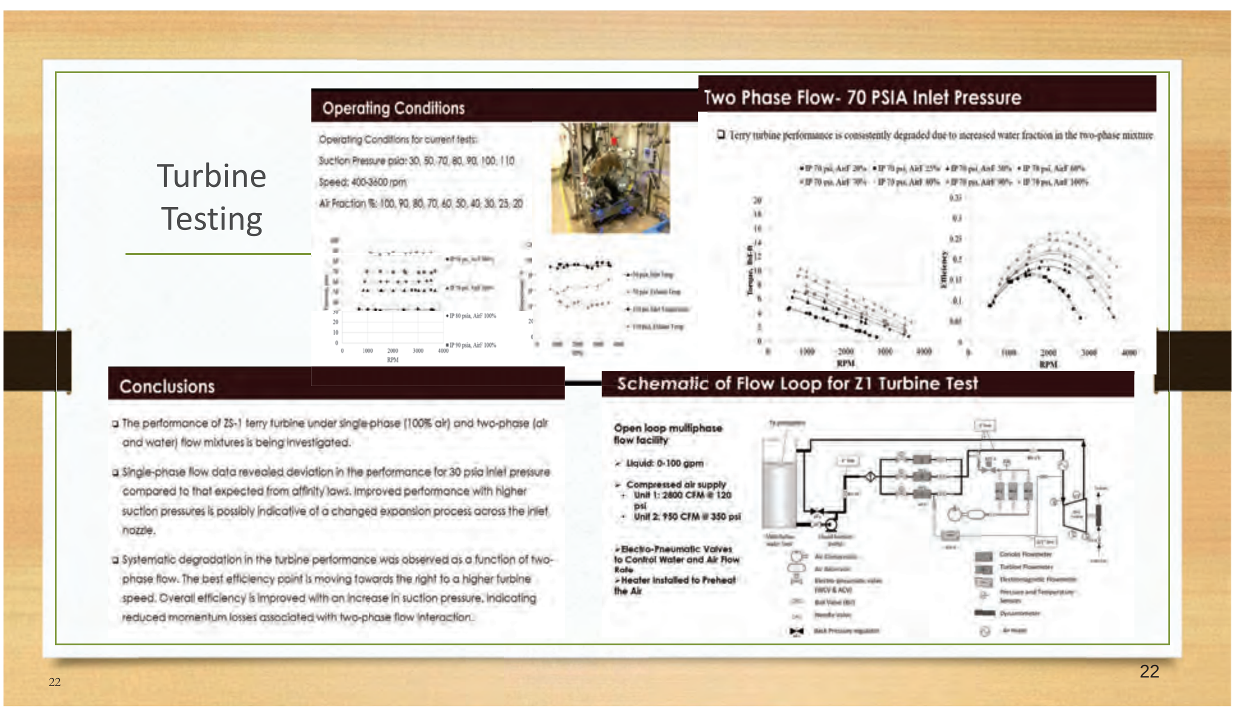

System Modeling
IAE
SAMPSON results based on the new model
No. 15
RPV pressure obtained from SAMPSON
Water temperature condition
THA1
THA2
RPV pressure
time(hr)
Tentative conclusions
● Work by flashing would be a main driven force.
● Only assuming turbine efficiency degradation at around 5.5MPa of pressure would be enough to reproduce RPV pressure of Unit-2
● Water temperature of S/P would be one of parameters.
IAE
18


Component Testing
Cv vs Stem Position
Cv if xT = 0.85
300
250
200
150
100
50
0
0
0.2
0.4
0.6
0.8
1
1.2
Stem Withdrawl (inches)
19


Oil Temperature Testing
Conclusion
TEXAS A&M ENGINEERING
• Oil Facility
– Oil changes at 325F were remarkably more significant than at 250F.
– Precipitates in the 325F test are measurable.
– As expected, viscosity changes drastically as the oil heats up from room temperature.
– There are noticeable differences between the viscosity-temperature spectrum of fresh and degraded oil.
• Viscosity of degraded oil decreases faster than that of fresh oil.
• The spectrum for degraded oil is a lot more smooth.
Oil Test Results – 1st Test
• Oil temperature held at 325°F (163°C) for 72hr
– 2250mL of Mobil DTE 732
– Noticeable color change
– Thermal expansion of oil
Oil Test Results – 2nd Test
• Oil temperature held at 250°F (121°C) for 72hr
– 2250mL of Mobil DTE 732
– Oil at temp for full 72 hrs
– Viscosity measured online
– Color change less noticeable than in 1st test
20

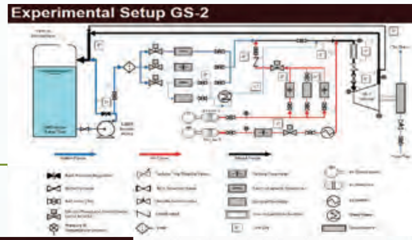


Nozzle Free
Jet Testing
Updated Experimental Facility
Single-Phase Normal Impinging Jet
H = 10 mm
Jet pressure 26 psig
Mean velocity vector fields, velocity magnitude contour
Jet pressure 26 psig impinging on a surface
Terry Turbine nozzle
Normal Shock
Impinging Plate
21

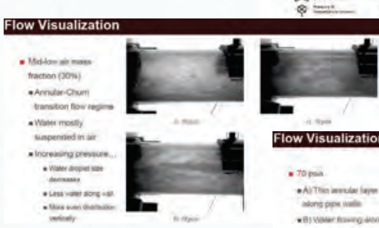


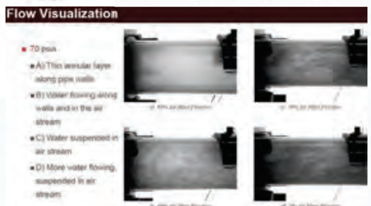
Turbine
Testing
Operating Conditions
Operating Conditions for current tests:
Suction Pressure psia: 30, 50, 70, 80, 90, 100, 110
Speed: 400-3600 rpm
Air Fraction %: 100, 90, 80, 70, 60, 50, 40, 30, 25, 20
Two Phase Flow- 70 PSIA Inlet Pressure
Terry turbine performance is consistently degraded due to increased water fraction in the two-phase mixture
Conclusions
The performance of ZS-1 terry turbine under single-phase (100% air) and two-phase (air and water) flow mixtures is being investigated.
Single-phase flow data revealed deviation in the performance for 30 psia inlet pressure compared to that expected from affinity laws. Improved performance with higher suction pressures is possibly indicative of a changed expansion process across the inlet nozzle.
Systematic degradation in the turbine performance was observed as a function of two-phase flow. The best efficiency point is moving towards the right to a higher turbine speed. Overall efficiency is improved with an increase in suction pressure, indicating reduced momentum losses associated with two-phase flow interaction.
Schematic of Flow Loop for Z1 Turbine Test
Open loop multiphase flow facility
Liquid: 0-100 gpm
Compressed air supply
Unit 1: 2800 CFM @ 120 psi
Unit 2: 950 CFM @ 350 psi
Electro-Pneumatic Valves to Control Water and Air Flow Rate
Heater Installed to Preheat the Air
22
22

