ANL/LWRS-15/2

Light Water Reactor Sustainability Program

US Efforts in Support of Examinations at Fukushima Daiichi

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ABSTRACT

This report has been developed as a first step toward ensuring that the US obtains the maximum benefit from information obtained from the affected reactors at the Fukushima Daiichi Nuclear Power Station [Daiichi] during Decontamination and Decommissioning (D&D). This present work results from an effort, which was funded by the Reactor Safety Technologies Pathway of the Department of Energy Office of Nuclear Energy (DOE-NE) Light Water Reactor Sustainability (LWRS) Program, to develop consensus input from US experts for prioritized time-sequenced examination information and supporting research and development (R&D) tasks that could be completed with minimal disruption of Tokyo Electric Power Company, Incorporated (TEPCO) D&D activities at Daiichi. In this document, special attention is devoted toward identifying why such information is important and how it will be used to benefit the US nuclear enterprise. Preliminary cost (level of effort) estimates are provided for tasks recommended to be initiated within the next five years. The document also includes schedule information, which is based on recent information published by the International Research Institute for Nuclear Decommissioning (IRID) in a Mid-and-Long-Term Roadmap.

This report provides a basis for obtaining stakeholder support and establishing appropriate funding for obtaining this information. In many cases, information needs identified in this effort fall within international programs. It is anticipated that the US will participate in these international programs. In other cases, information may be obtained from Japanese activities. Hence, this report also provides a basis for ensuring that there is no duplication of US efforts related to examination information from the affected reactors at Daiichi.

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ACRONYMS

AFW	Auxiliary FeedWater
AM	Accident Management
ANL	Argonne National Laboratory
BSAF	Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CAM	Containment Air Monitor
CCI	Core Concrete Interactions
CNWG	Civil Nuclear Energy and Research Working Group
CRIEPI	Central Research Institute of Electric Power Industry
CSNI	Committee on the Safety of Nuclear Installations
D&D	Decontamination and Decommissioning
DOE	Department Of Energy
DOE-NE	Department of Energy Office of Nuclear Energy
DW	DryWell
ECCS	Emergency Core Cooling System
ELAP	Extended Loss Of AC Power condition
ENSREG	European Nuclear Safety Regulators Group
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EU	European Union
FAI	Fauske and Associates, LLC
FLEX	Diverse and Flexible Mitigation Capability (for accident mitigation)
FTE	Full Time Equivalent
FY	Fiscal Year[a]
GEH	GE-Hitachi Nuclear Energy, Limited
HP	Hold Points
HPCI	High Pressure Coolant Injection
IAEA	International Atomic Energy Agency
INL	Idaho National Laboratory
IRID	International Research Institute for Nuclear Decommissioning
IRM	Intermediate Range Monitor
JAEA	Japan Atomic Energy Agency
LWR	Light Water Reactor
LWRS	Light Water Reactor Sustainability
MCCI	Molten Core Concrete Interactions
METI	Ministry of Economy, Trade and Industry
MEXT	Ministry of Education, Culture, Sports, Science and Technology
MHI	Mitsubishi Heavy Industry
MSL	Main Steam Line

[a] In the US, the FY runs from Oct 1 through September 30; in Japan, the FY runs from April 1 through March 31

NDF	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NPS	Nuclear Power Station
NRA	Nuclear Regulatory Authority (Japan)
NRC	Nuclear Regulatory Commission
NUGENIA	NUclear GENeration II & III Association
OECD	Organization for Economic Cooperation and Development
ORNL	Oak Ridge National Laboratory
PCV	Primary Containment Vessel
PDCA	Plan→Do→Check→Act.
PLR	Primary Loop Recirculation
POC	Point-Of-Contact
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group
R&D	Research and Development
RB	Reactor Building
RCIC	Reactor Core Isolation Cooling
RPV	Reactor Pressure Vessel
RPI	Rensselaer Polytechnic Institute
RST	Reactor Safety Technology
SAMG	Severe Accident Management Guideline
SAREF	SAfety REsearch opportunities post-Fukushima
SARNET	Severe Accident Research NETwork
SFP	Spent Fuel Pool
SGTS	Standby Gas Treatment System
SNL	Sandia National Laboratory
SRV	Safety Relief Valve
SSC	Structures, Systems, and Components
TAMU	Texas A&M University
TEPCO	Tokyo Electric Power Company, Incorporated
TIP	Traveling In-core Probe
TMI-2	Three Mile Island Unit 2
TVA	Tennessee Valley Authority
US	United States
VIP	Vessel and Internals Program

US Input in Support of Examinations at Fukushima Daiichi

1. INTRODUCTION

Much isn't known about the endstate of core materials within Units 1, 2, and 3 at the Fukushima Daiichi Nuclear Power Station (NPS) [Daiichi]. Some of this uncertainty can be attributed to lack of information related to cooling system operation and cooling water injection. There is also uncertainty due to a lack of validated models for predicting Boiling Water Reactor (BWR) severe accident progression. However, similar to what occurred after the accident at Three Mile Island Unit 2 (TMI-2),[1] these Daiichi units offer the international community a unique means to obtain prototypic severe accident data from multiple full-scale BWR cores related to fuel heatup, cladding and other metallic structure oxidation and associated hydrogen production, fission product release and transport, and fuel/structure interactions from relocating fuel materials. In addition, these units may offer data related to the effects of salt water addition, vessel failure, ex-vessel core/concrete interactions, and Mark I containment liner attack. Much of the information obtained from these units may not only offer the potential to reducing uncertainties in BWR severe accident progression but also may offer the potential for PWR safety enhancements.

This report has been developed to ensure that the US obtains the maximum benefit from information obtained from the affected reactors at Daiichi during Decontamination and Decommissioning (D&D). This document results from an effort, which was funded by Reactor Safety Technologies Pathway of the Department of Energy Office of Nuclear Energy (DOE-NE) Light Water Reactor Sustainability (LWRS) Program, to develop consensus input from US experts for prioritized time-sequenced examination information and supporting research and development (R&D) activities that could be completed with minimal disruption of Tokyo Electric Power Company, Incorporated (TEPCO) D&D plans for Daiichi. Special attention is devoted toward identifying why such information is important and how it will be used to benefit the US nuclear enterprise. Preliminary cost (e.g., level of effort) and schedule estimates are provided for tasks that the expert panel recommended be started in the next five years. Results in this report provide a basis for obtaining stakeholder support and establishing appropriate funding for obtaining this information. In many cases, information needs identified in this effort fall within international programs. It is anticipated that the US will participate in these international programs. In other cases, information may be obtained from Japanese activities. Hence, this document also provides a basis for ensuring that there is no duplication of US efforts related to examination information from Daiichi.

1.1 Background

It is widely recognized that the accident at Three Mile Island Unit 2 (TMI-2) ultimately led to numerous safety enhancements that significantly improved nuclear power world-wide. Critical to such safety enhancements were the post-accident examinations at TMI-2 and associated data qualification efforts that provided the international nuclear community numerous insights related to pressurized water reactor (PWR) accident progression. Information gleaned from video and ultrasound inspections, relocated core debris, reactor pressure vessel specimens, nozzles, removed instrumentation, and available data from plant instrumentation were used to improve severe accident simulation models that form the technical basis for reactor safety evaluations. Increased understanding associated with advanced phenomenological models and risk assessment methods led to improved guidance for operators related to both methods and instrumentation needed to mitigate severe accidents. However, in order to reap such benefits, extensive post-accident examinations and associated tests were required.

Available data suggest that post-accident inspections at Daiichi Units 1, 2, and 3 offer a means to gain at least a similar, if not a more complete, understanding of accident progression in the other major type of US operating commercial nuclear power plants, BWRs, and phenomena affecting both BWRs and PWRs, such as the effects of saltwater addition, vessel failure, and ex-vessel phenomena. In addition, inspections from the affected reactors at Daiichi are unique because multiple reactors were affected. Data, models, and insights from post-accident inspections will inform many aspects of reactor safety, including severe accident modeling and simulation tools, severe accident management guidelines, improved plant training, and new or revised safety requirements in response to Fukushima. Technologies developed and lessons learned from such information can be used to prevent or mitigate future accidents. To increase the benefit from post-accident inspections that should accompany any D&D endeavors, it was recognized that an effort is needed to (a) identify data needs to ensure that key information is not lost; (b) identify sampling techniques, sample types, and sample evaluations to address each information need; and (c) help finance acquisition of the required data and conduct of the analyses.

1.1.1 TMI-2 Post-Accident Evaluation Process

Post-accident insights related to what occurred at TMI-2 were not available until over a decade after the event and required an integrated process (see Figure 1) that included post-accident videos, examinations of samples of core debris and vessel structures, instrumentation data, calculations with ‘best-estimate’ severe accident analysis tools, separate effects laboratory tests, and in some cases, data from large integral tests.[2] Analyses to interpret and integrate these information sources were crucial, since insufficient data were available from any single source to uniquely define a consistent understanding about the TMI-2 accident scenario.

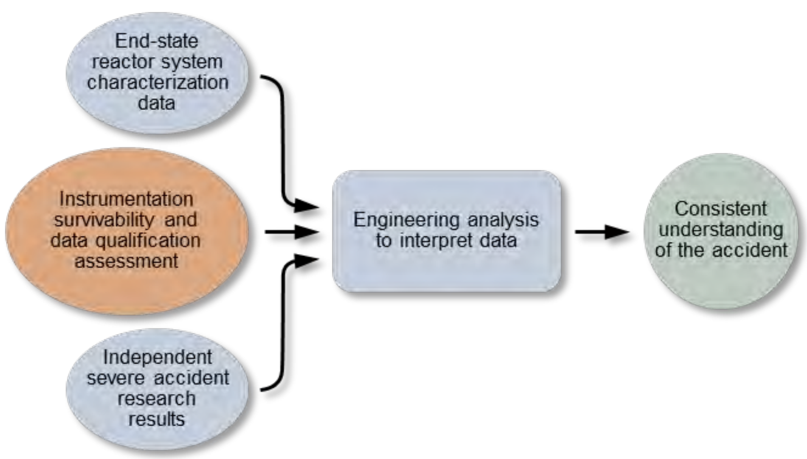

Figure 1. Integrated process used to develop understanding of TMI-2 accident. (Courtesy of Idaho National Laboratory [2])

Video examinations and ultrasonic scanning surveys were initially used to determine the shape and dimensions of materials remaining in the reactor vessel and the damage sustained by internal support structures and penetrations (see Figure 2). Several types of samples were removed from the reactor pressure vessel, including fuel, cladding, control rods, fuel support structures, and in-core instrumentation nozzles. Samples were also obtained from within the primary coolant system and the reactor containment building. Once initial inspections revealed the significant damage to the TMI-2 core, there was widespread interest in learning from this event and using these insights to improve plant safety. Inspections were supported by industry, government agencies, and international organizations. To minimize any downtime on the utility cleanup effort, post-accident inspection organizations teamed with industry to successfully develop and deploy specialized tools, such as a core bore sample drill, debris vacuum drying systems and casks for transporting samples to organizations for examination, and devices/methods for extracting samples and nozzles from the vessel lower head.

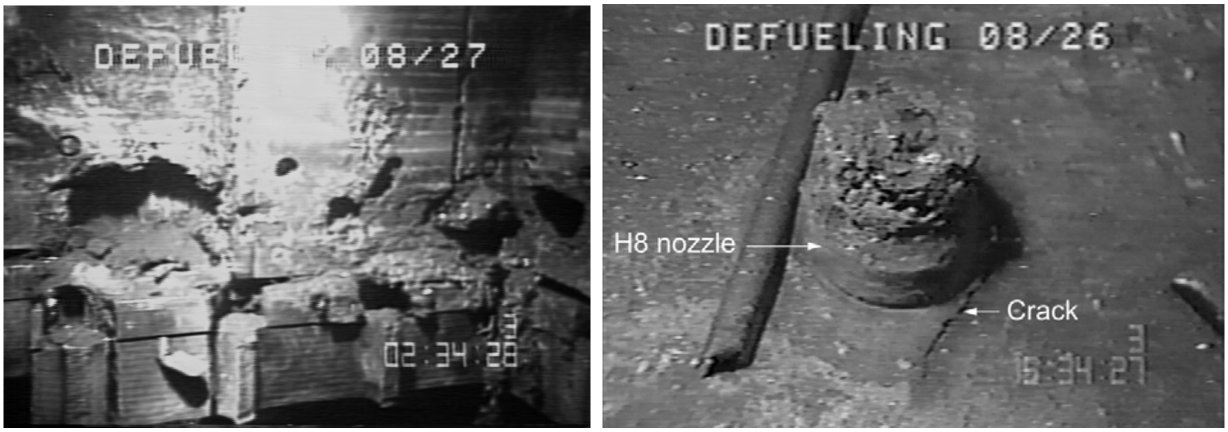

Figure 2. TMI-2 video examinations revealed locations where damage to core barrel and nozzles was more severe. (Courtesy of FirstEnergy)

1.1.2 Synergistic Efforts

The events at Fukushima have rekindled international interest in LWR severe accident phenomenology. Within the US, several activities have been initiated within the DOE-NE Reactor Safety Technologies Pathway of the LWRS program. In addition, there are several complementary international efforts that have been initiated. This section summarizes the objectives and recent accomplishments of synergistic US and international activities.

Within the US, post-event computational analyses [3, 4, 5, 6, 7] have identified several areas that may warrant additional R&D to reduce modeling uncertainties and to assist the industry in the development and refinement of Severe Accident Management Guidelines (SAMGs) to both prevent significant core damage and to mitigate source term release. Given the current state of LWR severe accident research and insights from the accidents that occurred at Daiichi, a technology gap evaluation [8] on accident tolerant components and severe accident analysis methodologies was completed with the goal of identifying any existing data and/or knowledge gaps. Results from this evaluation provide a basis for refining research plans to address key knowledge gaps in severe accident phenomenology that affect reactor safety and that are not being directly addressed by the nuclear industry or by the US Nuclear Regulatory Commission (NRC). The technology gap evaluation was completed by a panel of US LWR safety and operations experts with representatives from the DOE laboratories [i.e., Argonne National Laboratory (ANL), Idaho National Laboratory (INL), Oak Ridge National Laboratory (ORNL), and Sandia National Laboratory (SNL)], as well as industry [e.g., Electric Power Research Institute (EPRI), Nuclear Energy Institute (NEI), BWR Owner's Group (BWROG), PWR Owner's Group (PWROG), etc.]. Representatives from US government agencies (e.g., the NRC and the DOE-NE) and TEPCO participated as observers to inform this process. As noted by Lee [9] and Uhle [10], NRC experience related to severe accident phenomena is enhanced because of recent efforts to inform regulatory actions to address post-Fukushima activities, such as the Order for severe accident Capable Hardened Vents (EA-13-109), the Order for mitigating strategies for extended loss of AC power condition (ELAP) events (EA-12-049), and the Order for spent fuel pool instrumentation (EA-12-051). Based on the collective experiences of the US LWR safety and operations experts panel, thirteen gaps were identified in the areas of severe accident tolerant components and accident modeling. The results clustered in three main areas; namely, i) modeling and analysis of in-vessel melt progression phenomena, ii) Emergency Core Cooling System (ECCS) equipment performance under beyond-design-basis accident conditions, and iii) ex-vessel debris coolability and core-concrete interaction behavior relevant to accident management actions. In March 2015, a report was issued that summarizes the methodology by which the evaluation was completed, the identified gaps, and the R&D identified by the expert panel that may be appropriate for addressing these gaps.[8]

It is interesting to observe that the Atomic Energy Society of Japan [11] indicates that a similar gap analysis was recently completed within Japan. Twelve prioritized research topics, which were selected using input from the Japan Atomic Energy Agency (JAEA), Toshiba, Hitachi-GE Nuclear Energy, Mitsubishi Heavy Industry (MHI), Central Research Institute of Electric Power Industry (CRIEPI), and several universities (University of Tsukuba and Kyoto University), include development of new reactor materials (e.g., cladding and core catcher); evaluations of the performance of systems, such as the Passive Containment Cooling System, Autocatalytic Recombiners, Hydrogen Removal Systems, and Filter Venting Systems; and development of new instrumentation and measurement devices that can survive severe accident conditions.