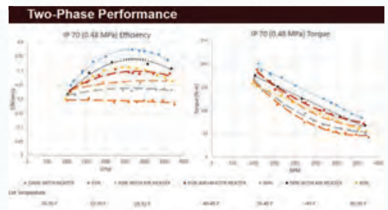
Turbine Testing



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







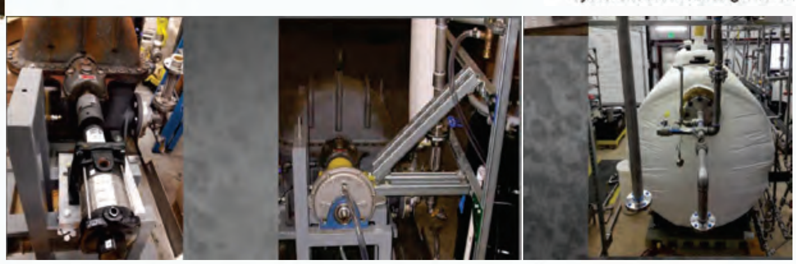
23

Turbine Testing

ZS-1 Turbopump Steam - Water Tests

TEXAS A&M ENGINEERING

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



24

C.4.5.3. Radiation Best Practices - Lessons Learned from Fukushima Daiichi D&D

U.S. DEPARTMENT OF ENERGY
Nuclear Energy

Radiation Best Practices – Lessons Learned from Fukushima Daiichi D&D

1

Background

U.S. DEPARTMENT OF ENERGY
Nuclear Energy

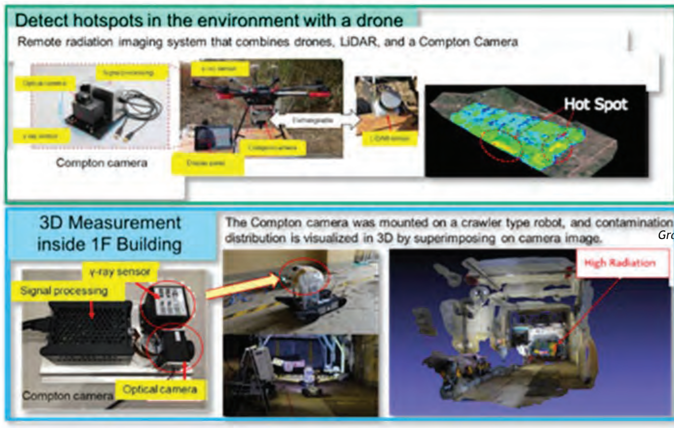
Progress made to learn about New D&D Technologies Deployed at Daiichi

- **Japan continues developing and deploying new technologies for surveying, characterization, stabilization, decontamination, and waste minimization**
- **BWROG emphasized that information about these technologies could benefit routine plant O&M activities, highlighting EPA report as an example, that could be produced to inform group regarding new technology effectiveness (efficiency, cost, waste generation, schedule, and safety)**
- **FY21 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 responded by developing draft document that:**
  - **Identifies new technologies/measures (with sample two-page brochures for selected technoologies)**
  - **Presents information to characterize effectiveness and areas where future research would be beneficial for routine O&M activities**

2

Gamma Cameras provide real-time radiation measurement and detection to facilitate D&D

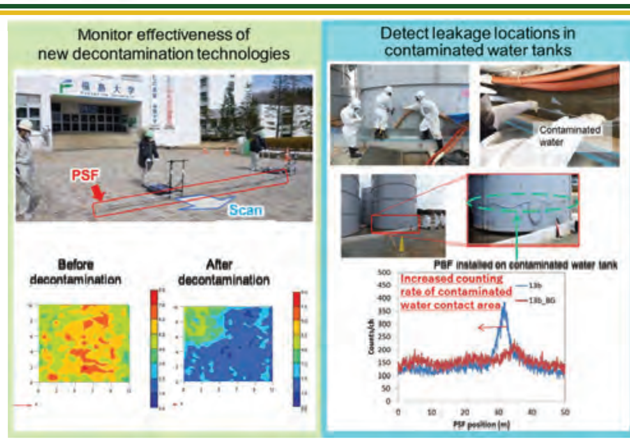


Graphics courtesy JAEA

- ***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***

3

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)***
- ***Simultaneous 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***

4

Several Steps to Increase Impact of New Daiichi D&D Technologies

- **Update letter document to consider information presented by Japan at FY22 meeting and include as Appendix to FY22 Report**
- **FY 22 Forensics Report should recommend:**
  - **Additional efforts needed to facilitate selected technologies becoming available for routine O&M activities**
  - **DOE NE funding opportunities (NEET, LWRS, etc.) should encourage collaborative bi-lateral R&D to deploy new technologies**
    - **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**
- **Resulting information needs to be in a format to be used by the plant operators, plant designers, accident model developers and researchers.**
    - **One format typically will not provide the information to feed all these stakeholders**
    - **Clear references to where other information is available will be a key to the report usefulness**

Graphic courtesy EPA

APPENDIX D. Topic Area 4 Supporting Information

This appendix provides supporting information related to selected topics discussed during the Topic Area 4 presentations.

D.1. Combustible Gas Generation from Cable Thermal Degradation within the 1F3 PCV

During the FY2022 Topic 4 discussions, the expert panel continued to be interested in reducing uncertainties associated with combustible generation during the 1F3 accident progression. In particular, there was interest in the possibility that significant amounts of combustible gases could be generated from thermal decomposition of cabling within the 1F3 PCV.

To gain insights on this topic, Dr. Luangdilok provided a scoping estimate of combustible gas generation from thermal decomposition of Hypalon® for Fermi Unit 2 (FERMI-2). FERMI-2 was selected because an estimated mass of cable insulation in this U.S. reactor (a BWR4 design housed in a Mark 1 containment) was readily available and bounded the estimated mass for 1F3. As discussed in this appendix, results from this scoping estimate indicate the total generation of combustible mostly-hydrocarbon gases, would be equivalent to less than 50 kg of hydrogen.

Several U.S. and Japanese experts (Drs. Sud Basu, Michael Corradini, Randall Gauntt, Chan Paik, Marty Plys, Michael Salay, and Shinya Mizokami) were invited to review a draft of this appendix. Comments by review panelists were considered in the version included in this report.

D.1.1. Approach and Scoping Values

As noted in NUREG-5950,[81] electrical cables have been modeled as a copper core with a jacket of Hypalon®* over insulation, which is typically 46.4 wt% Hypalon® and the remainder ethylene-propylene rubber (EPR). NUREG-5950 presents masses of EPR/Hypalon® cabling from several PWR and BWR Final Safety Analysis Reports (FSARs). This scoping analysis used NUREG-5950 values for the FERMI-2 BWR (a BWR4 design housed in a Mark 1 containment). Although similar in design to 1F3, FERMI-2 has a higher electrical power rating (1150 MWt versus 784 MWt). NUREG-5950 indicates that the total mass of EPR/Hypalon® cable insulation within the drywell and associated conduits is 2786 kg. Initial estimates from TEPCO (see [82]) indicate lower 1F3 cable masses, approximately 1050 kg.

Average values of pyrolysis yield data presented by Dr. Salay (see Slide 6 of C.4.4.2) were used to estimate the masses of combustible gases generated (in unit of grams per kilogram of Hypalon®). These average values were assumed as yield values (G-values) for this scoping effort.

The heat of combustion for individual gas products was used to convert the mass of various hydrocarbon gases into hydrogen equivalents (the amount of hydrogen required to produce the same amount of combustion heat). Values assumed for lower heating values (LHVs) in this scoping analysis were based on information in References [83], [84], and [85].

* Hypalon® is a registered trademark of DuPont for chlorosulfonated polyethylene rubber.

D.1.2. Results

Table D-1 presents scoping analysis results for FERMI-2. As shown in this table, scoping estimates indicate that the total generation of combustible (mostly-hydrocarbon) gases in FERMI-2 would be equivalent to less than 50 kg of hydrogen. Table D-2 compares this 50 kg of hydrogen with other in-vessel sources of hydrogen generation during the 1F3 accident progression estimated in Reference [75]. As indicated in Table D-2, if the amount of Hypalon® in the 1F3 drywell were assumed to be the same number as in FERMI-2, the amount of hydrogen equivalents generated by pyrolysis would only be around 2% of the total potential in-vessel hydrogen sources.

Table D-1. Estimate of Combustible Gas Generation for FERMI-2 from Pyrolysis of Hypalon® in the Drywell ($C_{85}H_{157}Cl_{13}SO_2$)

Pyrolysis gas product	Average G-values g/kg-Hypalon®	LHV MJ/kg	FERMI-2[a] kg	FERMI-2 H_2-equivalent kg
Carbon Monoxide (CO)	78.5	10.1	218.7	18.4
Methane (CH_4)	6.7	50.0	18.7	7.8
Ethane (C_2H_4)	2.2	47.2	6.0	2.4
Ethane (C_2H_4)	3.0	47.8	8.2	3.3
Propylene(C_3H_6)	1.7	45.8	4.7	1.8
Propane (C_3H_8)	1.6	46.4	4.3	1.7
Butene (C_4H_8)	0.8	45.3	2.3	0.9
Butane (C_4H_{10})	0.9	45.8	2.6	1.0
Pentane (C_5H_{12})	0.3	44.6	0.7	0.3
Cyclopentane (C_5H_{10})	0.1	44.6	0.3	0.1
Benzene (C_6H_6)	10.5	40.2	29.2	9.8
Toluene (C_7H_8)	1.0	40.6	2.7	0.9
Vinyl Chloride (C_2H_3Cl)	3.0	18.9	8.2	1.3
Hydrogen Chloride (HCl)	303.0	NA[b]	NA	NA
Carbon Dioxide (CO_2)	899.0	NA[c]	NA	NA
Total H_2-equivalent (kg)				49.5

a. Values include cabling in conduits.

b. Although not considered a combustible gas, HCl can react with metals and generate hydrogen gas.

c. Carbon dioxide is neither flammable or combustible.