A Civil Nuclear Energy Research and Development Working Group (CNWG) has been established under the U.S.-Japan Bilateral Commission on Civil Nuclear Cooperation to enhance coordination of joint civil nuclear R&D efforts between the DOE and Japan's Ministry of Economy, Trade and Industry (METI) and Ministry of Education, Culture, Sports, Science and Technology (MEXT).[12] Formal arrangements have been established covering collaboration in multiple areas including several areas relevant to LWR safety and post-accident evaluation; namely, i) severe accident code assessment, ii) accident tolerant fuel, iii) accident tolerant equipment, and iv) probabilistic risk assessment. Bilateral collaboration is underway in these areas. Discussions are underway to also include collaboration in the area of reactor examination planning.

The response to the Fukushima accident has been global, resulting in multiple activities by numerous national and international stakeholders. Post Fukushima-related topics, such as accident mitigation strategies, accident monitoring systems, and reactor safety, have already been the focus of international working groups and meetings sponsored by agencies, such as the International Atomic Energy Agency (IAEA), the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD), as well as associations and groups, such as NUclear GENeration II & III Association (NUGENIA) and the European Nuclear Safety Regulators Group (ENSREG).

One noteworthy example is the NEA's senior expert group on SAfety REsearch opportunities post-Fukushima (SAREF) which was created in 2013 by the Committee on the Safety of Nuclear Installations (CSNI) to establish a process for identifying and following up on opportunities for addressing safety research gaps and advancing safety knowledge related to the Daiichi nuclear accident.[13] Organizations from twelve countries are participating in this activity. The work scope includes identifying research opportunities based on information from Daiichi that will provide additional safety knowledge of common interest to the member countries. The SAREF identified 16 specific topics of interest in four main areas; namely i) severe accident progression, ii) structural/material behavior, iii) structures, systems, and components (SSC) performance, and iv) accident recovery. Activities are underway to further refine research recommendations for submission to the CSNI by June 2016. DOE and NRC are participants in this NEA project.

An ongoing parallel activity that is relevant to this effort is the OECD/NEA Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF) project.[14] The project, which is hosted by the Japan Atomic Energy Agency and other Japanese organizations, is an international effort aimed at performing accident reconstruction analyses using a number of severe accident codes, including MELCOR and MAAP. The reconstruction analyses make use of known accident boundary conditions and measurements, such as estimated water injections, operations of emergency equipment [e.g., Reactor Core Isolation Cooling (RCIC), High Pressure Coolant Injection (HPCI), etc.], reactor depressurization actions, and containment venting actions. These analyses both assess the collective of international severe accident analysis codes and provide analytical insights into the estimated damage state of each reactor. These damage states include estimates of melted core regions, relocated core materials to the lower head, possible pressure vessel rupture locations (e.g., lower head or steam line), and reactor cavity concrete attack by molten core materials. Hence, these analyses can help inform decommissioning activities by providing spatial estimates of core relocations and inform data needs that may be met during the decommissioning activities. In return, examination and photography of upper reactor vessel internals and steam lines can provide valuable information for validating code estimates of damage in these regions. The first phase of the BSAF project, which focused mainly on the accident progression and core damage phase, has been completed. Phase 2 of the BSAF project, which started in June 2015, is aimed at characterizing release and transport of fission products through the reactor vessel, containment, and reactor building, and ultimately, to the environment. Environmental releases will include both aqueous pathways as well as atmospheric releases. Validation information will be sought from sampling of radiological depositions along these release pathways, including the ground deposition data for cesium in the countryside around the accident site. Jäckel [15] illustrates the type of information from the Daiichi site [16, 17] that will be used in this effort and conclusions that can be obtained from these evaluations. Phase 2 of this project is anticipated to proceed over the next three years. DOE and NRC are also participants in this NEA project.

Synergistic activities sponsored by NUGENIA[18] include the Severe Accident Research NETwork (SARNET), which has the objectives of:

- Improving knowledge on severe accidents in order to reduce uncertainties on pending issues, thereby enhancing plant safety
- Coordinating research resources and expertise available in Europe
- Preserving the research data and disseminating knowledge

Participants in SARNET include representatives from 47 organizations; although most organizations are based in Europe, there are organizations from Korea, India, Japan, and the US (e.g., the NRC).

ENSREG [19] efforts to complete stress tests in the European Union (EU) are also of interest to the effort documented in this report. These “stress tests,” which were requested in March 2011, are defined as targeted reassessments of the safety margins in nuclear power plants. They consider ‘extraordinary’ external events, such as earthquakes and floods, and the consequences of other initiating events, such as airplane crashes, that have the potential to lead to multiple losses in safety functions. All operators of nuclear power plants in the EU had to review the response of their nuclear plants to those extreme situations. The operators’ reports were first reviewed by the national nuclear regulators. They then prepared summary national reports that are being reviewed by teams organized by ENSREG.

The need for better instrumentation was recognized after the TMI-2 event, and the events at Fukushima have again emphasized the importance of having a critical set of reliable, rugged, post-accident instrumentation. Several organizations, including the National Research Council of the National Academies of Science,[20] the OECD-NEA,[21] and the IAEA [22] have developed recommendations regarding the need for enhanced reactor and containment instrumentation to ensure that plant operators have appropriate information to help them cope with the effects of accidents with significant core damage. To address this need, several US efforts have been initiated by the DOE,[23, 24, 25] industry groups,[26, 27, 28, 29] and the NRC.[30, 31, 32, 33] Efforts have also been initiated by organizations within France [34] and Japan.[35] As described in References [20] through [35], these efforts differ; their scope ranges from survivability studies for existing instrumentation to development and deployment of new instrumentation systems for severe accident applications.

In summary, a range of post-Fukushima activities are underway; but none duplicate the effort documented in this report. However, the effort documented in this report clearly benefits from and provides input to other on-going efforts. Future efforts within the DOE Reactor Safety Technologies pathway will continue to be cognizant of and coordinate with other on-going efforts to ensure that duplication is avoided and the benefits are maximized.

1.2 Motivation

Post-accident examinations at Daiichi Units 1, 2, and 3 offer a means to gain at least a similar, if not a more complete, understanding of BWR accident progression relative to that obtained from TMI-2 for PWRs. BWR-specific features, such as channel boxes and cruciform control blades, certainly affect early stages of core degradation. Subsequent accident progression will be affected by natural convection to the steam separators and steam dryers that are massive steam structures above the reactor core. Until now, modeling of the progression of BWR accidents has largely relied upon expert opinion augmented with a few small, stylized experiments [see Reference 1 for a listing and description of the applicable BWR-specific tests] that did not attempt to represent all features of a BWR core, the core support structures, the reactor vessel, or the containment. The examination and dissection of the Fukushima reactors will provide a means to critically assess and validate these modeling assumptions. In cases where existing models are not valid, selected examination data will be crucial as input to developing new or improved phenomenological PWR and BWR models. For example, information on the timing and nature of seawater injection, potential data for modeling ex-vessel phenomena related to core concrete interactions (CCIs), and the survivability of selected instrumentation providing data used by codes are crucial inputs to new models applicable to PWR and BWR plants. Clearly, Daiichi examination information offers the potential for significant international reactor safety enhancements. Although Japanese organizations must lead efforts to obtain such examination information, a coordinated effort is needed to ensure that the US nuclear enterprise obtains the maximum benefit from information obtained from the affected reactors at Daiichi.

The significant benefits from such an effort can be grouped into the following areas:

- ***Improved LWR Accident Management Strategies and Staff Training*** **-** As uncertainties in predicting BWR and PWR accident progression and associated source terms are reduced, strategies for mitigating severe accidents will be improved, allowing the international community to use lessons learned from this work to prevent or reduce the consequences of future accidents. Likewise, the training simulations, which are based on severe accident systems analysis code models, will be improved, allowing plants to better prepare the staff to make decisions during severe accidents.
- ***Component and System Survivability*** **-** In addition to uncertainties related to instrumentation survivability, there are also uncertainties related to the performance of cooling systems, seals, and relief valves at Daiichi. Examinations could provide key information related to the performance of structures, systems, and components at the damaged Daiichi units.
- ***Reduced Uncertainties in Accident Progression and Source Term Models*** **-** Current BWR severe accident models are heavily based on expert opinion. Prototypic data from multiple full-scale reactors would greatly reduce modeling uncertainties related to BWR severe accident phenomena and possibly provide insights related to CCIs that are of interest to both BWR and PWR plant designs. Once additional information is gleaned from these Daiichi units, BWR models (and in the case of ex-vessel phenomena, PWR models) will be revised and, if required, additional experimental data could be obtained to validate code models.
- ***Instrumentation Data Qualification and Survivability Assessment*** **-** As discussed in Section 1.1.2, many improvements were made to plant instrumentation after the TMI-2 accident. However, the events at Daiichi suggest that additional enhancements may be needed to ensure that operators have adequate data to assess the status of the plant and the effects of mitigating actions that may be taken.

As part of this activity, data qualification assessments should be completed to provide confidence in input data for computational simulations. Examinations of instrumentation and cabling within and extracted from the affected units will provide important insights related to instrumentation survivability for post core damage diagnosis of plant conditions and selection of appropriate strategies.

- ***Preserving US Severe Accident Capabilities -*** In the US, funding for severe accident R&D decreased significantly in the late 1990s. There are few US severe accident experts under the age of 40. Similar to the way that post-accident examinations at the TMI-2 plant stirred interest and increased expertise, the examinations at Daiichi will provide exciting and important research, rekindling much needed expertise within the younger generation of nuclear engineers in the US.

Because of the benefit to global nuclear reactor safety, it is recognized that ultimately an international framework should be established to support these post-accident examinations. Although this international framework must be led by the Japanese, the US has a vested interest in such examinations. The US has the largest number of operating nuclear power plants in the world. Hence, US organizations – both industry and government—are major beneficiaries from any improvements in LWR severe accident knowledge just as the US was a major beneficiary of significant Japanese participation in prior international TMI-2 programs. US collaborative work with the international community in establishing this framework to support our Japanese colleagues would also be beneficial not just to the US and Japan but would maximize advances in reactor safety across the global nuclear energy community.

1.3 Objectives and Organization

The objective of this report is to provide the first step in a DOE-NE coordinated effort to ensure that the maximum US benefit is obtained from examinations at Daiichi. To this end, this report summarizes results from an effort to develop consensus input from US experts for a prioritized set of time-sequence examination information needs and supporting R&D activities that could be completed with minimal disruption of TEPCO D&D plans for the affected reactors at Daiichi. In addition to identifying information needs, special attention was devoted to providing the technical rationale for each need and to describing how obtained data would be used to benefit the US nuclear enterprise. This report also provides preliminary estimates for the cost (level of effort) and schedule for near-term tasks that would ensure that identified information is obtained and documented in an easy-to-understand and easy-to-access format for US researchers. Such estimates include working with TEPCO to identify what information has already been obtained and is planned to be obtained as part of their D&D efforts, as well as what additional effort would be required to address needs identified by US experts.

Information in this report will be used as a basis for obtaining stakeholder support and establishing appropriate funding for obtaining this information. In many cases, information needs identified in this effort fall within international programs. It is anticipated that the US will participate in such programs. In other cases, information may be obtained from Japanese activities. Hence, this report also provides a basis for ensuring that there is no duplication of effort as information is obtained from the affected reactors at Daiichi.

Remaining information in this report is organized as follows. Section 2 discusses current TEPCO and International Research Institute for Nuclear Decommissioning (IRID) efforts to support D&D activities at Daiichi and the significant amount of information already available from D&D activities completed by TEPCO. Section 3 documents the approach used to obtain a consensus among US experts on information needs from the affected reactors at Daiichi. Section 4 reports US information needs identified in this consensus effort. To ensure that these needs are addressed, near-term tasks (i.e., tasks that should be initiated within the next five years) are identified with preliminary cost and schedule estimates. Section 5 summarizes insights from this effort and recommendations related to the path forward for this program.

References cited in this report are listed in Section 6. Appendices to this document provide more detailed information. Appendix A provides lists of attendees and agendas from expert meetings held during FY15 as part of this effort. Appendix B provides tables with detailed information needs developed during US severe accident expert meetings. Appendix C provides results from an expert meeting held in FY14 as part of this effort. Appendix D contains roadmaps produced by the Japanese Government that detail planned D&D activities. Appendix E contains information needs provided by the US industry for inspections and identifies where these needs were incorporated into Appendix B tables.

2. DECONTAMINATION & DECOMMISSIONING

Daiichi D&D efforts must address new technical challenges not encountered at TMI-2. For example, the use of saltwater may have corroded and degraded the strength of structures and may affect the waste materials that must be processed and disposed of after the accident. Likewise, the properties of materials that may have formed as fuel interacted with in-vessel structures, as well as possible interactions with ex-vessel structures and containment concrete, may differ from what was observed at TMI-2. In addition, the multiple core damage and hydrogen explosion events left these units in more damaged conditions than observed at TMI-2.

While the primary objective is to complete the D&D efforts as early as possible, D&D efforts must not adversely impact the safety of the general public and the safety of plant workers. D&D activities must be monitored to alleviate concerns about recriticality, increasing radiation levels, radiation releases, increasing hydrogen concentrations, increasing temperatures, structural degradation, and non-nuclear industrial accidents. To accomplish all these objectives simultaneously, the risks of proposed D&D work processes have been and will continue to be evaluated by TEPCO.

In 2015, the government of Japan reorganized organizations involved in D&D efforts at Daiichi. Major organizations involved in this new structure are shown in Figure 3. [36, 37] The Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF) has been established to strengthen decommissioning strategies, and work was initiated to revise the strategic plan and medium/long-term roadmap. In this new organizational structure, the NDF will play a major role as a coordinator of decommissioning strategy, and R&D. Within the Japanese government, the Ministry of Economy, Trade and Industry (METI) has a lead role in establishing policy for and monitoring progress of D&D efforts. In the area of R&D, the roles of IRID, JAEA, and TEPCO are extremely important. In addition, the Nuclear Regulatory Authority (NRA) oversees D&D activities to ensure that necessary safety measures are taken and that the plant is maintained in a stable condition.

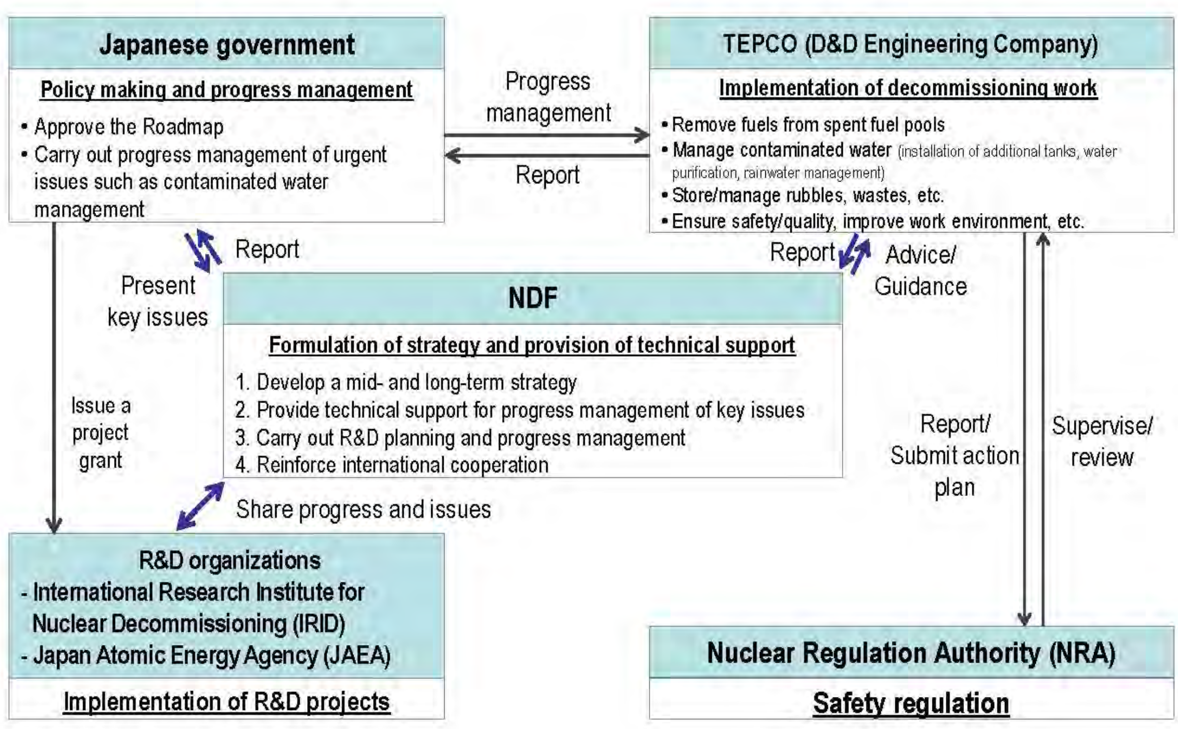

Figure 3. Organizations involved in decommissioning Daiichi. (Courtesy of NDF [37])

The NDF issued a strategic plan [37] that documents the technical foundation for the government of Japan's medium/long-term roadmap. In this plan, policies governing the fuel debris removal method are determined by stressing the following:

- PDCA[b] approach for decommissioning and R&D
- Formulation of a risk-based decommissioning strategy
- Consideration of multiple methods of debris removal
- Examination of waste strategies
- Formulation of a decommissioning R&D plan
- Broadening of international collaboration
- Broadening of technical coordination and discussion

Current plans emphasize activities that will reduce risk and recover the disaster-affected area at Daiichi. As documented in the strategic plan, the risk profile of Daiichi is developed based on analytical results for the 'hazard potential' and the 'likelihood of loss of containment function' (see Figure 4). D&D activities are prioritized based on risk reduction. Because there is uncertainty in many aspects of the plant conditions, especially with respect to the internal conditions of the primary containment vessel (PCV), various approaches are being considered for D&D activities.

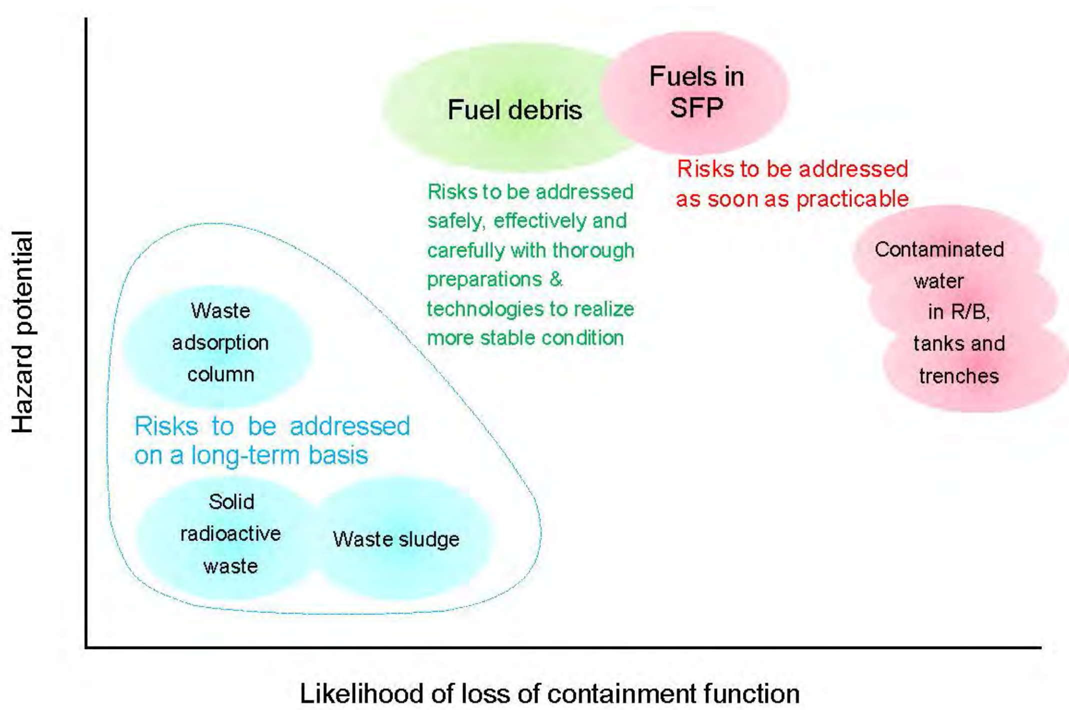

Figure 4. Risk profile evaluation process. (Courtesy of NDF [37])

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[b] PDCA cycle: A method for carrying out work smoothly and achieving continuous improvement by repeatedly cycling through the steps: Plan→Do→Check→Act. This approach is somewhat akin to the Integrated Safety Management (ISM) approach utilized by some US organizations.

2.1 Mid-and-Long-Term D&D Roadmap

Current TEPCO and Government of Japan D&D plans are documented in a roadmap, which is updated periodically as new knowledge is gained from the affected reactors at Daiichi. The initial “Mid-and-Long-Term Roadmap towards the Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station Units 1-4” (i.e., the Mid-and-Long-Term Roadmap) was finalized in December 2011 at the ‘Government and TEPCO's Mid-to-Long Term Countermeasure Meeting’ to indicate processes to recover from the accident at Daiichi. In June 2013 and June 2015, the Council for the Decommissioning of TEPCO's Fukushima Daiichi NPS issued revised versions of the roadmap.[38] Further revisions will take place based on actual conditions.

The Mid-and-Long-Term Roadmap divides the time until completion of D&D into phases, identifies major tasks to be undertaken onsite, and the associated R&D schedule (see Figure 5; more detailed roadmap figures for Units 1 through 4 are found in Appendix D). The schedule was prepared based on current knowledge of the plants and analyses of differences in conditions of the units. For example, because Unit 2 experienced less damage to the reactor building, several D&D activities within the building were completed earlier in this unit. Efforts were made to optimize opportunities to overlap required processes and operations between units. However, schedules may change as additional knowledge is gained.

Figure 5. Summary definition of roadmap phases. (Courtesy of IRID [39])

Phase 1 represents the time period between plant stabilization (i.e., radiation levels are low and releases are minimized) until the time when fuel removal from the spent fuel pool (SFP) begins. Phase 2 started in November 2013 with activities to remove the spent fuel from Unit 4 and will continue until fuel is removed from the reactors. Phase 2 includes R&D for fuel removal and PCV repair operations. This includes R&D related to removing fuel from the spent fuel pools, preparing for removal of fuel from the Reactor Pressure Vessel (RPV), and processing and disposal of solid radioactive waste. In addition, there is R&D related to alternative options for remote technologies that could reduce the challenges associated with D&D. Reference 40 provides additional details related to the scope and schedule of R&D activities. Phase 2 activities will require approximately 10 years to complete.

Phase 3 spans from the completion of Phase 2 until the plant is decommissioned. It is estimated that Phase 3 activities will be completed within 30 years (resulting in up to 40 years for the complete decommissioning of the affected units). Because of the technical challenges associated with Phase 2 and 3 activities, "holding points" (HPs) were established as important junctures where decisions will be made regarding the transition to the next step. Such decisions include whether additional R&D is required, which option of multiple options for completing a task will be pursued, etc. As an example, consider the hold points, HP1-1 and HP3-1, related to which option was pursued in Unit 1 and Unit 3, respectively, related to the reactor building (RB) cover (see Figure 6 through Figure 8; additional roadmap figures are found in Appendix D). Similar HPs exist for Units 1, 2, and 3 related to which technology option will be pursued for removing the fuel debris. For example, it is currently considered that the TMI-2 'submersion approach,' in which fuel is removed under water to minimize worker exposure, will be used at Daiichi. However, the submersion approach requires that equipment be developed that can fit within the PCV and that water leakage from the PCV be stopped. Hence, alternate methods for debris removal are under consideration. As discussed in Section 2.2, much of the R&D required for input for these decisions is being managed by IRID. Furthermore, as noted in Reference 38, it is recognized that information collected from these reactors is important to Japan and foreign organizations. Hence, all data obtained from D&D activities will be carefully collected, analyzed, and archived.

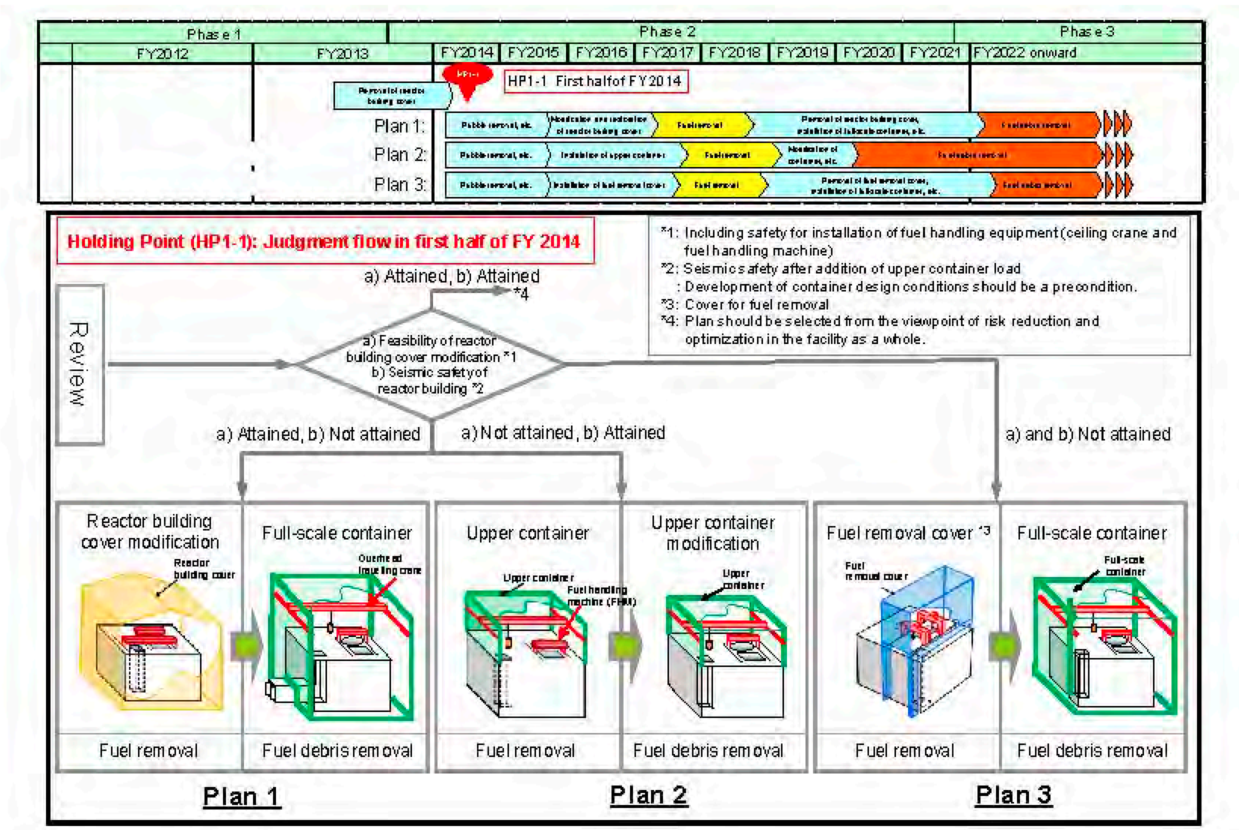


Figure 6. Unit 1 spent fuel pool and fuel debris removal roadmap with detail HP1-1. (Courtesy of TEPCO [38])

As of June 2014, TEPCO has obtained limited views inside the PCV of Units 1 and 2 [41 through 43] but no views inside the RPVs of any of the affected units. Limited insights about conditions inside the PCVs are obtained by measuring temperatures and dose rates. As observed in References 44 through 46, TEPCO continues to perform activities that will allow them to better estimate debris location; and they are accumulating data to evaluate various decommissioning methods and new approaches to handle technical difficulties and issues which hinder the progress of decommissioning R&D efforts.

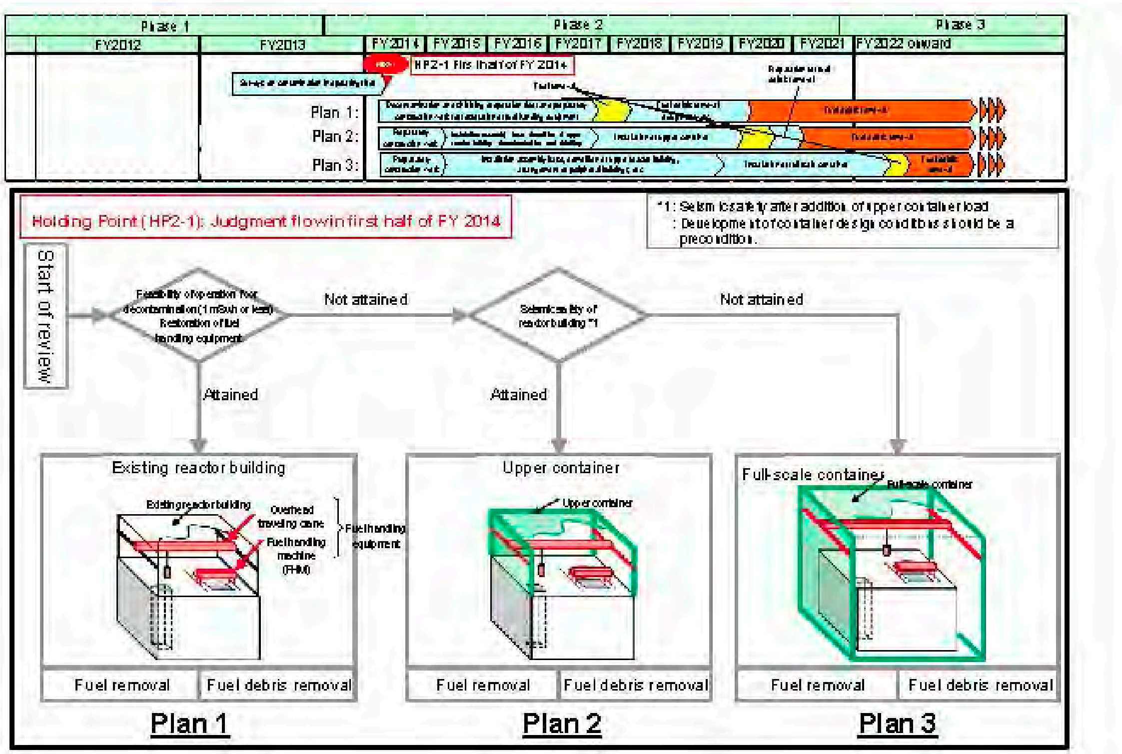


Figure 7. Unit 2 spent fuel pool and fuel debris removal roadmap with detail HP2-1. (Courtesy of TEPCO [38])

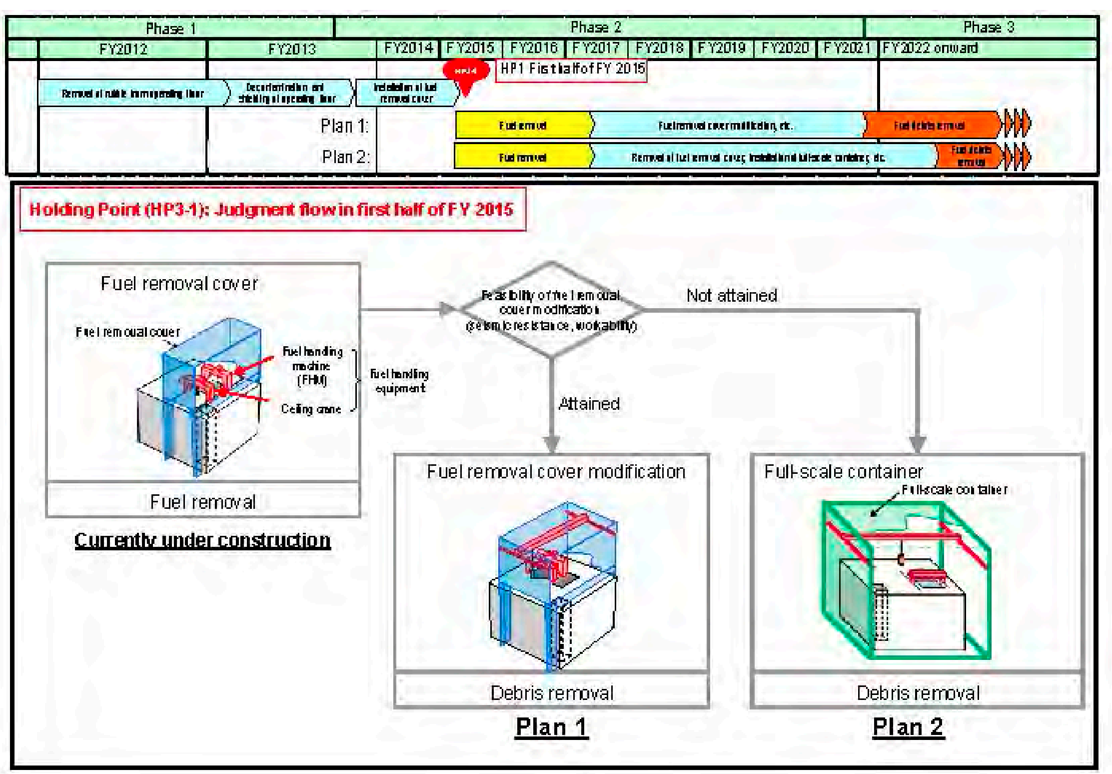


Figure 8. Unit 3 spent fuel pool and fuel debris removal roadmap with detail HP3-1. (Courtesy of TEPCO [38])

2.2 IRID

IRID is a consortium composed of eighteen organizations that formed in August 2013 to develop nuclear power plant decommissioning technologies based on the collective wisdom from Japan and abroad (see Table 1).[38] IRID's long-term objective is to foster, accumulate, and improve decommissioning and other related technologies for the decommissioning of reactors throughout Japan. In the near term, IRID's focus is to develop technologies required for decommissioning of Daiichi with consignment expenses and subsidies received from the government of Japan. IRID works with TEPCO's Fukushima Daiichi D&D Engineering Company to identify D&D needs at Daiichi and uses this information to integrate management systems for developing various decommissioning technologies. IRID R&D follows the Mid-and-Long-Term Roadmap (see Section 2.1). However, there are instances where IRID makes suggestions which are incorporated into the Mid-and-Long-Term Roadmap. Figure 9 illustrates IRID's organizational framework.