Table D-2. Potential in-vessel and cable combustible gas generation sources during 1F3 accident progression.

Source	Potential Mass kg	Potential H_2-equivalent kg	Percent H_2 generation
Zirconium (Zr) in Fuel Cladding	29,000	1,272	42
Zirconium (Zr) in Channel Box	18,000	789	26
Iron (Fe) in Absorber	12,800	641	21
Boron Carbide (B4C) in Absorber	960	243	8
Cable Pyrolysis (assuming FERMI-2 values)	2,790	50	2
Total H_2-equivalent		2,994	100

D.1.3. Concluding Remarks

The NRAJ investigation of an unaccounted source of combustible gases from thermal decomposition of hydrocarbon materials in cable insulators and paint may provide the missing piece in the global mass balance between combustible gas generations and consumptions. If the magnitude of the unaccounted source of hydrocarbon gases is shown to be substantial, it would reduce the mass of combustible gas generation from MCCI needed. However, the preliminary scoping estimate presented in this appendix does not indicate that this would be the case.

APPENDIX E. Topic Area 5 Supporting Information

This appendix provides supporting information related to selected topics discussed during the Topic Area 5 presentations.

E.1. Advanced Engineering Training Initiative

The Advanced Engineering Training (AET) is a nuclear industry effort to develop technical computer-based training modules for targeted areas of engineering expertise. The modules are designed to maintain industry engineering expertise and facilitate engineering knowledge transfer. AET is funded and sponsored by U.S. nuclear utility engineering vice presidents and directors. In response to a request for additional information the AET initiative, Phil Amway provided input for this section of Appendix E.

In 2021, the AET staff determined that a module should be developed to support the FLEX Program. The overall objective of this training is to maintain the long-term FLEX Program sustainability and knowledge level so that the 1F lessons learned remain effectively implemented. The effort is supported by a Technical Consultant (Sargent and Lundy) and a Training Developer (InfoPro). A five member U.S. Industry Subject Matter Expert (SME) Working Group (see Table E-1) is guiding the development of this training with implementation planned for the 1st quarter of 2022.

Table E-1. SME working group members

Name (*Beta Test Volunteer)	Company
Phil Amway	Exelon - Project Manager and SME
Greg Bixby	Exelon - Training Rep
Randy Bunt	Southern Company - SME
Marco Ruvalcaba	South Texas Project - SME
Eric Schindelbeck	Dominion Energy - SME
Tracy StClair	Energy Harbor - SME

A two-tier approach is guiding development and maintenance of this AET module (see Figure E-1). Tier I candidates will complete Chapters 1, 2 and 3, and Tier II candidates will complete all 5 chapters and a final written exam. While considered AET, this training may be useful for anyone involved in or interfacing with the FLEX program, including plant operations, emergency planning and engineering staff. Training objectives (Table E-2) show the broad array of post 1F actions included (it is not limited to FLEX).

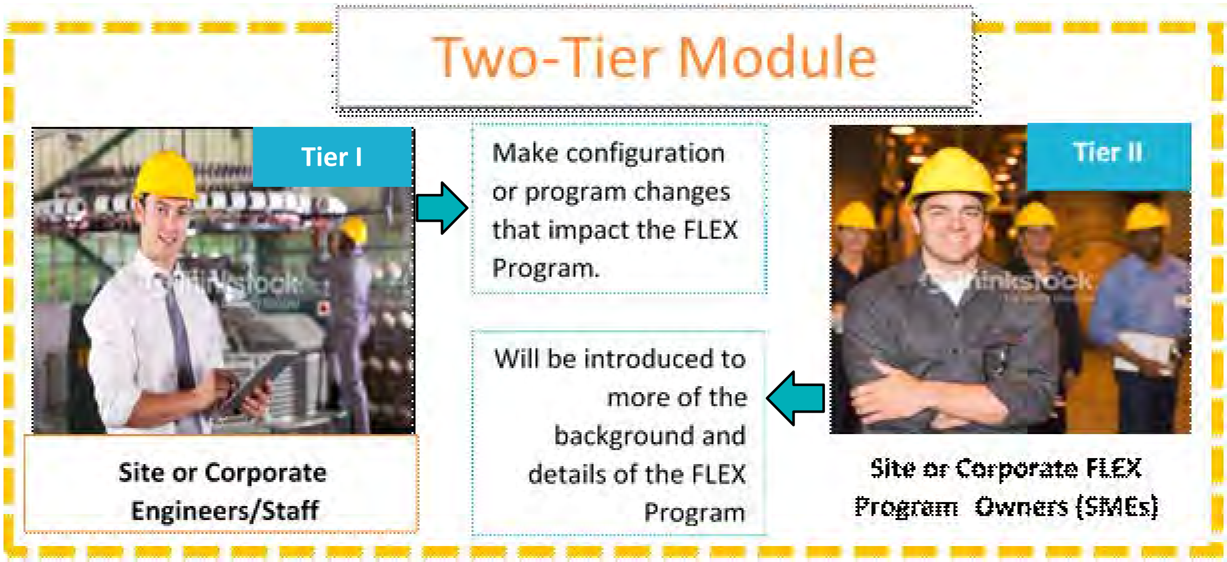


Figure E-1. Two-tier module approach [Image courtesy of Exelon Powerlabs, LLC]

Table E-2. FLEX Terminal Learning Objectives (TLOs) [Table courtesy of Exelon Powerlabs, LLC]

TLO Number	Chapter	Tier	Company
1	1	I,II	Describe an overview of the Fukushima Event including the key sequence of events, plant impact, e.g., extended station blackout, loss of key indications, high radiation levels, and lack of procedural guidance
2	1	I,II	Describe FLEX mitigation strategies as discussed in NEI 12-06 [86] and Reg. Guide 1.226 [87]
3	2	I,II	Describe FLEX support equipment and connections including deployment, testing, inspections/maintenance, design, and storage requirements as discussed in NEI 12-06
4	2	I,II	Describe strategic alliance for FLEX Emergency Response (SAFER) off-site response capabilities as discussed in NEI 12-06 and the SAFER equipment technical requirements document
5	2	I,II	Describe SFP Level (SFPL) instrument requirements as discussed in NEI 12-02 [88] and Reg. Guide 1.227 [89]
6	2	I,II	Describe Hardened Containment Vent System Requirements as described in NEI 13-02 [90] and NRC Order EA-13-109 [91]
7	3	I,II	Identify process and configuration changes that can impact the FLEX program as described in NEI 12-06
8	3	I,II	State the actions necessary when potential impacts to the FLEX program are identified including when FLEX support equipment functional requirements are not satisfied
9	3	I,II	State the intent of the N and N+1 structures and equipment staging, including use of FLEX equipment during outages to improve outage risk and preventative maintenance strategies to maximize FLEX equipment availability / minimize risk
10	3	I,II	Differentiate between Design and Beyond Design Basis Events
11	3	I,II	Identify applicable extreme external hazards as described in NEI 12-06
12	3	I,II	List site documents that comprise/implement the FLEX program as required by 10 CFR 50.155[92]
13	4	II	Recall the U.S. regulatory actions implemented in response to Fukushima from the Near Term Task Force (NTTF) recommendations and in Orders EA-12-049 [93] and EA-12-051 [94]

Table E-2. FLEX Terminal Learning Objectives (TLOs) [Table courtesy of Exelon Powerlabs, LLC]

TLO Number	Chapter	Tier	Company
14	4	II	Recall the U.S. nuclear industry (INPO) response to Fukushima as described in IER L1-11-02 [95] and IER L1-13-10 [96]
15	4	II	Explain special or related terms and definitions contained in NEI 12-06
16	4	II	Recall the assessment assumptions of NEI 12-01 [97] and the baseline assumptions in NEI 12-06.
17	5	II	Describe the treatment of design basis seismic and flooding hazards and seismic and flooding hazard re-evaluations, 10 CFR 50.54(f) letter and associated documents [98]
18	5	II	Recall common FLEX equipment failures from the EPRI database.
19	5	II	Describe the purpose and use of the EPRI FLEX collaboration website and EPRI database [99]
20	5	II	Identify INPO Industry Reporting and Information System (IRIS) requirements for FLEX issues, NEI EB 16-17 [100]
21	5	II	List external organizations that interface with FLEX and their roles such as Institute of Nuclear Power Operations (INPO)/World Association of Nuclear Operators (WANO), EPRI, NEI, and Pooled Inventory Management.