Table 1. IRID organization members [39]

Type	Organization
Administrative Agencies	• Japan Atomic Energy Agency
	• National Institute of Advanced Industrial Science and Technology
Manufacturers, Vendors	• Toshiba Corporation
	• GE-Hitachi Nuclear Energy, Limited (GEH)
	• Mitsubishi Heavy Industries, Limited
	• ATOX Company, Limited
	• Japan Nuclear Fuel, Limited
Electric Utilities	• Hokkaido Electric Power Company, Incorporated
	• The Chugoku Electric Power Company, Incorporated • Electric Power Development Company, Limited
	• TEPCO
	• Chubu Electric Power Company, Incorporated
	• Hokuriku Electric Power Company
	• Kansai Electric Power Company, Incorporated
	• Tohoku Electric Power Company, Incorporated
	• Shikoku Electric Power Company, Incorporated
	• Kyushu Electric Power Company, Incorporated
	• The Japan Atomic Power Company

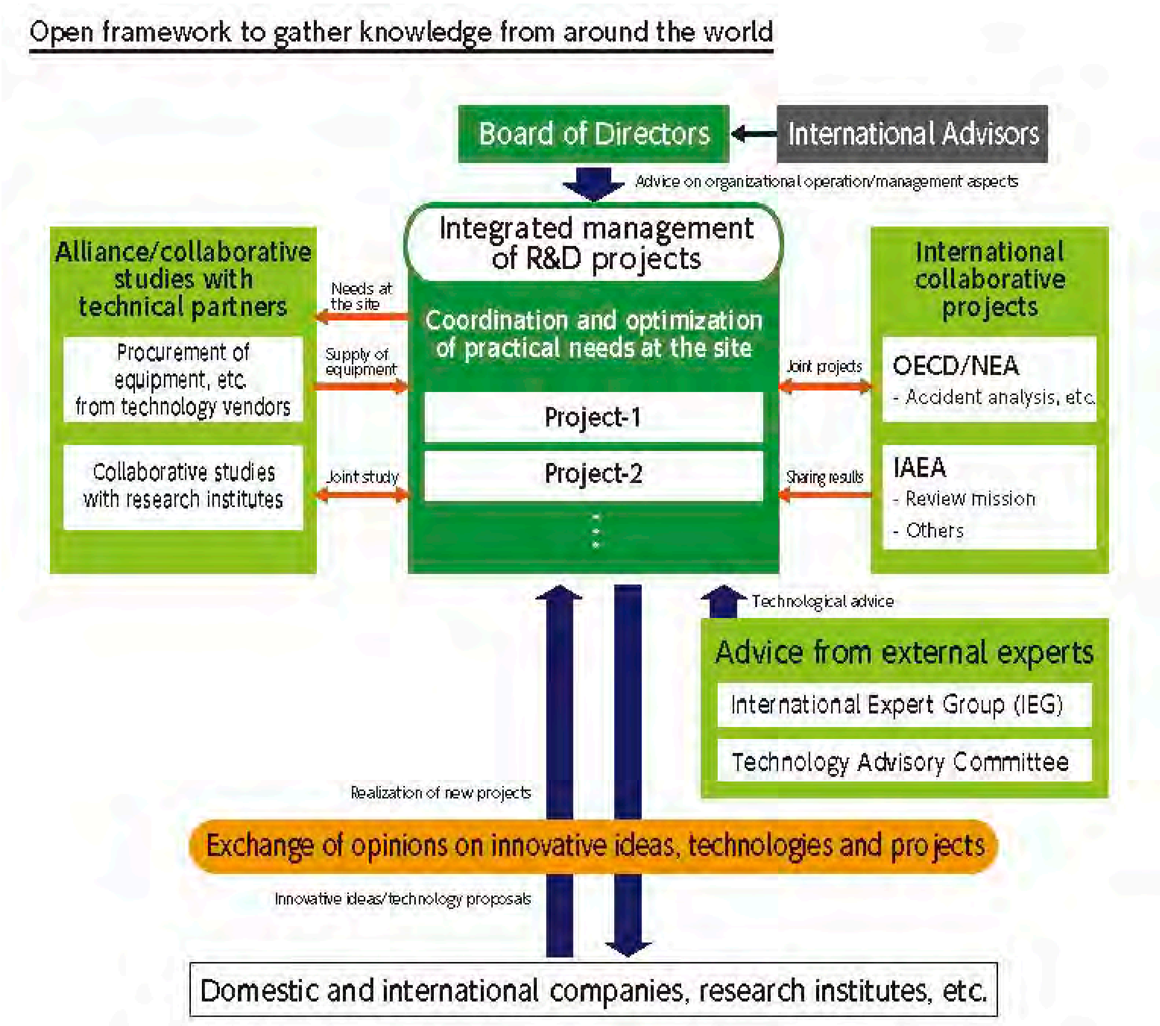


Figure 9. IRID framework to integrate research and collaborative projects. (Courtesy of TEPCO [39])

Since its inception, IRID has allocated funding from the government of Japan to support R&D of technologies required for decommissioning of Daiichi.[39] Current IRID projects (see Table 2) address issues that have been identified to safely retrieve nuclear fuel from the cores of the reactors and address issues related to the treatment/disposal of radioactive waste generated by the accident. Recent progress on each of these projects is documented in several references.[39,47] Hence, as depicted in Figure 3, progress in decommissioning Daiichi is accomplished as a coordinated effort between the NDF for making strategy- and technology-related decisions, TEPCO for on-site operational activities, and IRID for overseeing technology development.

Table 2. Current IRID projects (as of September 2014) [39]

Group[c]	Project Title
Fuel Removal from SFP	Evaluation of long-term integrity of the fuel assembly removed from the SFP
	Study of treatment method of damaged fuel removed from the SFP
Fuel Debris Removal	Development of technology for remotely operated decontamination in the RB
	Development of technology to identify and repair leakage points in the PCV
	Development of technology for investigation inside the PCV
	Development of technology for investigation inside the RPV
	Development of technology for collecting, transferring and storing fuel debris
	Development of technology for evaluating integrity of the RPV/PCV
	Development of technologies for controlling fuel debris criticality
	Assessing conditions inside reactor through application of severe accident analysis code
	Development of technology for identifying properties of and treating fuel debris
Treatment and Disposal of Radioactive Waste	Study to examine technologies for the construction of a conceptual framework for treatment and disposal of accident waste
	Radioactivity analysis for standing trees sampled at the site (Evaluation of contamination conditions within the site)

2.3 Selected Information from Affected Reactors

As part of their D&D activities, TEPCO has been and will continue obtaining information of interest to the international community. As noted in Reference 38, TEPCO recognizes that information collected from these reactors is important to Japan and foreign organizations. Hence, all data obtained from D&D activities is carefully collected, analyzed, and archived. However, TEPCO efforts are primarily focused on obtaining data required to support D&D efforts, rather than providing data to the international community that could be used to enhance safety (e.g., data for validating severe accident models, source term models, etc.). The DOE is working to establish a focused activity to work with TEPCO to learn what information is being obtained and to communicate this information to cognizant US experts that could use this information to enhance safety of the commercial fleet. To emphasize this point, selected examples are described in this section. These examples were selected because information and inspections being conducted or completed by TEPCO could address specific data needs identified by the expert panel in Section 4.

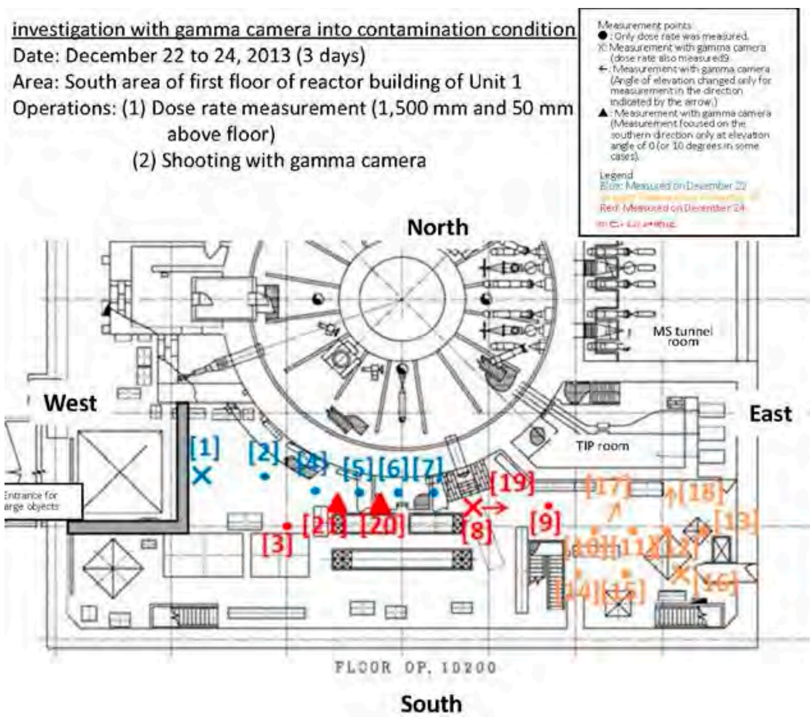
2.3.1 Dose Readings in the Reactor Buildings

As noted in Section 4 (and the detailed RB information tables in Appendix B), there is consensus amongst US experts that dose readings and radionuclide surveys obtained from the reactor building are important for gaining insights related to what occurred during the events at each unit, especially with respect to

[c] In addition to the three primary R&D research groups, IRID holds workshops and promotes collaboration with universities and research institutes to encourage development of the human resources required for the 40 year Daiichi D&D activities.

component or penetration seal failure and for benchmarking dose assessment codes. Ultimately, this information and associated evaluations could provide a basis for plant modifications, improved accident simulation/dose assessment code models, better severe accident management guidance, and improved staff training based on the enhanced accident simulation codes. As discussed in Section 1.1.2, measurements [16, 17] from the affected units at Daiichi, such as radionuclide surveys and evaluations of activity in concrete and water samples, can be used to validate radiological deposition along release pathways. Insights gained from this information have both BWR and PWR applications.

TEPCO and IRID presentations indicate that dose measurements and radionuclide survey information are critical to D&D activities and are being obtained from the affected reactor buildings. For example, as shown in Figure 10, dose rates (in mSv/hr) are being obtained from Unit 1 using a dosimeter mounted on a robot at locations 50 and 1500 mm from the reactor building floor. In addition, at selected (high dose) locations, concrete samples were removed using a robot for further evaluation. Likewise, as shown in Figure 11, dose rates (mSv/hr) were obtained at 5 meters above the floor in Unit 3 reactor building at selected dates and locations, starting on November 6-7, 2013 (red dots and bracketed numbers for other dates) and continuing through August 2014 (mustard dots).



Measurement point	Dose rate measurement (mSv/h)	
	Height: 1,500 mm	Height: 50 mm
[1]	12.97	15.87
[2]	30.84	20.19
[3]	105.5	–
[4]	61.95	40.83
[5]	65.17	42.99
[6]	25.52	31.85
[7]	41.65	17.55
[8]	52.32	30.64
[9]	–	–
[10]	1661.79	185.92
[11]	1096.64	330.98
[12]	510.51	274.30
[13]	313.57	244.23
[14]	139.11	38.17
[15]	91.31	35.44
[16]	166.98	133.37
[17]	659.38	157.75
[18]	202.94	144.04

Figure 10. Unit 1 reactor building dose map. (December 2013; courtesy of TEPCO [47])

Although this information is being obtained by TEPCO to support D&D activities, there are currently no efforts by TEPCO to use this information as a basis for future enhancements to reactor safety. If the US wishes to use this information, an effort is needed to have cognizant experts evaluate the adequacy of the data for potential future applications, make requests for additional information (if needed), make recommendations related to any required plant modifications or accident management guidance, and to assess the data with accident simulation codes and make any required modeling improvements.

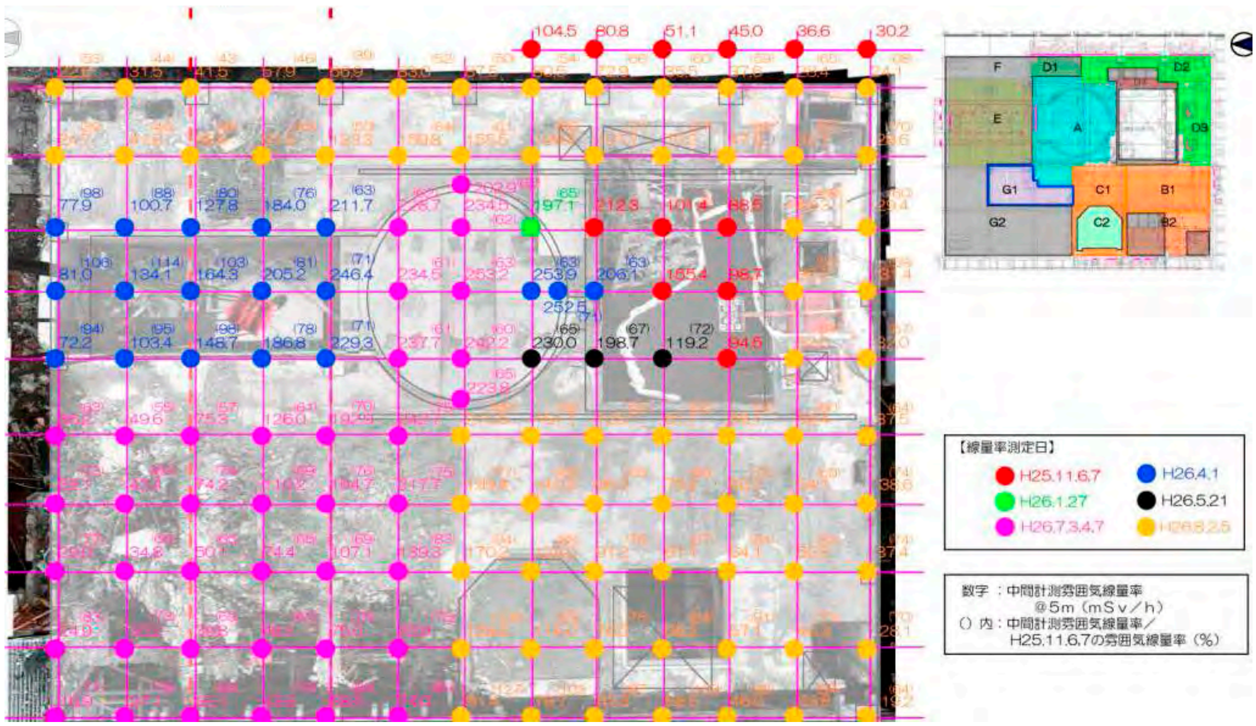


Figure 11. Unit 3 RB dose map. (Nov. 2013 through Aug. 2014; courtesy of TEPCO [48])

2.3.2 Component Leakage

As noted in Section 4 (and the detailed information needs tables in Appendix B), there is consensus amongst US experts that information related to penetration and piping integrity is important for understanding what occurred at each unit. Ultimately, this information and associated evaluations could provide a basis for plant modifications, improved accident simulation/dose assessment code models, improved severe accident management guidance, and improved staff training based on enhanced accident simulation codes. Insights gained from this information could potentially be applied to both BWRs and PWRs.

TEPCO and IRID presentations indicate that penetration and piping leakage information is critical to D&D activities. As discussed in References 45 and 47, leakage information is being obtained by TEPCO to support D&D activities to stabilize the PCV, and specialized equipment has been developed and is being deployed to detect and repair leakages found in Units 1, 2, and 3. For example, Figure 12 shows PCV leakage locations identified in Unit 1. If the US wishes to use this information as a basis for enhancements to reactor safety, an effort is needed to have cognizant experts evaluate the adequacy of the data for potential future applications and make requests for additional information (if needed).

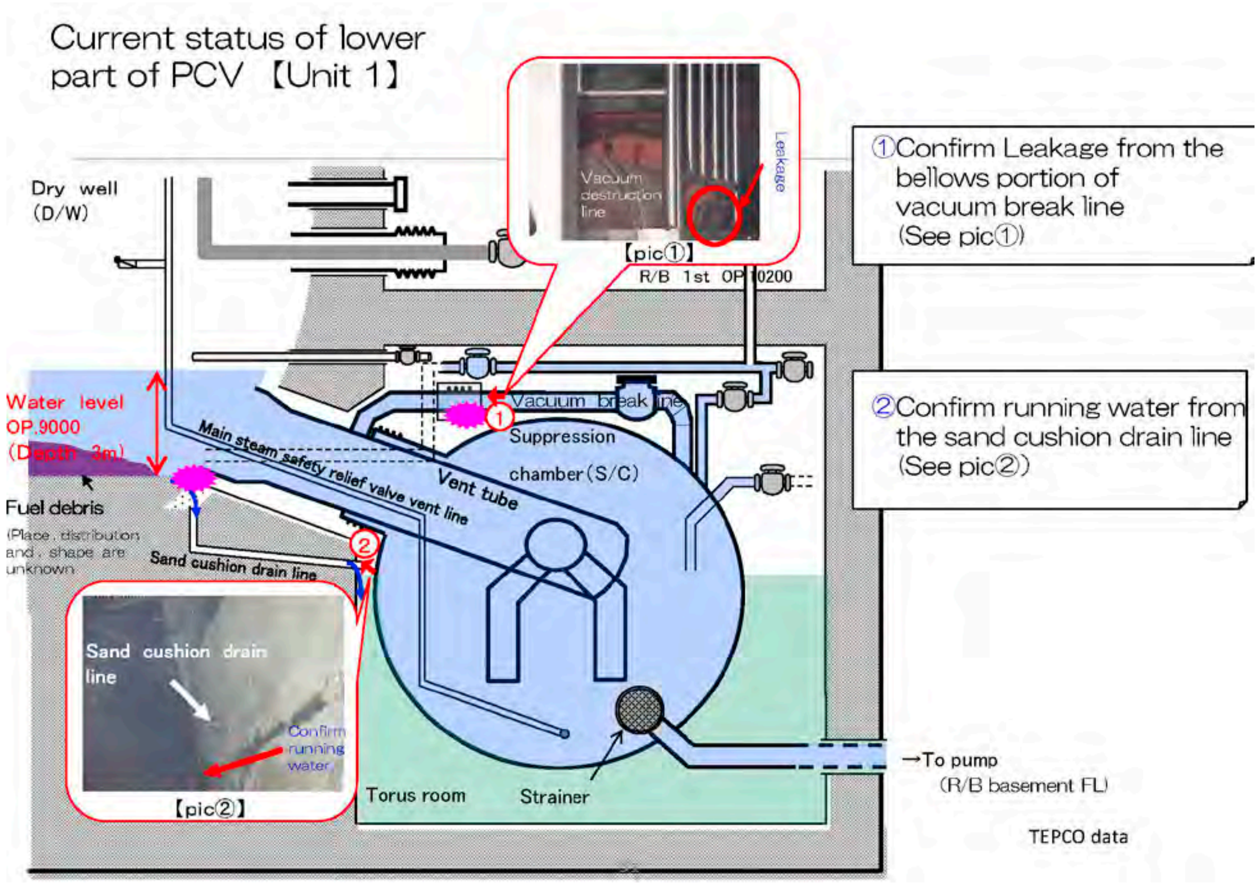


Figure 12. Unit 1 PCV leakage locations. (Courtesy of TEPCO [47])

2.3.3 Spent Fuel Cask Inspections

Section 4 results (and more detailed information needs documented in Appendix B tables) indicate that US experts identified a consensus need for survivability inspections of plant systems and components, such as dry storage casks. Ultimately, this information and associated evaluations could provide a basis for possible plant modifications and improved severe accident management guidance. Insights gained from this information have both BWR and PWR applications.

A recent TEPCO presentation [49] indicates that inspections of nine dry storage casks were completed in May 2013. As indicated in Figure 13, results from these inspections indicate that there was no damage to the casks, the fuel bundles, or the fuel baskets.

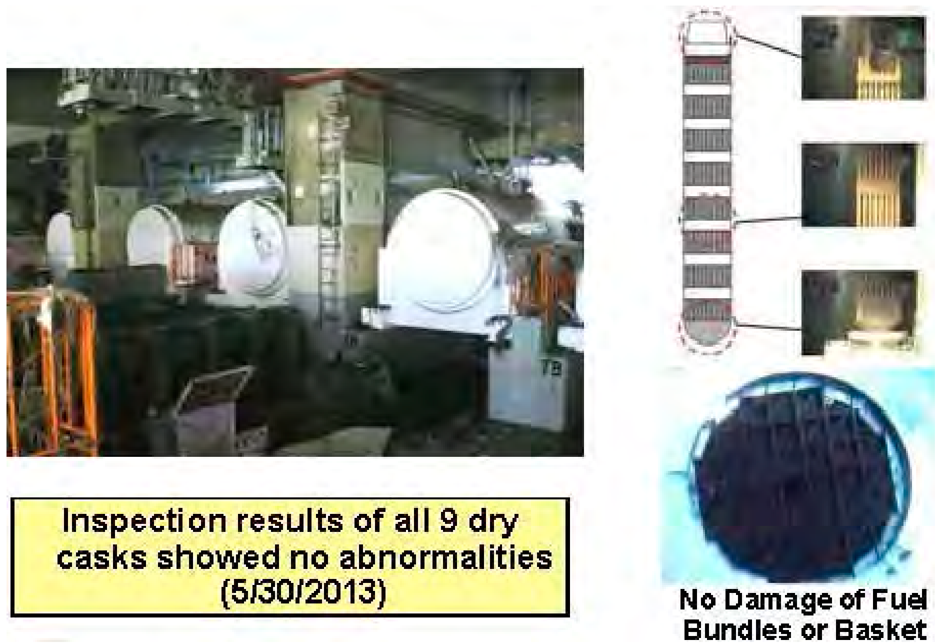


Figure 13. Dry cask containment inspections. (May 2013; courtesy of TEPCO [49])

Such inspections were obtained by TEPCO to support D&D activities. Currently, there are no efforts by TEPCO to use this information as a basis for future enhancements to reactor safety. As noted above, if the US wishes to use this information, an effort is needed to have cognizant experts evaluate the adequacy of the data for potential future applications, make requests for additional information (if needed), and make recommendations related to any required plant modifications or accident management guidance based upon insights obtained from this information.

2.4 Summary

D&D efforts at Daiichi present new technical challenges not previously encountered in the nuclear industry. The damage incurred from the combination of earthquake, flooding, saltwater, and hydrogen explosions, coupled with uncertainties in the performance of SSCs and in core material behavior during BWR accident progression, presents unique D&D challenges. Based on the best available information at this time, the Japanese government has developed a roadmap that outlines all planned D&D activities. The roadmap identities locations or holding points (HPs), where current uncertainties must be overcome before major decisions are made. For each HP, contingency options and required R&D to support various options are identified (to the extent possible). R&D activities, which are managed by IRID, are underway to provide input to these HPs. The roadmap is periodically updated by the Japanese government as additional information from each unit becomes available.

The roadmap provides US experts general insights about the schedule and types of activities being completed by TEPCO. In addition, results from these activities are posted on TEPCO's website and discussed during weekly teleconferences hosted by TEPCO. Several example results are discussed in this section to illustrate that TEPCO is already obtaining information that could address data needs identified by US experts. Nevertheless, it is not easy at this time for an individual US researcher to be cognizant of specific D&D activities underway and data being obtained from these activities. Basic knowledge gives experts some insights (e.g., decontamination activities require dose measurements and radionuclide surveys), but additional interactions are needed to obtain insights about the density and frequency of data being obtained and evaluate if such data are adequate for desired reactor safety enhancement activities.

TEPCO has publicly expressed their willingness to collaborate globally for the purpose of enhancing nuclear safety. In fact, TEPCO has acknowledged that international participation may be beneficial because of expertise related to severe accident progression and, in the case of the US, because of expertise gained from prior TMI-2 D&D efforts.

However, financial constraints and national needs dictate that TEPCO's primary responsibility is to obtain information required to support D&D activities at Daiichi. Still, TEPCO has requested (as a first step towards maximizing the international benefit from information obtained during their D&D efforts) that the US document consensus information needs, along with suggested methods for obtaining the requested information and the intended use of that information. In addition, if there are situations where current D&D plans could preclude TEPCO's future ability to obtain desired information, such situations should be identified. This report documents this information.

3. APPROACH

The approach followed to develop consensus input from US experts for the Daiichi forensics examinations is illustrated in Figure 14. The FY 15 approach involved two expert panel meetings (shown in orange boxes). These meetings were held in January 2015 (at ANL in Argonne, IL) and in May 2015 (at DOE Headquarters in Washington, DC). Agendas for each of these meetings are included in Appendix A of this report.

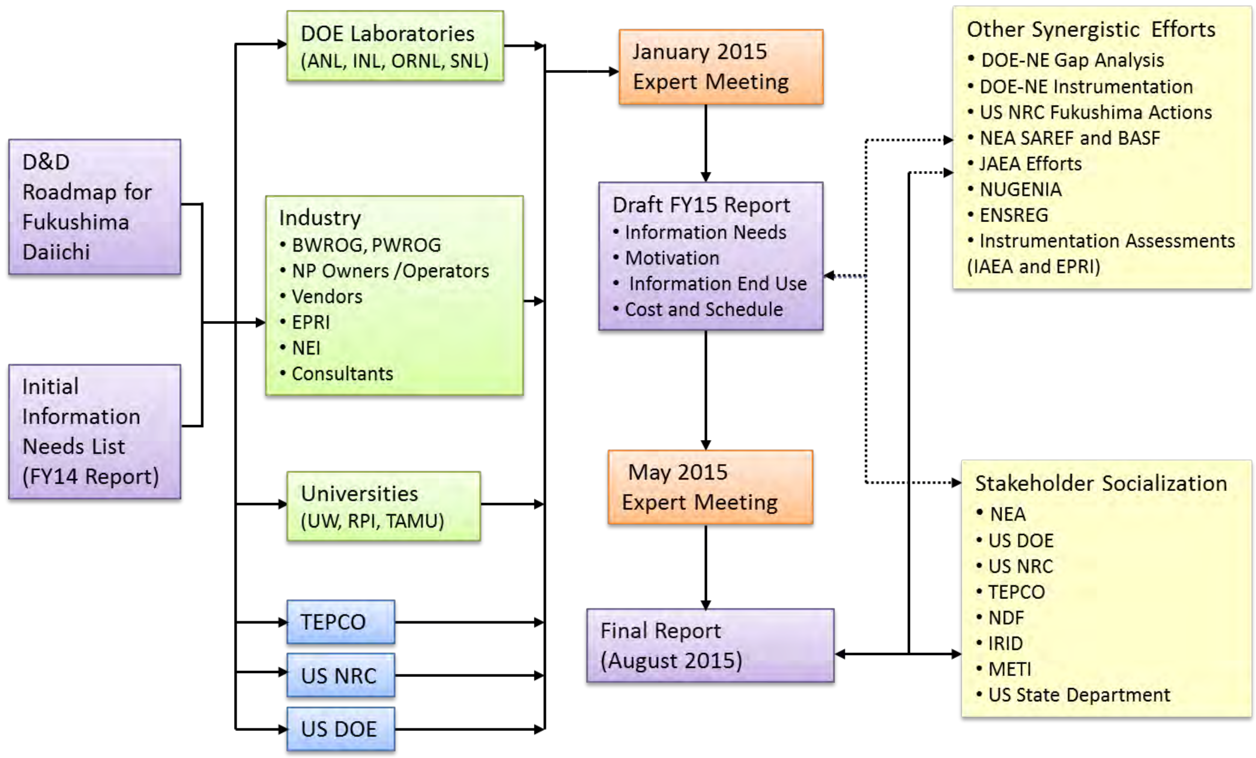


Figure 14. Approach for developing US input.

As indicated by the green boxes in Figure 14, experts from industry, universities, and national laboratories participated in this process. Experts from US government organizations (the NRC and the DOE) and from TEPCO also attended and informed participants during these meetings. In particular, presentations by representatives from TEPCO [44 through 46, 50 through 52] and the BWROG [53] were particularly helpful at these meetings. Table 3 lists specific organizations represented by experts at meetings held in FY14 and FY15. Clearly, this effort to develop consensus has strived to include a broad spectrum of stakeholder input.

Four documents (shown as purple boxes in Figure 14) were important to this process. As a starting point, this effort considered the Mid-and-Long-Term Roadmap for D&D (see Section 2) and a preliminary document issued by DOE-NE that reported results from an Expert Panel Meeting held in November 2013. The main report and Attachment A documenting results from this meeting, which was issued as a draft in May 2014, is included as Appendix C of this report.

Table 3. Organizations represented in expert meetings

Type	Organization
Government Agencies	• US Department of Energy Office of Nuclear Energy (US DOE-NE)
	• US Nuclear Regulatory Commission (US NRC)
National Laboratories	• Argonne National Laboratory (ANL)
	• Idaho National Laboratory (INL)
	• Oak Ridge National Laboratory (ORNL)
	• Sandia National Laboratory (SNL)
Plant Owner Groups	• BWR Owner's Group (BWROG)
	• PWR Owner's Group (PWROG)
Plant Owners/ Operators	• Exelon Corporation
	• Southern Nuclear Company
	• Tennessee Valley Authority (TVA)
	• Tokyo Electric Power Company (TEPCO)
Universities	• Rensselaer Polytechnic Institute (RPI) (only participated in November 2013 meeting)
	• Texas A&M University (TAMU) (only participated in November 2013 meeting)
	• University of Wisconsin - Madison (UW)
Vendors	• AREVA (only participated in November 2013 meeting)
	• GE-Hitachi (GEH)
	• Westinghouse Electric Corporation (retired)
Other	• Electric Power Research Institute (EPRI)
	• ERIN Engineering and Research, Incorporated
	• Fauske and Associates, LLC (FAI)
	• MPR Associates, Inc. (only participated in November 2013 meeting)
	• Nuclear Energy Institute (NEI)
	• Rempe and Associates, LLC

The third document is a draft version of this report, which was initially issued in March 2015. This draft summarized results from the January 2015 expert panel meetings held to obtain US consensus on information needs from the affected reactors at Daiichi. This document differed from the FY14 report because of its different objectives. As observed in Section 1.3, the FY15 report was developed to not only identify a prioritized list of information that was of interest to US stakeholders but also to identify why it was of interest and how such information would benefit the US nuclear enterprise. In addition, preliminary cost and schedule information for near term tasks (i.e., tasks that should be started within the next five years) are included in this report. As discussed in Section 4, cost and schedule information was obtained by working with TEPCO to understand if such information was already available or planned to be obtained as part of their D&D efforts and what additional effort would be needed.