E.2. Duane Arnold Energy Center Event

On August 10, 2020, a derecho hit the 1912 MWt NextEra Energy Duane Arnold Energy Center (DAEC) in Iowa, damaging the plant's cooling towers and causing a loss of offsite power (LOOP) that lasted over 24 hours and significant damage to the local power grid.[101, 102, 103] The plant status on that date is summarized in Figure E-2. At the time of the event, DAEC was operating at ~80% power, coasting down for its end of cycle (EOC). A dry cask storage campaign was also under way in the spent fuel pool. In response to a request for additional information about operator response during the DAEC event, Phil Ellison provided input for this section of Appendix E.

A derecho is a widespread, long-lived wind storm associated with a band of rapidly moving showers or thunderstorms. Derechos can cause hurricane- or tornado-force winds, tornadoes, heavy rains, and flash floods. Although a derecho can produce destruction similar to the strength of tornadoes, the damage typically is directed in one direction along a relatively straight swath. At DAEC, wind speeds exceeded 100 mph with onsite peaks between 100 and 130 mph. At 12:49 on August 10, a LOOP occurred due to sustained strong winds in excess of 100 mph.

As a result of the LOOP, the 'A' and 'B' emergency diesel generators automatically started and supplied power to the safety related busses. As expected, the reactor water level initially lowered rapidly to Level 2 (lo-lo) due to the loss of feedwater (see Figure E-3 for DAEC RPV water levels). RCIC and HPCI systems automatically initiated and restored the reactor water level until it reached a high water level (Level 8) trip. Both systems automatically tripped, per design. The operators placed both systems in manual control and intentionally increased water levels to promote natural circulation cooling. By operating the RCIC system and controlling level high, operators were able to maintain significant natural circulation which helped with cooldown and maintaining the temperature differential across the RPV during the cooldown. Hence, several post-1F EOP changes simplified the plant's coast down to cold shutdown. Revised guidance for RCIC operation, which was informed by insights from Fukushima and TAMU testing (Section 2.4.5), allowed operators to quickly establish pressure control using the steam line drains and the RCIC system. In particular, revised Level 8 trip inhibits values, which prevented additional RCIC and

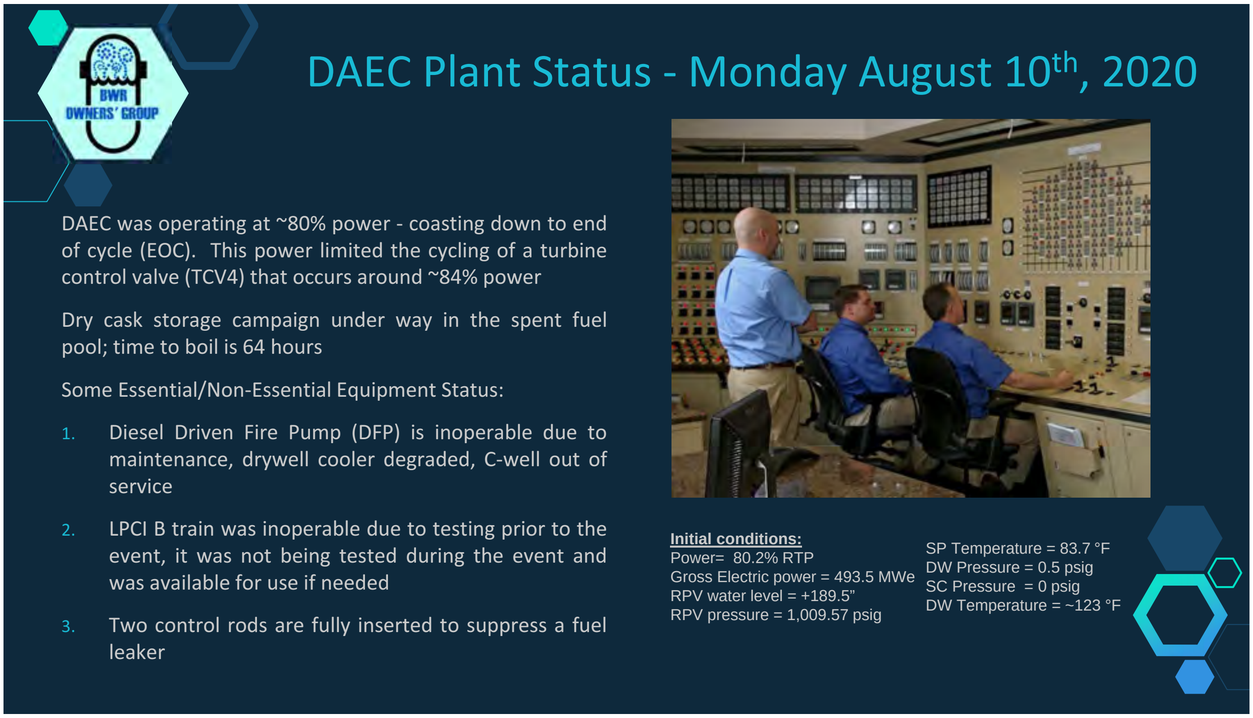


Figure E-2. DAEC plant status on Monday, August 10, 2020 (Courtesy of BWROG [101])

HPCI system trips, were important in maintaining the performance of these systems and in reducing SRV cycling.

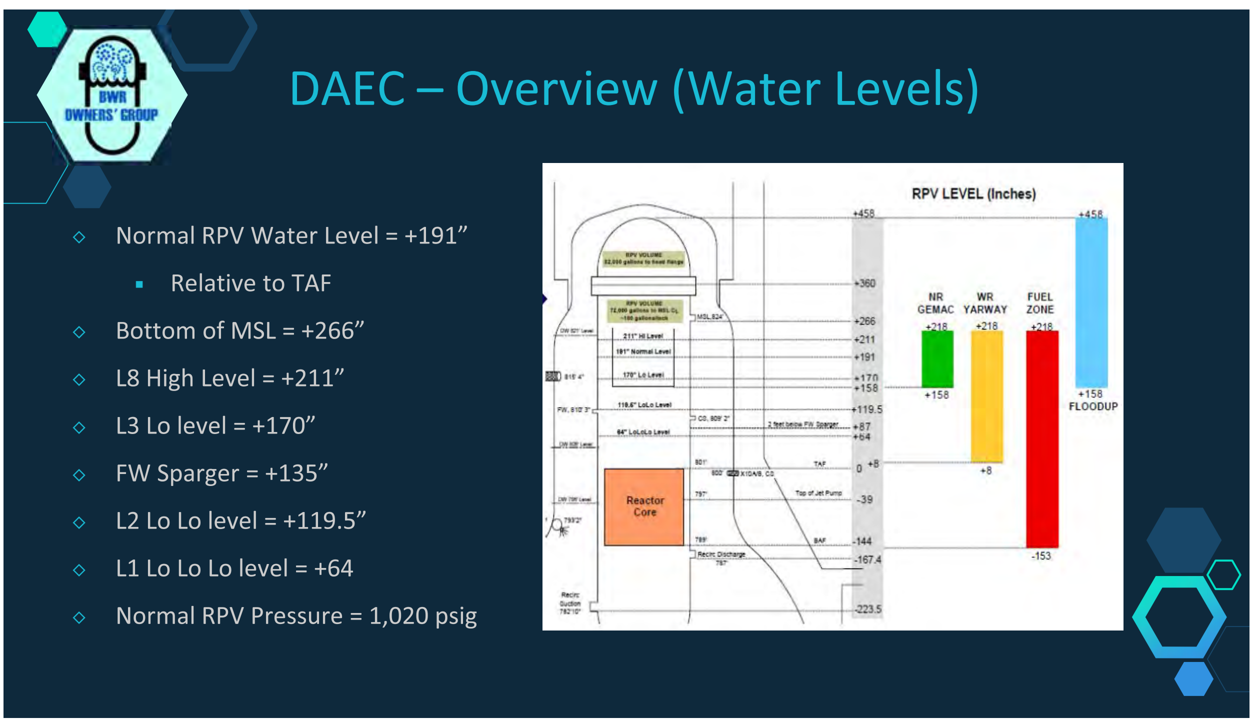


Figure E-3. DAEC RPV water level instrumentation (Courtesy of BWROG [101])