The last document important to this process is the final version of this report. This final report was issued in August 2015 after being vetted by the severe accident expert panel at a May 2015 meeting. There are other important considerations (shown in yellow boxes) that contributed to this approach. The first consideration relates to other synergistic efforts that were discussed in Section 1.1.2. These other efforts, including those funded by DOE and those organized by other agencies, were considered in developing this report. In addition, results from forensic activities support several aspects of these other synergistic efforts. The second consideration relates to interactions with other stakeholders that affect the feasibility of proposed forensics activities. These stakeholders are identified because it is recognized that the success of this effort requires that information in this report be discussed with and supported by representatives of these organizations.

4. RESULTS

As discussed in Section 3, two expert panel meetings were held during FY15 to develop consensus input for information needs. These meetings were held in January 2015 (at ANL in Argonne, IL) and in May 2015 (at DOE Headquarters in Washington, DC).

Prior to the first expert panel meeting, two activities were completed to facilitate achievement of the meeting objectives. Activity 1 was to review and tabulate data needs identified by experts at the November 22, 2013 meeting. These data needs, which were identified in a preliminary report issued by DOE-NE (see Appendix C), were grouped according to location. In addition, preliminary information was provided in this report related to why such information was of interest and how such data could be obtained. No effort was made in the FY14 report to identify the costs associated with obtaining such data or how to interface these data needs with current TEPCO plans for D&D of the affected reactors. Nevertheless, the information in this report laid the foundation for FY15 efforts to develop this report. Activity 2 was completed by TEPCO engineers and provided to US experts during the meeting.[54] For each tabulated data need, TEPCO provided information related to what was already available and additional information that will be obtained as part of their planned D&D efforts. Information provided by TEPCO is summarized in tables found in Appendix B of this document. As discussed in Section 2, much of the information desired by US severe accident experts will be (or is already being) obtained as part of TEPCO's efforts to D&D these affected reactors. Hence, this effort by TEPCO engineers provided much needed and timely information for this US effort. After the first expert panel meeting, a draft report was prepared and submitted to expert panel members and other meeting attendees for review.

Prior to the second expert panel meeting, two activities were completed to facilitate meeting objectives being achieved. Activity 3 pertained to documentation of results from the first expert meeting. As noted above, a draft version of this report was issued after the January expert meeting. The draft report identified information needs, consensus findings and recommendations, and near-term R&D tasks. Initial comments on the draft report were documented for discussion at the May meeting. Activity 4 related to gaining consensus on specific areas for investigation in FY16. To gain consensus on a path forward, a process was developed using available examination information for Daiichi Unit 1. This information was distributed in advance to the group and discussed during the meeting.

This section summarizes results, findings, and recommendations from these meetings. As noted above, more detailed information may be found in appendices to this report. Specifically, Appendix A provides lists of attendees and agendas from expert panel meetings held during FY15. Appendix B provides tables with detailed information needs developed during the January 2015 expert meeting. Appendix C contains results from an expert meeting held in FY14 that provided initial input to this effort. Appendix E lists information needs provided by industry and identifies locations where these information needs have been incorporated into Appendix B.

4.1 Detailed Data Information Needs

During the January 2015 expert meeting (see Appendix A), meeting participants reviewed the Activity 1 tabulated data needs and Activity 2 input provided by TEPCO. Prior to and after the May 2015 expert meeting, participants again reviewed the information needs and provided comments. Detailed results from this effort are documented in Appendix B tables. As indicated in these tables, the review resulted in revisions to the data need information found in the May 2014 report. Revisions were due to expert panel activities to:

- identify omissions and delete nonessential items;
- consider additional information provided by TEPCO, such as the Activity 2 input describe above and information in TEPCO presentations [44 through 52];

- combine items so that information needs are logically organized with respect to TEPCO's roadmap and location where information would be obtained;
- document anticipated methods for obtaining data, such as videos or photos, sampling, destructive examinations, etc.;
- document why information is needed and expected use;
- note any D&D activities that might preclude data needs from being obtained (none were identified);
- provide preliminary information related to schedule (based on the TEPCO roadmap) and cost (based on TEPCO input related to what information is already available and current D&D plans) for near term tasks (i.e., tasks that should be initiated within the next five years).

Although details varied for each information need, tables in Appendix B generally indicate information is desired to answer fundamental questions related to how the accident progressed in each unit, understand equipment and component survivability, and benchmark severe accident progression and dose assessment codes. Ultimately, such information would be used to enhance such severe accident /dose assessment codes, improve staff training and accident management guidelines, and possibly provide a basis for plant modifications. An important point emphasized by the experts is that this information is important for BWRs and PWRs; i.e., many of the insights would not only be applicable to BWRs, but also could have potential significant impacts for enhancing PWR safety.

Most of the information needs identified by the expert panel are related to the affected units at Daiichi (e.g., Units 1, 2, 3, and/or 4). These needs are organized according to location (e.g., the RB, the PCV, and the RPV) in Appendix B tables and the applicable units for each information need are identified with information requested by TEPCO (e.g., how information should be obtained, why it is needed, its expected use or benefits, when it should be done, and the estimated level of effort).

Table 4 summarizes activities identified by the expert panel for addressing information needs from the affected units at Daiichi (Appendix B tables identify specific units for each information need). This summary table identifies the desired examination information and the priority of the type of examination for each information need (based upon the number of asterisks in each box). The expert panel agreed that some information is required for all identified information needs to obtain a complete picture of the events. The information needs represent the entire accident progression and conditions that occurred during the events. Although Reference 8 prioritizes knowledge needs related to severe accident phenomena, this prioritization is based upon the need to first understand equipment performance and in-vessel behavior before one can address uncertainties related to late phase accident progression. In the case of forensic examinations, the expert panel concluded that information needs would be best prioritized with respect to 'cost' and the logical sequence for obtaining such information. The results of this prioritization, which are shown in Table 4, indicate (by the number of asterisks), that the expert panel typically placed the most emphasis upon information that could be obtained from visual examinations, such as videos and photographs. In general, the consensus was that such information was more easily obtained and could provide critical information related to whether additional examinations were required.

Table 4. Summary of proposed activities

Region	Examination Information Classification[d]			
	Visual	Near-Proximity	Destructive	Analytical
Reactor Building				
RCIC	****	***	**	
HPCI	****		***	
Building	****	***	**	*
Primary Containment Vessel				
MSL and Safety Relief Valves (SRVs)	****		***	
Drywell (DW) Area	****	***	**	*
Suppression Chamber	****	***		
Pedestal / RPV-lower head	****		***	**
Instrumentation		****	***	
Reactor Pressure Vessel				
Upper Vessel Penetrations	****		***	**
Upper Internals	****	***	**	*
Core Regions & Shroud	****		***	**
Lower Plenum	****		***	**

In the May 2015 expert panel meeting, it was emphasized that there were advantages associated with obtaining information from Daiichi units that did not experience the damage observed at Units 1, 2, 3, and 4 and from units at the Daini plant.[55] In particular, it was emphasized that information from components and fuel assemblies that did not experience such damage was useful because it would provide insights about repair and maintenance and increase confidence in the safety-enhancing strategy proposed by the US nuclear industry strategy – known as "diverse and flexible mitigation capability" or FLEX for accident mitigation. In addition, it was observed that such information could be more easily obtained at a lower cost and with less radiation exposure to personnel. Specific US industry requests for information from units at Daiichi and Daini related to coatings, instrumentation, containment and reactor pressure vessel performance, and RCIC performance are compared in Table E-1 of Appendix E. Table E-1 also identifies how these requests were incorporated into Tables B-1 through B-4 of Appendix B.

Table B-4 of Appendix B also identifies R&D tasks that are not related to any particular unit at Daiichi or Daini. For example, one task is to establish a US point-of contact (POC) that can coordinate US inquiries to TEPCO, be familiar with information obtained by TEPCO from the Daiichi and Daini sites, and distribute this information in an easy-to-read format. Another task is to perform periodic evaluations of

[d] ***Examination Classification Examples:***

Visual– Videos, Photographs, etc.

Near-Proximity– Radionuclide Surveys, Seismic Integrity Inspections, Bolt Tension Inspections, and Instrumentation Calibration Evaluations

Destructive– System or Component Disassembly, Sampling, etc.

Analytical– Chemical Analysis, Metallurgical Analysis, Gamma Scanning, etc.

severe accident code models and ensure that obtained data are considered and incorporated into code models, as needed.

Schedule and cost (or level of effort) information identified by the expert panel differed according to information need. The schedule information was derived from the Mid-and-Long-Term Roadmap (see Appendix D). However, an important point is that much information is already available and that efforts should immediately be initiated to assess if available information is sufficient to address the identified need (and make additional requests, if required). Selected examples of such information are identified in Section 2.3 of this report, and more were discussed at the May 2015 expert panel meeting. The expert panel concluded that costs were primarily associated with the effort required to evaluate information already obtained by TEPCO or information that would be obtained as part of planned D&D activities. Although there were specific examples where TEPCO had not planned to obtain selected information, such cases were often due to the fact that the information had not yet been considered. Furthermore, future events may affect the need for, and current decisions related to, the desired information. For example, as indicated in Appendix B, several reactor building information needs are affected by TEPCO's ability to drain the torus. The feasibility of obtaining such information is much more difficult if it must be obtained by underwater examinations. Likewise, desired ex-vessel information needs may be eliminated if it is determined that the vessel remained intact. Nevertheless, the expert panel agreed that it was important to document all information needs so that decisions could be made as additional information (and funding) become available.

4.2 Findings and Conclusions

During the January 2015 meeting, the expert panel agreed upon several conclusions and findings. Because of their importance, these points are listed in this section:

- Information obtained from the affected reactors at Daiichi provides a unique means to obtain full-scale, prototypic data for enhanced understanding and safety (e.g., improved severe accident guidance, possible plant modifications, improved simulation codes for staff training, etc.).
- Insights gained from collecting and comparing similar observations and data from each of the three units are valuable because the accident progression at each of the three units was unique in many respects.
- This information is important for BWRs as well as PWRs. Insights gained from this information are not only applicable to BWRs, but also could have potential impacts for enhancing PWR safety.
- Some information is required for all identified items to obtain a complete picture of the events. It is only meaningful to prioritize data needs with respect to the 'cost' and 'logical sequence' for obtaining such information.
- Information from other units at Daiichi and other plants, such as the Daini plant, also provide valuable insights for forensics, repair, maintenance, and field applications. Critical information from these plants can be more easily obtained at lower costs and with less radiation exposure to personnel.
- TEPCO D&D plans (or activities already completed) address much of the information identified by the US expert panel.
- Ultimately, an international framework should be established to benefit from information obtained during TEPCO's D&D efforts at Daiichi.
- Some information/data are already available or are being obtained now. Several tasks should be immediately initiated if the US nuclear enterprise is to benefit from this information and future information obtained from TEPCO. These near term tasks (e.g., tasks that should be initiated within the next five years) are outlined in Section 4.3.

The expert panel recommended that these points be emphasized in future discussions with stakeholders and decision makers that can affect program funding and the feasibility of completing R&D tasks.

4.3 Recommendations for Near Term R&D Tasks

An important objective of this document is to provide planning input with tasks and estimated level of effort for US support of this examination effort. During the expert panel meetings, tasks were identified that should be started within the next five years. These tasks, with preliminary cost (e.g., labor and non-labor level of effort per year) and schedule information, are summarized in Table 5.

Table 5. Near-term tasks with preliminary level of effort and schedule estimates

Task			Schedule		Cost (Level of Effort per Year)	
ID	**Name**	**Description**	**Start**	**End**	**Labor (FTE)**	**Materials/Travel**
1	US Point -of - contact (POC)	Establish US POC to review available TEPCO/IRID information, interact with TEPCO, and extract existing information from data sources. Provide in easy-to-read format for US expert review. Interact with US organizations and remain cognizant of related international efforts. Coordinate annual program reviews to update information needs (as needed).	Now	FY2020 (or longer)	1	1 domestic trip (for program reviews) and 1 international trip (to Japan)
2	Information Evaluations	Cognizant experts review information for consistency and adequacy, provide additional information requests (if needed), draw reactor safety insights, and post results in easy-to-read format and easy-to-access location for global access. Selected areas are: • Component Inspection • Radiological Sampling and Swiping • Core Debris Location Evaluations Document results in Task 1 annual report.	FY2016	FY2020 (or longer)	1 (various experts) + 0.25 (website interface)	2 domestic trips (for program reviews)
3	Code Evaluations of New Information	Review reactor examination information for implications to severe accident/dose assessment codes and work with responsible organizations to incorporate new information into code models and provide feedback on recommended forensics (as needed).	On-going	FY2020 (or longer)	TBD[e]	TBD[e]
4	Obtain Detailed Inspection Information	Conduct new survey/workshops to review results and update information inspection needs by industry with expert input (e.g., instrumentation, structure survivability, etc.). Document results in Task 1 annual report.	FY2016	FY2020 or later	1 (split amongst participants)	8 domestic trips (for new expert workshop)
5	Facilitation of Reactor Examinations	Provide advanced technology and /or funding to facilitate examinations and sample removal to address information needs or field deployment means of new technology. Document results in Task 1 annual report.	FY2017	FY2020 or later	1	~$300 K Materials 2 domestic trips (for program reviews)

As noted in Section 4.1, the expert panel agreed that some information is required for all identified information needs to obtain a complete picture of the events. In the case of forensic examinations, the expert panel concluded that forensic examinations would be best prioritized with respect to 'cost' (or level

[e] It is anticipated that organizations responsible for development and maintenance of codes used in the evaluations of new information would fund these activities separately; on this basis, no attempt has been made to quantify effort associated with this task.

of effort) and the logical sequence for obtaining such information. As noted in Table 5, three areas were identified as higher priority near-term R&D activities:

- Component Inspection (based on industry prioritized list and code analysis)
- Radiological Sampling and Swiping
- Core Debris Location Evaluations

These areas were selected based on results from Activity 4. Prior to the May 2015 meeting, available TEPCO examination information was identified that could address each information need identified for Daiichi Unit 1 in Table B-1. At the May 2015 meeting, experts identified the first two areas because of the abundance of component photograph and video examination information and radiation dose survey information being obtained from the affected units at Daiichi. In addition, because of on-going TEPCO D&D activities to use muon tomography and robots to identify the endstate of debris within the affected units, a third area was identified as a high priority near-term R&D activity. Additional details related to information needs on which FY16 efforts will focus upon in each area are provided in this section.

4.3.1 Component Inspections

Examinations of components and systems within the RB, PCV, and RPV provide critical information related to their survivability, operability, and peak conditions (e.g., pressure and temperature) they experienced during the accident. Damage incurred from hydrogen explosions and temperature indicators can provide insights related accident progression during these events. As observed in Reference 2, component examinations in the TMI-2 containment provided critical evidence of peak temperatures and pressures when instrumentation data were inconsistent.

TEPCO information indicates that component photographic and video examinations are already underway to support D&D efforts. As indicated in Figure 15, important insights may be obtained from component examinations at Unit 1; similar information is available for other affected units. It is critical that such information be collected and evaluated now before it is destroyed by on-going D&D efforts. As noted in Appendix B tables, photos and videos should be obtained with a reference length (ruler) at appropriate locations (with the exception of general area views). Appendix B tables note where such reference lengths are of special importance, enabling the nuclear enterprise to obtain the most benefit from examinations.

In light of limited funding resources, US evaluations of components will be prioritized considering industry input based on assumptions related to the selection of FLEX equipment and informed by the need to reduce uncertainties in severe accident code evaluations. Input provided from industry for component examinations at Daiichi and Daini is included in Table E-1 of Appendix E. As noted in Table E-1, this information will provide industry important insights related to the adequacy of FLEX equipment, system and component survivability (or failure modes), the progression of the accident, and peak conditions incurred during the accident. Table E-1 also identifies where this information need is located in Appendix B tables.

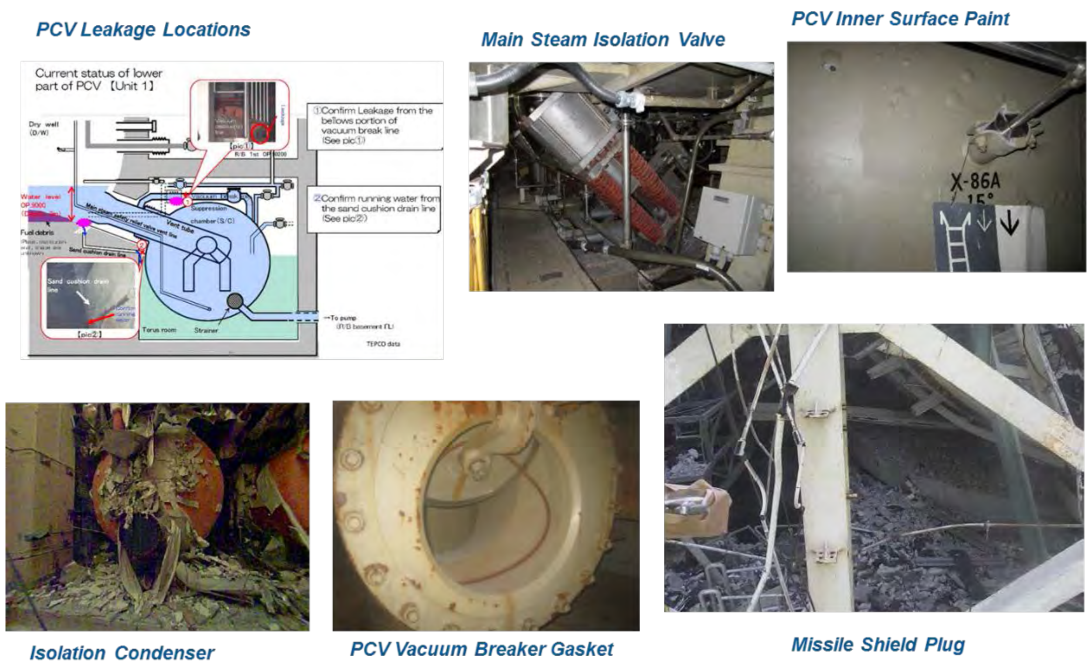


Figure 15. Representative Unit 1 component examination information. (Courtesy of TEPCO)

If this examination area is funded in FY16, the following tasks will be completed:

- Available TEPCO examination information will be listed that could address each information need identified in Appendix B for Units 1, 2, and 3.
- Lists for components and available information will be distributed to TEPCO, component inspection expert panel for completeness and prioritization (S. Kraft, NEI; B. Williamson, TVA; B. Lutz, PWROG; R. Linthicum; PWROG; R. Bunt; Southern Nuclear; and D. Peko, DOE-NE), and severe accident code modeling experts (R. Gauntt and D. Kalinich, SNL; M. Francis and K. Robb; ORNL; J. Gabor and D. Luxat, Erin; C. Paik, M. Plys, W. Luangdilok; FAI)
- Available information will be collated and provided to the component inspection expert panel. This panel will request additional examination information (if needed) and document results in a white paper with conclusions related to component performance, increased understanding of events, and reactor safety impacts.
- The white papers will be discussed and presented at the FY16 forensics expert panel meeting.
- The results will be documented in the Task 1 annual report.

As indicated above, the expert panel has determined that this component examination information offers the potential to obtain critical safety insights with respect to accident progression at the affected Daiichi units and with respect to proposed accident mitigation strategies in the US.

4.3.2 Radiological Sampling and Swiping

An important category of data acquisition from the decommissioning activities is collecting radionuclide deposition samples in the RB, PCV, and SFP. These samples or swipes can provide evidence of fission product release fractions and possibly on fission product speciation. As indicated in Figure 16, TEPCO is already obtaining radiological sampling and swipes to support D&D efforts for Unit 1 and other affected units.

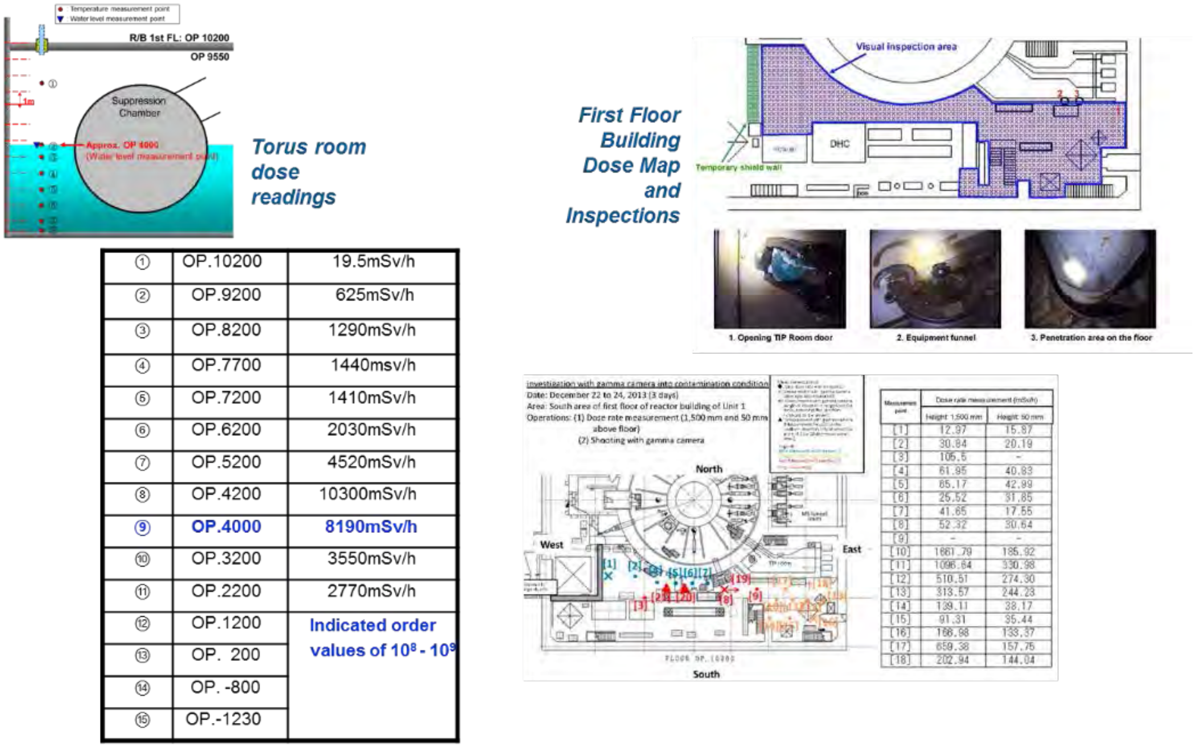


Figure 16. Representative Unit 1 dose reading information. (Courtesy of TEPCO)

For this effort, it is proposed to collect samples of deposited radioactivity (e.g., smears or swipes) from a significant number of different locations in the PCV, the RB, and the vent lines for each of the three damaged units. Samples should be obtained from both vertical and horizontal surfaces that are accessible from the robotic tools being used by TEPCO. These should be collected in much the same manner as a radiation technician would obtain samples to measure for surface contamination.

Although it has been over four years since the accident and the shorter lived radionuclides will have decayed to daughter radioactive products or stable species, it should be possible to back-out the species that were deposited at the time of the accident. It may also be possible to determine the compounds that the radioactive species may have formed (e.g., oxides, hydroxides, ionic salts, etc.) at the time of the accident. The deposition patterns and the chemical species of the radionuclides is important for accident management for validating (or enhancing) severe accident analysis code models of radionuclide behavior. For example, radioactive cesium is modeled primarily as an aerosol that is combined with either iodine or as a molybdate or hydroxide form.

A first step in this campaign is to use MELCOR and ORIGEN to estimate the fractional releases and depositions anticipated to have deposited in the reactor building. These release fractions will range between 0.1 for volatiles (Cs, Te, I) down to 10^{-5} for lower volatiles. These fractions can be translated to expected quantitative depositions per unit area and therefore, absolute quantities that can reasonably be expected from sample extractions. Next, the radioactive decay that has taken place must be evaluated. This requires very good signatures for Cs 134 and Cs 137, which can be expected with good radioassay counting statistics. However, we are also very interested in isotopes of Ba, Sr, La, and Ce. Many of these are of intermediate activity and will not be detectable by radioassay directly. However, radioactive or stable daughter products in the decay chain may be present and other assay methods also may be useful, including radioassay, wet chemistry or neutron activation analysis.

The RBs from Units 1, 2, and 3 may also yield good data, as may previously obtained SFP water samples. [16, 17] The SFP water samples are thought to constitute a fingerprint of the releases from the damaged reactor; i.e., the reactor building would have been fumigated with radioactive releases that were subsequently deposited by gravitational settling into each respective unit SFP. Samples from within the PCV will likely be of little use owing to the relentless rain, fog, and washout of the walls and surfaces that has been ongoing for the past several years. However, very limited sampling may be useful to determine if any strongly adherent species are present on vertical surfaces.

The campaign will include US efforts to provide instructions for detecting desired fission product information that consider detection limits and assay method capabilities, given expected fractional releases and depositions.

If this examination area is funded in FY16, the following tasks will be completed:

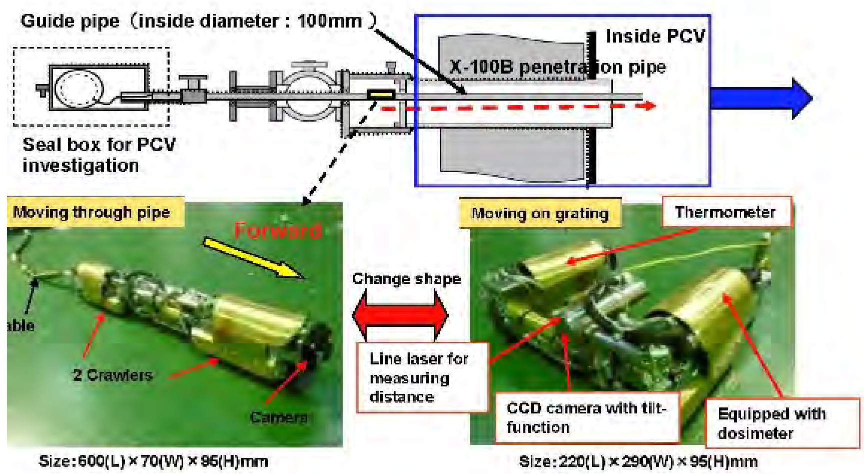
- Available TEPCO radiological survey and sampling information will be listed that could address each information need identified in Appendix B.
- Lists will be distributed to TEPCO, radiological survey and sampling expert panel for completeness and prioritization (R. Gauntt and D. Kalinich, SNL; M. Farmer, ANL; M. Francis, ORNL; and B. Lutz, PWROG), and severe accident code modeling experts (R. Gauntt and D. Kalinich, SNL; M. Francis and K. Robb; ORNL; D. Luxat, Erin; and W. Luangdilok; FAI)
- Available information will be collated and provided to the radiological survey and sampling inspection expert panel. This panel will request additional examination information (if needed) and document results in a white paper with conclusions related to component performance, increased understanding of events, and reactor safety impacts.
- The white papers will be discussed and presented at FY16 forensics expert panel meeting.
- The results will be documented in the Task 1 annual report.

As indicated above, the expert panel has determined that this radionuclide survey and sampling information offers the potential to obtain critical safety insights with respect to accident progression at the affected Daiichi units and with respect to proposed accident mitigation strategies in the US.

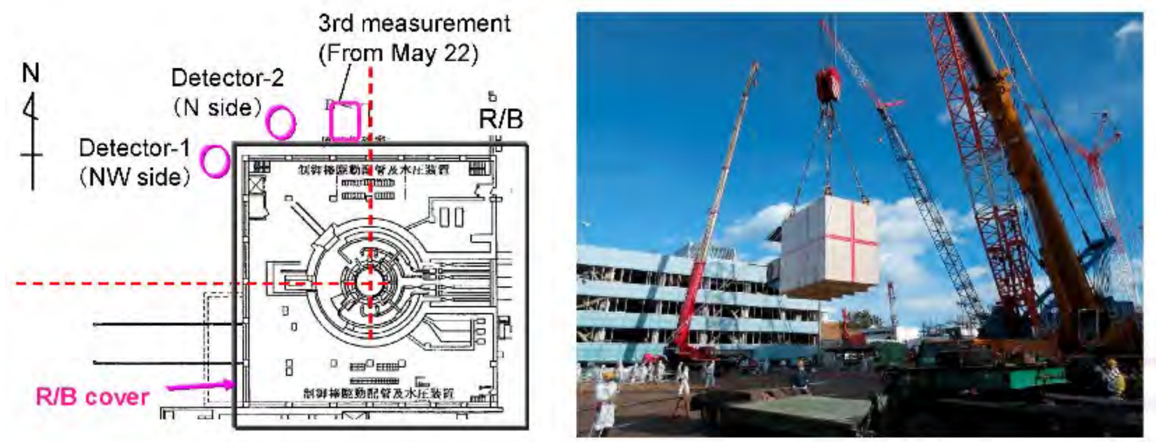
4.3.3 Debris Endstate Location

Post-accident examinations at TMI-2 [56] demonstrated that the endstate of debris is an important finding from forensics inspections and critical for developing and validating models within severe accident analysis codes. Debris endstate location information is of particular interest at Daiichi because comparisons can be made between the multiple units that were affected.

As indicated in Figure 17, TEPCO has been deploying robots and muon tomography systems to gain insights about the endstate location of debris in Unit 1, and additional evaluations are planned for Units 2 and 3.[51, 52] The IRID roadmap indicates that additional information will be obtained as part of D&D efforts (see Appendix D).



(a)



(b)

Figure 17. Unit 1 robot (a) and muon tomography (b) examinations. (Courtesy of TEPCO)

Appendix B tables provide additional details about information needs identified by US experts related to debris endstate location. Clearly, the first need is to obtain definitive data indicating if debris has relocated to ex-vessel locations and, if this has occurred, to understand the mode of vessel failure (e.g., whether global, localized, or penetration failures occurred).[57] In addition, it is desired to gain insights about debris coolability, the effects of saltwater, and debris spreading from examinations.

If this examination area is funded in FY16, the following tasks will be completed:

- Available TEPCO examination information pertaining to debris endstate location will be listed that could address each information need identified in Appendix B.
- Lists will be distributed to TEPCO and debris endstate location expert panel for completeness (M. Farmer, ANL; R. Gauntt, SNL; K. Robb; ORNL; J. Gabor, Erin; C. Paik and M. Plys, FAI)
- Available information will be collated and provided to the debris endstate location expert panel. This panel will request additional examination information (if needed) and document results in a white paper with conclusions related to component performance, increased understanding of events, and reactor safety impacts.

- The white papers will be discussed and presented at FY16 forensics expert panel meeting.
- The results will be documented in the Task 1 annual report.

As indicated above, the expert panel has determined that debris endstate location information offers the potential to obtain critical safety insights with respect to accident progression at the affected Daiichi units and with respect to proposed accident mitigation strategies in the US.

4.4 Future Activities

Upon issuance of this report, several activities are envisioned. As noted in Figure 14, activities identified in this document and expected benefits to LWR safety will be socialized with stakeholders from Japan (e.g., TEPCO, IRID, NDF, etc.) and from the US (e.g., DOE, NRC, State Department, and Congress). Based upon funding allocations, activities listed in Table 5 will be initiated. In particular, activities will be initiated to establish a POC (Task 1), to initiate examination information evaluations (Task 2), and to conduct periodic code assessments (Task 3). As noted in Sections 4.2 and 4.3, forensic examination results will be presented and discussed at future expert panel meetings. As results from evaluations associated with forensic examinations are found to benefit reactor safety, it is anticipated that this effort will grow. Ultimately, it is anticipated that activities may be conducted as part of an international framework.

5. SUMMARY

As a first step toward ensuring that the US obtains the maximum benefit from information obtained from the affected units at Daiichi during D&D, this report was developed using the approach shown in Figure 18. This objective of the report is to document consensus input from US experts for prioritized time-sequenced examination information and supporting R&D activities that could be completed with minimal disruption of planned TEPCO D&D activities for Daiichi. The report describes why such information is important and how it will be used to benefit the US nuclear enterprise. Finally, this report prioritizes near-term R&D tasks that should be funded by DOE-NE so that the US nuclear enterprise is to benefit from the examination information obtained from the affected reactors at Daiichi.