As documented in [101], evaluations of the DAEC derecho event led to several important insights:

- Post-1F EOP changes to high level trip inhibits were important in maintaining RCIC and HPIC system performance and in reducing SRV cycling
- The DAEC event re-emphasizes the need for symptom-based procedures in the Emergency Planning Guidelines (EPGs)/SAGs and FLEX
- Procedures and proficiency are important to restoring systems out of service for testing or maintenance and for returning failed systems to operation during a LOOP
- Event modeling assumptions need to be consistent with actual plant operations or conditions. RCIC testing provides specifics about the turbine and pump operation that improve modeling
- Plant transient response was as expected and agreed with simulator training for LOOP response

E.3. Radiation Best Practices - Lessons Learned from Fukushima Decontamination and Decommissioning

E.3.1. Introduction and Objective

Although fundamental principles* for reducing radiation exposure and contamination apply, the boundary conditions (e.g., types and magnitude of radioactive isotopes, land topography, and cleanup scale) for 1F D&D differ. Hence, the applicability of prior cleanup experience is limited. To address these limitations, Japan is developing and deploying new methods for decontamination and waste volume reduction. In response to a request for additional information about advanced technologies being used to facilitate 1F radiation characterization and cleanup (see Section 2.4.4), Joy Rempe provided input for this section of Appendix E.

The international community [107,108,109,110] is aware of the importance of information gained from on-going Japan D&D activities, not only for clean up after an accident but for other applications (e.g., routine plant maintenance, decommissioning, and military response to a radioactive or nuclear incident). Of particular interest to the U.S. nuclear enterprise are new measures for surveying, characterization, stabilization, decontamination, and waste minimization. This section of Appendix E identifies these measures and presents selected results demonstrating their effectiveness and areas where future research would be beneficial.

E.3.2. Overarching Fukushima-related D&D Strategies

Japan emphasizes a holistic risk management strategy to optimize Fukushima D&D. This strategy requires identifying and evaluating each decommissioning operation and stage, including removing and

*. Several references [104,105,106] outline fundamental principles for exposure reduction and methods for decontamination being applied at Daiichi. The external exposure reduction principles are: remove radioactive materials; maintain sufficient distance; install shielding; and reduce working time (e.g., use lock-ups). The internal exposure reduction principles are: wear personnel protection equipment (PPE); utilize equipment and materials to prevent dust dispersal; if injured, move to non-contamination areas; and outline (and contain) contamination zones (and required PPE for each zone). Decontamination can be completed using physical, chemical, and biological methods.

storing previously melted or damaged fuel rods from Units 1 through 4, maintaining cold shutdown in Units 5 and 6, and environmental cleanup outside the plant. In their comprehensive risk evaluation, NDF emphasizes two steps:[23]

- A risk reduction step that considers all Long-Term Management (LTM) activities based upon the level of risk presented by each type of radiation source. This step simultaneously considers different risk sources, identifying where higher risks occur and prioritizing LTM actions accordingly.
- A step that identifies and quantifies the risk from every on- and off-site source (see Figure E-4). LTM tasks fall into two main categories: fuel debris or waste-related activities.

LTM activities ensure three "critical" safety functions: maintain sub criticality, maintain cooling in the RPV and in the PCV, and control radiation release. Future activities, such as decontamination or construction, are planned considering possible dose reduction measures. To holistically manage exposure dose rate information for all plant workers, information is centrally managed. Worker dose allocation plans are developed; these plans equalize exposure and maintain exposures below the 20 mSv/year limit.

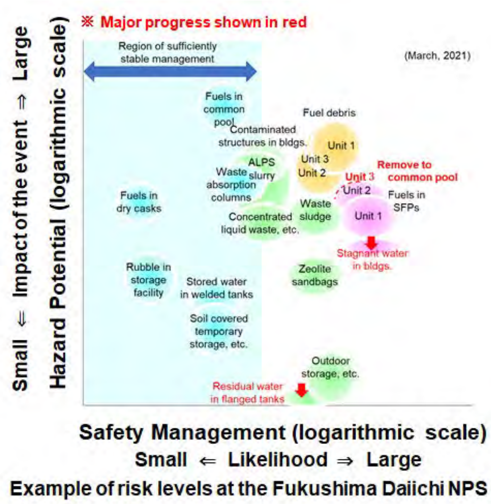


Figure E-4. Fukushima D&D emphasizes a holistic risk management approach characterizing the risk associated with each required operation and step.(Courtesy of NDF, [23])

D&D efforts also emphasize a step-by-step strategy in which new technologies are incrementally introduced. Efforts first focus on characterizing conditions, using existing technologies to initially address these conditions, and where needed, incrementally implement advanced technologies until recovery operations are completed. For example, considerable dose reductions (up to 80%) were initially achieved by increased use of shielding.[104] Incremental improvements are designed to reduce work exposure and radioactive waste generation. R&D for new technologies is focused on addressing problems identified with current approaches (e.g., structure damage due to high pressures, secondary waste production, and slow cleaning rates).

Several Decontamination Pilot Projects (DPPs) [111,112,113] were completed to obtain an initial basis for estimating decontamination efficiency and anticipated worker doses, identifying decontamination options, and estimating expected contamination levels and generated waste after performing such activities. DPP results were used to develop guidance for technology selection and developing best practices for obtaining air dose measurements and characterizing contamination. A catalogue is being maintained with comments regarding specific practices for various decontamination options. This catalogue identifies technologies/methods appropriate for different types of structures (e.g., residential, industrial), land types (e.g., residential, agricultural, parks, forest, grasslands), and roads (e.g., paved, unpaved, bridges).

The DPPs provided characterization data for evaluating the following:

- The availability and efficacy of proven and new techniques
- The cost, work period, workforce, waste generated, and radiation exposure to workers for each technique examined
- Management of resultant waste, including volume reduction, treatment of secondary waste, and temporary storage until centralized interim storage becomes available
- Worker safety, both conventional and radiation protection
- Radiation monitoring (before, during and after contamination)
- Public communication

The DPPs systematically compared decontamination options by evaluating their "technical effectiveness", considering factors such as efficiency, required resources (cost and labor), waste generation, schedule, and safety. Hence, evaluations considered not only the obtained dose reduction, waste generation and the required resources (time and labor), but also the potential for secondary contamination, the potential for damage to structures or surfaces, and the need for additional decontamination efforts. Result variability was attributed to factors, such as: (a) differences in the initial status, including aging deterioration (e.g., uneven surfaces and cracks); (b) differences in material composition (e.g., porosity, density); (c) variations in application techniques (e.g., blasting angle); and (d) the trial-and-error nature of the tests. Nevertheless, results (Table E-3) indicate that larger dose reductions occurred in residential land and farmland with higher initial contamination levels. In fact, results indicate the fractional reduction in dose rate of any target tended to depend on the original dose rate before decontamination and there was limited potential to further reduce dose at sites with the lowest contamination- probably because self-cleaning at these locations had already washed away any readily removed cesium.

Table E-3. Decontamination result summary [112]

Land use type	Dose rate range before decontamination (μSv/hr)	Number of measurement points	Average dose rate range before decontamination (μSv/hr)	Average dose rate range after decontamination (μSv/hr)	Dose rate reduction
Residential land	≥ 1.0	484	1.19	0.54	54%
	0.75-1.0	1235	0.83	0.31	40%
	0.50-0.75	2973	0.60	0.49	34%
	< 0.5	1772	0.40	0.40	23%
Farmland	≥ 1.0	119	1.11	0.74	34%
	0.75-1.0	708	0.83	0.59	29%
	0.50-0.75	1711	0.60	0.46	24%
	< 0.5	458	0.43	0.36	15%
Forest	≥ 1.0	680	1.17	0.80	31%
	0.75-1.0	1147	0.84	0.66	21%
	0.50-0.75	1814	0.62	0.53	15%
	< 0.5	338	0.43	0.40	7%
Roads	4	222	1.20	0.87	28%
	4	690	0.83	0.60	27%
	4	2021	0.60	0.44	26%
	4	1255	0.40	0.32	21%

E.3.3. Optimized Existing and New Novel New Technologies

Existing technologies are being optimized in 1F D&D activities, and novel new technologies are being developed and deployed to address problems identified in current D&D approaches. Table E-4 summarizes approaches of potential interest to the U.S. nuclear enterprise. As discussed within this section, new technologies (e.g., detectors, PPE, and software) are often combined to accomplish required tasks. Several references [104,105,106] provide quantitative values for the benefit of enhanced and new measures implemented in D&D.

Table E-4. Enhanced existing and novel new technologies applied in 1F-related D&D

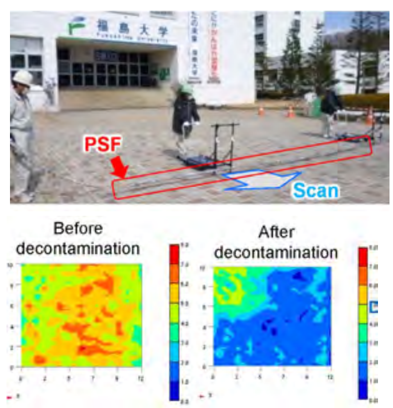
Technology	Features / Applications	
Optimized Personnel Protection Equipment (PPE) and Training with Centralized Exposure Optimization [13,108,104,105]	• Rezoned contamination areas and optimized PPE for each zone • Enhanced worker communication • Portable (Smartphone type) dose display • Mockups, special tools, 3D visualization, and close-proximity rest areas • Centralized data system for optimizing worker exposure	
Plastic Scintillation Fibers (PSFs) [114,115,116]	• Simultaneous detection of beta (β) and gamma (γ) radiation • Deployed on robots, drones, and unmanned aerial vehicles with optical cameras, dosimeters, and dust samplers • Real time monitoring with 2D or 3D radiation visualization (hot spot detection, D&D progression) • Leak detection of contaminated tanks (avoid sampling) • Facilitate public communication	
Wearable Global Navigation Satellite System (GNSS) [117, 118]	• Light-weight (1/5th original weight), tablet-embedded wearable system with GNSS • Enables one person to obtain radiation and position data • Data displayed as 2D contamination and dose rate maps • Monitor D&D progress	 

Table E-4. Enhanced existing and novel new technologies applied in 1F-related D&D

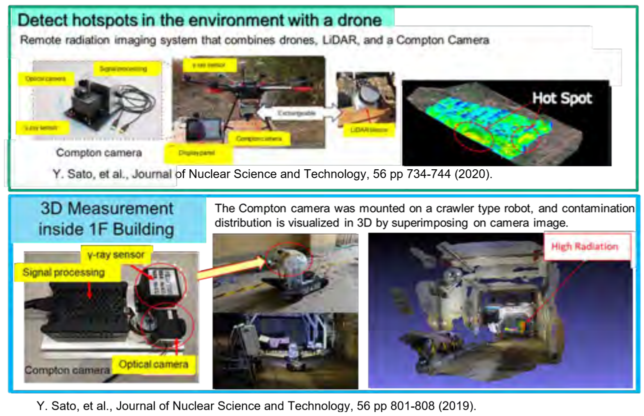
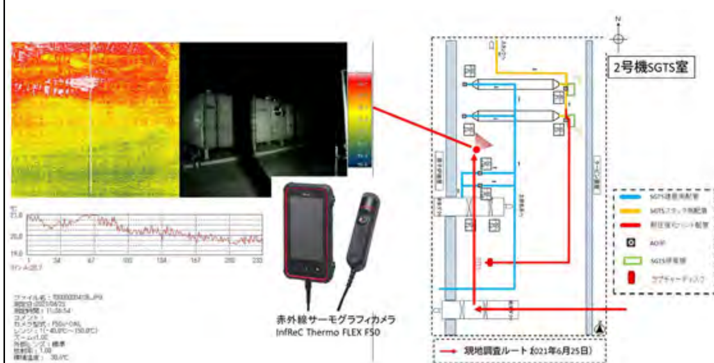
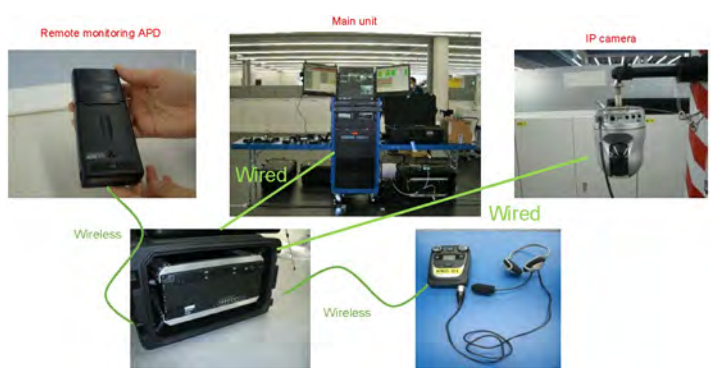
Technology	Features / Applications	
Gamma Cameras [112,115,116, 119, 120]	• Remote radiation imaging system combines Compton camera images with robot • Portable, can be mounted drones or robots with Light Detection And Ranging (LiDAR) to measure source • Allows 2D or 3D radiation visualization (hot spot detection) and D&D progress • Facilitates public acceptance of proposed D&D activities	
Infrared Thermography [121]	• Allows remote 2D temperature measurements • Portable hand-held device • Data stored and subsequently displayed as 2D images • Supported SGTS filter train investigations	
Remote Monitoring System [104,105]	• Continuous energy-efficient monitoring • Compact with easy to install semiconductor detectors and dosimeters, shielded (dust-proof, waterproof) IP cameras, and wireless communications • Results displayed in 2D contamination maps	

Table E-4. Enhanced existing and novel new technologies applied in 1F-related D&D

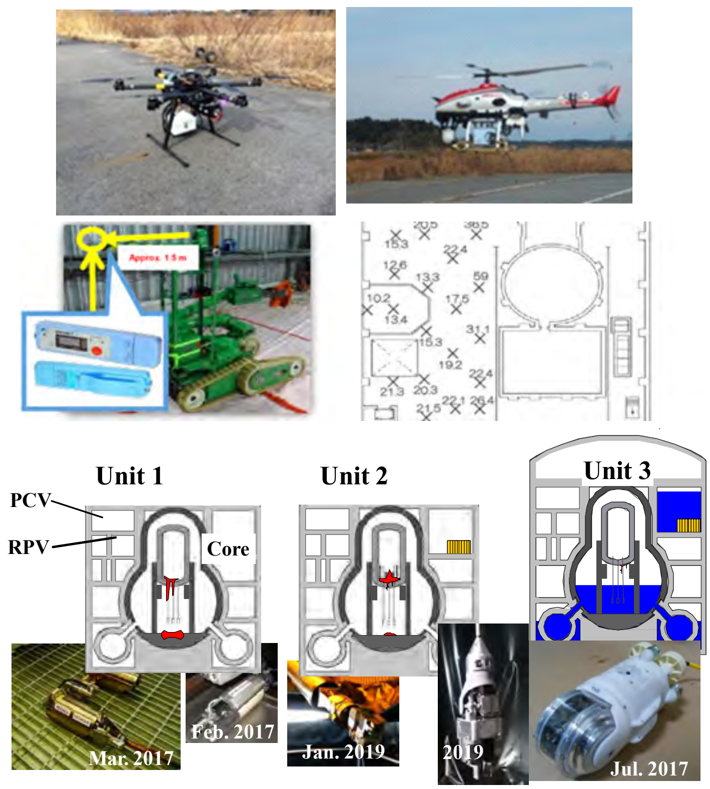
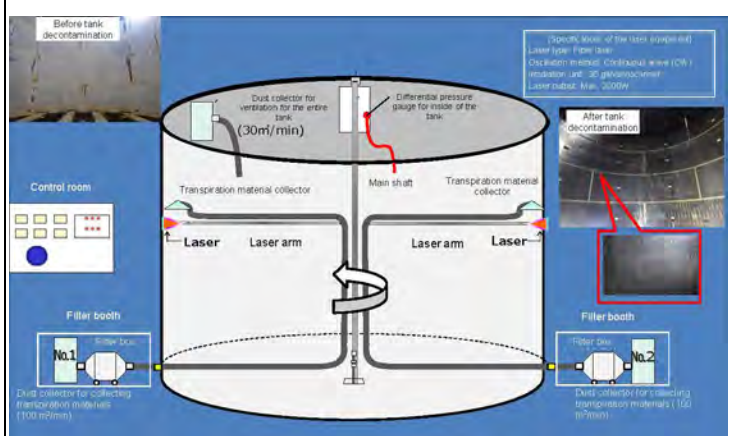
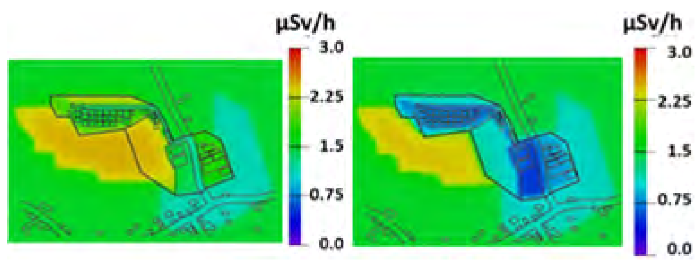
Technology	Features / Applications	
Robots, Drones, and Unmanned Aerial Vehicles (UAVs) [105,106,122]	• Robots, drones, and UAVs with detector systems (e.g., sodium iodide, plastic scintillators, etc.) and dust samplers • Can be controlled by a Global Positioning System (GPS) • Dose measurements obtained for range of heights and mission durations (beyond 60 minutes) • Collect data (in air and underwater), with and without leads (depending on application) • Includes several devices (high-definition cameras, thermocouple, radiation detectors [Cd-Zn-Te (CZT) semiconductor devices, gamma cameras] • Data transmitted and processed using Artificial Neural Network (ANN) to provide real-time 2D or 3D dose rate and contamination maps (with hot spot detection) • Create 180 and 360 3D images for remote damage monitoring	
Laser system decontamination to reduce β-ray exposure [105]	• After spray coating tank inner surface, removes contamination with enhanced laser system • System includes rotating, continuous wave fiber lasers with dust collector • Reduces worker exposure during disassembly of flange-type tanks for treated water storage	

Table E-4. Enhanced existing and novel new technologies applied in 1F-related D&D

Technology	Features / Applications	
Flexible Containers for Waste Storage and Transport [117,118,123]	• Durable, lightweight and weather-resistant, cloth-type flexible containers • Increase effectiveness (reduced cost, reduced radiation dose) for contaminated soil and material storage and transport.	
Integrated Dose and Contamination Assessment Software Tools - Calculation System for Decontamination Effect (CDE), N-Visage, PhotoScan, and Restoration Support System for Environment (RESET) [111, 112]	• Provides real-time 2D and 3D maps based on dose rate and contamination monitoring data • Can be used to estimate dose rate and contamination data after decontamination activities • Real time radiation dose measurements from installed meters integrated and displayed on screens available to the public • Facilitates public acceptance of proposed D&D activities	 

Type	Photograph	Characteristics
Flexible container (cloth-type)		• The assumption is that they will only be used once. • Not as good as the running-type in terms of weather resistance and waterproofness. • Some have improved weather resistance as a result of UV treatment and the like, while another type has improved waterproofness as a result of being lined with inner pouches and having an inner coating, etc.
Flexible container (running-type)		• The assumption is that they will be used by having soil repeatedly stored in and removed from them. • Outstanding weather resistance and waterproofness
Large sandbag		• Water permeable. • Some have improved weather resistance as a result of UV treatment and the like, while another type has improved waterproofness as a result of being lined with inner pouches, etc.

- ***Optimized Personnel Protection Equipment (PPE) and Training for Various Contamination Levels*** - PPE weight and design can adversely affect worker performance, increasing the time in radiation areas and worker dose. At Fukushima, the weight and design of PPE limited line-of-sight and caused rapid heat fatigue, nausea, and intense headaches. Hence, PPE made personnel less capable and limited the amount of time it could be worn. To address this issue, TEPCO reorganized on-site zoning of radiation-controlled areas to optimize required PPE, minimizing the use of unnecessary protection where possible. In addition, TEPCO established new co-located rest areas and enhanced training for donning PPE. TEPCO is also utilizing mockups, pre-assembly options, specialized tools, 3D visualization of proposed tasks, and temporary shielding to reduce exposure. An integrated dosimeter, dose display, and communication device (smartphone type) has been provided for workers to wear inside their PPE (data from workers are fed to the on-site centralized Remote Monitoring System (RMS). Dosimetry data are centrally managed to develop plans that equalize worker exposure and maintain exposures below the 20 mSv/year limit
- ***Plastic Scintillation Fibers (PSFs)*** - JAEA has demonstrated that PSF detectors work as well or better than conventional detectors. PSFs allow rapid measurement of radiation profiles, making it possible to

quickly see the difference between an area that has gone through a remediation effort already and a neighboring area that has not. PSFs are also being used to provide real-time detection of contaminated water leakage. Because of the possibility for contaminated water leakage into drainage channels, sampling of channel water in those channels was conducted to measure beta (β)-rays emitted from strontium 90, which is often high in contaminated water. To ensure immediate response and high efficiency in radiation monitoring, there was a need for the development of a more convenient, novel real-time monitoring technology that allows for distinguishing β rays, which are difficult to directly measure because their small travel distance (i.e., their range) in water is too small, from γ rays, which originated from background radiation (due to prior radioactive fallout accumulated on the ground). JAEA and other institutes succeeded in developing a fiber-type monitor that provides real-time distinguishing between β and γ radiation (by comparing differences in results of detectors with and without stainless steel shielding). The achievement of real-time monitoring of β radiation avoids on-site sampling and analysis of drainage channels; furthermore, PSF deployment accelerates detection of contaminated water leakage and reduces radiation worker exposure.

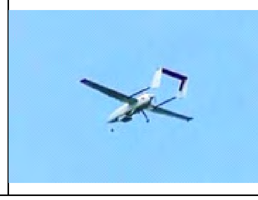
- ***Wearable Global Navigation Satellite System (GNSS)*** - Radiation measurement positions must be known. Although accurate, conventional position measuring GNSSs are heavy and bulky. Typically, they require two persons to operate: one person reads the measurements, while a second person records the data on paper or a tablet device. Such workflow is inefficient and susceptible to human error. A tablet-embedded global positioning system (GPS) is light and easy to handle, but it has poor accuracy and requires long time periods to locate measurement points. To address these issues, a wearable GNSS system has been deployed. The system had several advantages: (i) only one worker is needed for measuring and recording data; the equipment weight was reduced to 1/5th of a conventional apparatus; and measured data are sent by a wireless transmitter to a recording unit; (ii) work efficiency was improved more than ten-fold by having a high-speed navigation system, similar to a car-navigation system; and (iii) the progress of decontamination work can be visualized using measured data and the geographic information system (GIS).
- ***Gamma Cameras*** - Portable gamma ray imaging systems (with Compton cameras) are another new technology pioneered in Fukushima D&D. It is possible to visualize radiation data by color-coding radiation levels in images. Simultaneous Localization Mapping (SLAM) software combines data from gamma cameras and high permeability laser data to create 2D- or 3D-visual images of contamination that identify the location of hot spots. Obtained images are used to confirm the effectiveness of decontamination and the safety of temporary storage sites. Images are also used to communicate risk to workers and the public.
- ***Infrared Thermography*** - During plant examinations, NRAJ staff used portable hand-held devices to remotely obtain 2D temperature measurements. Data are stored and displayed as 2D images. Measurements supported NRA investigations of the standby gas treatment system filter train investigations.
- ***Remote Monitoring System (RMS)*** - Radiation workers that take measurements, and supervisors that manage the working environment (for removing rubble and other waste, installing communication and power equipment to enable future remote operation, etc.) account for 10% of the workers with the highest radiation exposure. To reduce worker exposure, a lightweight compact RMS was developed to continuously measure dose rate and contamination levels. The RMS consists of shielded (dust-proof and waterproof) IP cameras, a headset, and a remote monitoring Alpha Particle Detector (APD) with shielded wireless air monitors that are easy to install and remove. Data are also collected from lightweight personal dosimeters provided to workers. RMS data are transmitted to software with 2D and 3D display capabilities to develop easy-to-understand visualization of radiation contamination and dose maps. In some case, the RMSs have been demonstrated to reduce the length of time and radiation exposure of workers in radiation areas by over an order of magnitude.

- ***Robots and Unmanned Aerial Vehicles (UAVs)*** - TEPCO Holdings has emphasized the use of remote-controlled robots, drones, and UAVs in D&D activities. For example, remote-controlled coating equipment was used to deposit synthetic plastic emulation and encapsulants within buildings and on the site. Remotely operated vehicles (ROVs), such as heavy equipment and robots using portable shields when needed, were successfully used to remove the 1F1/1F2 exhaust stack.[105] Radiation measurements were obtained using robots and drones with the ability to communicate to computers with software that can provide real time 2D and 3D maps. However, robots are frequently impaired by issues, such as mobility limitations (due to obstructions and tangled cords), electronics and camera degradation due to radiation, mission time limitations due to inadequate power sources, transmission failures due to building blockages or communication network degradation, harsh environmental conditions (radiation, darkness, heat, flooding, etc.), and inadequate strength, dexterity, or mobility. It is desirable to continue research to provide robots with more dexterity, autonomy, and endurance, allowing them to accomplish additional decontamination tasks. For example, to overcome power limitations, Reference [124] suggests exploring the use of "energy scavenging" options, in which a robot collects energy from its environment (e.g., solar, wind, or possibly radiation) and transforms it into electricity.

  Shortly after the 1F accidents, aerial surveys were conducted to enable response decision-making. Such surveys typically used a high efficiency scintillator (e.g., thalium activated sodium iodide) or, occasionally, a high energy resolution semiconductor detector (e.g., cadmium tellurium), positioned within (or held below) the aircraft. The gamma spectrum was logged for a set time and linked to a GPS system, while the aircraft flew at constant speed and height (to the extent possible). Aerial survey methods were complemented by manual field and direct contamination measurements. Because gamma dose point measurements give better defined data than aerial survey methods, several assumptions are needed to convert these into radionuclide contamination levels. For example, radiation levels can vary appreciably over a few meters, due to topography, biological concentration, and other environmental factors. In addition, radiation levels also change dynamically over time, as radioactive particles migrate and decay. Furthermore, air conditions (wind, precipitation, and particulate matter) can interfere with radiation characterization. Moreover, aerial surveys and measurements from other methods are resource-intensive and costly; simpler, less-expensive methods were needed to characterize contamination levels.

  JAEA is addressing this need using an UAV system, which includes drones, helicopters, and airplanes, that can be used in areas that are impossible to reach by car. For example, UAV helicopters are being deployed with detectors (sodium iodide and plastic scintillators) and dust samplers. The helicopter, which is radio- and GPS-controlled, performs scans at a height of 20-80 m for mission times of about 90 minutes. Data are transmitted and displayed in real-time contamination and dose rate maps. Because of the range of techniques (Table E-5), inter-calibration and establishment of reference measurement sites are important to assure data can be integrated into a common database. Improved accuracy requires accounting for topographic features and complex terrains (e.g., mountains, forests, etc.) in survey areas. To improve the accuracy of data collected from different types of detectors and for topographical features in survey areas, Artificial Neural Networks (ANNs) are used.[125] JAEA has also developed a system that allows the public to view air dose rates (in real time) within several highly populated residential areas in the Fukushima prefecture.

Table E-5. Characteristics of different airborne survey systems (courtesy of JAEA, [113])

Survey area	Regional > 1000 km	Semi-regional < 1000 km	Local > 1 km	Small < 1 km
Option	Manned helicopter	Unmanned airplane	Unmanned helicopter	Micro UAV
Altitude	~ 300 m	~ 150 m	~ 50 m	< 10 m
Features and Availability	Standardized methodology available for efficient regional surveys	Allows remote controlled long-time flight (e.g., 6 hrs); under development	Higher resolution mapping available	Allow focused surveys, e.g., above urban areas or in forests; under development
Illustrations				

- ***Laser Decontamination to reduce b-ray Exposure*** - When cutting/disposing of the side and bottom plates or sheets from flange-type tanks, measures to shield against high-energy β rays (2.27 MeV) are implemented to prevent sudden increases in eye lens and skin exposures from deposits adhered to tank surfaces. Tanks are first spray-coated to minimize dust generation. Then, a system, which included two lasers attached to a rotating arm and a dust collector, is deployed. Workers also wear face guards to reduce exposure to their skin and eyes.
- ***Flexible Containers for Waste Storage and Transport*** - Flexible containers, large sandbags or "flex-con bags" of about 1 m capacity (1.1 m in diameter by about 1.1 m in height) are mainly used as shorter-term storage containers. Durable materials are used for long-term storage (a few years) and for storing removed soil containing water. Examples of flexcon bags, which increase the effectiveness of transportation and storage (reducing cost, radiation exposure, etc.), include cloth-type containers (for one-time use) with a weather-resistant inner bag, repeated-use containers, and large weather-resistant sandbags with an inner bag. Although all are light-weight, their weather-resistance, water permeability, durability, and expected lifetime varies.
- ***Integrated Dose and Contamination Assessment Software Tools*** - To facilitate D&D, several organizations apply existing or develop new software tools for integrating obtained radiation measurement data. To facilitate D&D technology selection, JAEA developed the Calculation System for Decontamination Effect (CDE) code to assess the impact of different decontamination factors on gamma dose rates. In addition to predicting changes in air dose rate after decontamination, CDE considers other effects, such as structural stability (damage from tsunami, earthquake, etc.), structure composition (clay, asphalt, metallic, etc.) and morphology (cracks, porosity, etc.), and initial contamination amount, type, and profile. JAEA has also developed the Restoration Support System for Environment (RESET) code for predicting the effects of decontamination. These tools allow JAEA to makes predictions for national and local governments; findings are used to provide advice and technical guidance concerning decontamination implementation

  TEPCO, along with other companies, also have software for displaying integrated three-dimensional integrated dose rate and contamination data obtained from drones, gamma cameras, PSFs, handheld equipment, car-borne detectors, remote-controlled helicopters, backpacks, and installed detectors. The software displays contamination and dose rate maps, showing hot spots, within and outside the reactor buildings as well as within the surrounding community. The N-Visage system from Sellafield is used for characterizing existing conditions (using measured data) and for predicting decontamination effects based on realistic conditions (e.g., consideration of terrain and sky shine). Predictions allow assessment of proposed decontamination work, promote understanding of obtained data, and facilitate efforts to obtain consent from residents.

Integrated dose information is used to determine how much material (e.g., leaf litter, organic material) needs to be removed during decontamination, to select the appropriate decontamination techniques (soils, road surfaces), and to check concentrations in water (both swimming pool water and water used for decontamination) to assure that it can be safely disposed of in normal drainage / sewage systems. When appropriate, dust samples are used to determine worker PPE and the potential for dust to spread contamination. Because of the range of measurement equipment and sampling approaches used, it is important to calibrate / inter-calibrate equipment and assess uncertainties associated with obtained data. In addition, environmental conditions need to be carefully recorded to allow for influences of weather (e.g., changing water content of soils and the presence of snow). Using instantaneous dose rates from various sources and locations, cumulative doses can be estimated as a function of time, wind speeds, and wind directions. Sensors need to be networked so that they can provide data in real time to analysts and modelers, who integrate and analyze the data and provide decision makers with actionable information. Inexpensive GPS and wireless data transmission technologies can be incorporated into personal Smartphone devices, collecting a wealth of data not only about individuals' external radiation doses, but also about where and when they received doses. The NRA has announced its plan to have citizens returning to evacuated areas wear personal dosimeters. However, it is not clear that these devices will achieve widespread public acceptance due to privacy concerns.

E.3.4. Summary and Future Considerations

To facilitate D&D related to the accidents at Fukushima, Japan is implementing several novel new technologies: new sensors, systems, and PPE are being deployed with new software to facilitate an integrated assessment of current conditions and predict conditions after various cleanup activities are completed. Results are also used to optimize future cleanup activities. In addition to a holistic evaluation of D&D risk reduction, cleanup measures are selected based on their predicted effectiveness, considering factors such as efficiency, required resources (cost and labor), waste generation, schedule, and safety. Recognizing that a varied ‘toolkit’ of response capabilities was needed, Japan is also conducting R&D to develop new technologies the facilitate this large-scale cleanup effort.

Several of these new technologies are of interest to the U.S. nuclear enterprise. As indicated in Section 2.4.5 it is suggested that additional efforts be devoted to facilitate deployment of these technologies for routine O&M activities. Candidate funding sources, such as the DOE NEET, NEUP, and LWRS programs, should encourage bi-lateral cooperation to facilitate deployment of these technologies.

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