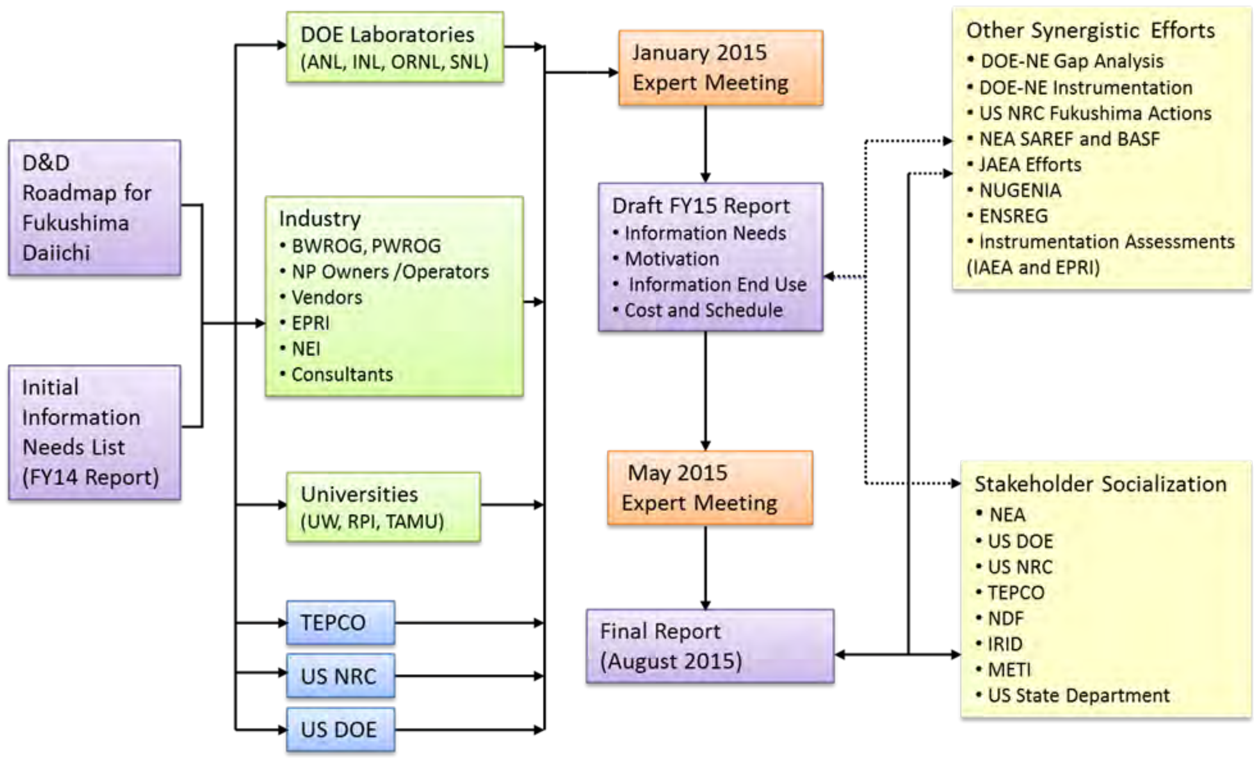


Figure 18. Approach used to develop US input.

As shown in Figure 18, this effort included three meetings with experts participating from industry, universities, and national laboratories. Experts from US government organizations (the NRC and the DOE) and from TEPCO also attended and informed participants during these meetings. Efforts were also made by meeting organizers to be cognizant of other synergistic efforts that could inform meeting participants. Draft reports summarizing meeting results were prepared and distributed to participants for review, and review comments were incorporated into subsequent documents. Clearly, this effort to develop consensus has strived to include a broad spectrum of stakeholder input.

During these meetings, US experts agreed upon several significant findings and recommendations:

- Information obtained from the affected reactors at Daiichi offers a unique means to obtain full-scale, prototypic data for enhancing reactor safety (e.g., improved severe accident guidance, possible plant modifications, improved simulation codes for staff training, etc.).

- Insights gained from collecting and comparing similar observations and data from each of the three units are valuable because the accident progression at each of the three units was unique in many respects.
- This information is important for BWRs and PWRs; i.e., many insights gained from this information are not only applicable to BWRs, but also could have significant impacts for enhancing PWR safety.
- Some information is required for all identified items to obtain a complete picture of the events. It is only meaningful to prioritize data needs with respect to the 'cost' and 'logical sequence' for obtaining such information.
- Information from other units at Daiichi and other plants, such as the Daini plant, also provide valuable insights for forensics, repair, maintenance, and field applications. Critical information from these plants can be more easily obtained at lower costs and with less radiation exposure to personnel.
- TEPCO D&D plans (or activities already completed) address much of the information identified by the US expert panel.
- Ultimately, an international framework should be established to benefit from information obtained during TEPCO's D&D efforts at Daiichi.
- Some information/data are already available or are being obtained now. Several tasks should be immediately initiated if the US nuclear enterprise is to benefit from this information as well as future information to be obtained from TEPCO.

Table 6 summarizes activities for addressing information needs from the affected units at Daiichi that were identified by the expert panel. This summary table identifies the type of examinations desired and the type of examination information desired. As noted above in the fourth item, the expert panel concluded that some information is required for all identified data needs to obtain a complete picture of the events. Hence, the expert panel concluded that information needs would be best prioritized with respect to 'cost' or ‘level of effort’ and the logical sequence for obtaining such information. The results of this prioritization indicate (by the number of asterisks) that the expert panel placed the most emphasis upon information that could be obtained from visual examinations, such as videos and photographs. The consensus was that such information was more easily obtained and could provide critical information related to whether additional examinations were required.

With respect to the last item, the expert panel recommended that DOE-NE fund the near-term tasks listed in Table 7. These tasks were identified as higher priority tasks that should be initiated within the next five years if the US nuclear enterprise is to benefit from the examination information obtained from the affected reactors at Daiichi. Preliminary cost (level of effort) and schedule information for each task are found in Section 4. Schedule information is based on information published in the Mid-and-Long-Term Roadmap. Although the expert panel agreed that some information is required for all identified information needs to obtain a complete picture of the events, the panel also recognized that forensic examinations activities must be prioritized. As noted in Task 2 of Table 7, the expert panel identified three areas as higher priority near-term R&D activities:

- Component Inspection (based on industry prioritized list and code analysis)
- Radiological Sampling and Swiping
- Core Debris Location Evaluations

Section 4.3 provides additional details related to activities envisioned on the higher-priority near-term areas where forensics information will be evaluated.

In summary, this report provides a basis for obtaining stakeholder support and establishing appropriate funding for obtaining this information. In many cases, information needs identified in this effort fall within international programs. It is anticipated that the US will participate in such international programs. In other cases, information may be obtained from Japanese activities. Hence, this report also provides a basis for ensuring that there is no duplication of US efforts related to examination information from Daiichi.

Table 6. Summary of proposed activities

Region	Examination Information Classification[f]			
	Visual	Near-Proximity	Destructive	Analytical
Reactor Building				
RCIC	****	***	**	
HPCI	****		***	
Building	****	***	**	*
Primary Containment Vessel				
MSL and SRVs	****		***	
DW Area	****	***	**	*
Suppression Chamber	****	***		
Pedestal / RPV-lower head	****		***	**
Instrumentation		****	***	
Reactor Pressure Vessel				
Upper Vessel Penetrations	****		***	**
Upper Internals	****	***	**	*
Core Regions & Shroud	****		***	**
Lower Plenum	****		***	**

[f] ***Examination Classification Examples:***

Visual– Videos, Photographs, etc.

Near-Proximity– Radionuclide surveys, Seismic Integrity Inspections, Bolt Tension Inspections, and Instrumentation Calibration Evaluations.

Destructive– System or Component Disassembly, Sampling, etc.

Analytical– Chemical Analysis, Metallurgical Analysis, Gamma Scanning, etc.

Table 7. Near-term tasks recommended by the expert panel

Task			Schedule		Cost (Level of Effort per Year)	
ID	**Name**	**Description**	**Start**	**End**	**Labor (FTE)**	**Materials/Travel**
1	US Point -of - contact (POC)	Establish US POC to review available TEPCO/IRID information, interact with TEPCO, and extract existing information from data sources. Provide in easy-to-read format for US expert review. Interact with US organizations and remain cognizant of related international efforts. Coordinate annual program reviews to update information needs (as needed).	Now	FY2020 (or longer)	1	1 domestic trip (for program reviews) and 1 international trip (to Japan)
2	Information Evaluations	Cognizant experts review information for consistency and adequacy, provide additional information requests (if needed), draw reactor safety insights, and post results in easy-to-read format and easy-to-access location for global access. Selected areas are: • Component Inspection (based on industry prioritized list and code analysis) • Dose Measurements and Smears for Isotopic Concentration Evaluations (based on code analysis evaluations, etc.) • Core Debris Location Evaluations Document results in Task 1 annual report.	FY2016	FY2020 (or longer)	1 (various experts) + 0.25 (website interface)	2 domestic trips (for program reviews)
3	Code Evaluations of New Information	Review reactor examination information for implications to severe accident/dose assessment codes and work with responsible organizations to incorporate new information into code models and provide feedback on recommended forensics (as needed).	On-going	FY2020 (or longer)	TBD[g]	TBD[g]
4	Obtain Detailed Inspection Information	Conduct new survey/workshops to review results and update information inspection needs by industry with expert input (e.g., instrumentation, structure survivability, etc.). Document results in Task 1 annual report.	FY2016	FY2020 or later	1 (split amongst participants)	8 domestic trips (for new expert workshop)
5	Facilitation of Reactor Examinations	Provide advanced technology and /or funding to facilitate examinations and sample removal to address information needs or field deployment means of new technology. Document results in Task 1 annual report.	FY2017	FY2020 or later	1	~$300 K Materials 2 domestic trips (for program reviews)

[g] It is anticipated that organizations responsible for development and maintenance of codes used in the evaluations of new information would fund these activities separately; on this basis no attempt has been made to quantify effort associated with this task.

6. REFERENCES

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

2. J. L. Rempe and D. L. Knudson, "Instrumentation Performance during the TMI-2 Accident," invited paper for ANIMMA 2013 Special Edition, *IEEE Transactions on Nuclear Science*, **61**, Issue 4, January 2014, pp 1963-1970.

3. R. O. Gauntt et al., *Fukushima Daiichi Accident Study*, SAND2012-6173 (2012).

4. D. Luxat and J. Gabor, *Fukushima Technical Evaluation: Phase 1 – MAAP5 Analysis*, EPRI Report No. 1025750 (2013).

5. D. Luxat, J. Hanophy, and D. Kalinich, *Modular Accident Analysis Program (MAAP) – MELCOR Crosswalk, Phase 1 Study*, EPRI Report No. 3002004449 (2014).

6. K. R. Robb, M. T. Farmer, and M. W. Francis, *Enhanced Ex-Vessel Analysis for Fukushima Daiichi Unit 1: Melt Spreading and Core-Concrete Interaction*, ORNL/TM-2012/455 (2013).

7. K. R. Robb, M. T. Farmer, M. W. Francis, *Ex-Vessel Core Melt Modeling Comparison between MELTSPREAD-CORQUENCH and MELCOR 2.1*, ORNL/TM-2014/1 (2014).

8. M. T. Farmer (editor), R. Bunt, M. Corradini, P. Ellison, M. Farmer, M. Francis, J. Gabor, R. Gauntt, C. Henry, R. Linthicum, W. Luangdilok, R. Lutz, C. Paik, M. Plys, C. , J. Rempe, K. Robb, R. Wachowiak, *Reactor Safety Gap Evaluation of Accident Tolerant Components and Severe Accident Analysis*, ANL/NE-15/4 (2015).

9. R. Lee, US NRC, "Thoughts on Fukushima Forensics on BWR Accident Analysis," Embedded Workshop on Fukushima," *Japan-U.S. Seminar on Two-Phase Flow Dynamics 2015*, Purdue University, West Lafayette, Indiana, May 13, 2015.

10. J. Uhle, "US NRC Regulatory Response to Fukushima," *Japan-U.S. Seminar on Two-Phase Flow Dynamics 2015*, Purdue University, West Lafayette, Indiana, May 13, 2015.

11. H. Nakamura, K, Arai, H. Olkawa, T. Fujii, S. Umezawa A. Ohnuki, Y. Nishi, Y. Abe, J. Sugimoto, S. Koshizuka, and A. Yamaguchi, "AESJ New Thermo-Hydraulics Roadmap for LWR Safety Improvement based on Lessons-Learned from Fukushima Daiichi Accident," *Japan-U.S. Seminar on Two-Phase Flow Dynamics 2015*, Purdue University, West Lafayette, Indiana, May 13, 2015.

12. US Department of State, "Facts on U.S.-Japan Commission on Civil Nuclear Cooperation", http://iipdigital.usembassy.gov/st/english/texttrans/2013/11/20131105285763.html#axzz3eIjx8UhK, last visited June 27, 2015.

13. Nuclear Energy Agency, "Senior Expert Group on Safety Research Opportunities Post-Fukushima," see https://www.oecd-nea.org/nsd/csni/saref.html, last accessed June 20, 2015.

14. Nuclear Energy Agency, "NEA Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF) Project," see https:// http://www.oecd-nea.org/jointproj/bsaf.html, last accessed June 20, 2015.

15. Bernd Jäckel, PSI "Activity Data Interpretation from Land Contamination of Fukushima Daiichi Accident, BSAF Phase II, Paris, France, 2014.

16. K. Tateiwa, "Spent Fuel and Spent Fuel Storage Facilities at Fukushima Daiichi," *presentation at "Lessons Learned from the Fukushima Nuclear Accident for Improving Safety and Security of U.S. Nuclear Plants: Phase 2," Meeting #3*, National Academies, Washington, DC, January 29, 2015.

17. K. Tateiwa, TEPCO, email to J. Rempe, Rempe and Associates, "IF Forensics Examination' Concrete Core Samples from Units 1 and 2 Reactor Building," with attachment, http://www.tepco.co.jp/nu/fukushima-np/roadmap/images/d130829_06-j.pdf, May 27, 2015.

18. NUclear GENeration II & III Association (NUGENIA), http://www.nugenia.org/, last accessed June 28, 2015.

19. European Nuclear Safety Regulators Group (ENSREG), http://www.ensreg.eu/, last accessed June 28, 2015.

20. N. P. Neureiter (Chair), et al., Committee on Lessons Learned from the Fukushima Nuclear Accident for Improving Safety and Security of U.S. Nuclear Plants, National Research Council of the National Academies, Lessons Learned from the Fukushima Nuclear Accident for Improving Safety of U.S. Nuclear Plants, The National Academies Press, Washington, D.C., USA (2014)

21. J. Niel (Chair), et al., NEA Committee on Nuclear Regulatory Activities (CNRA), B. Sheron (Chair), et al., NEA Committee on the Safety of Nuclear Installations (CNSI), and A. McGarry (Chair) et al., NEA Committee on Radiation Protection and Public Health (CRPPH), The Fukushima Daiichi Nuclear Power Plant Accident: OECD/NEA Nuclear Safety Response and Lessons Learnt, NEA No. 7161, Organisation for Economic Co-operation and Development, Paris, France (2013).

22. IAEA, "Accident Monitoring Systems for Nuclear Power Plants," IAEA Nuclear Energy Series Report, No. NP-T-3.16, (2015).

23. J. L. Rempe and D. L. Knudson, "Instrumentation Performance during the TMI-2 Accident," IEEE Transactions on Nucl. Sci., 61 (2014).

24. J. L. Rempe and D. L. Knudson, "Qualification of Data Obtained during a Severe Accident - Illustrative Examples from TMI-2 Evaluations," *9th International Conference on Nuclear Plant Instrumentation, Control & Human–Machine Interface Technologies* (NPIC & HMIT 2015), Charlotte, North Carolina, Feb 21-26, 2015.

25. D. Clayton, M. Poore, and D. Peko, "A Case Study for BWR Instrumentation and Control Systems Performance During a Severe Accident," *9th International Conference on Nuclear Plant Instrumentation, Control & Human–Machine Interface Technologies* (NPIC & HMIT 2015), Charlotte, North Carolina, Feb 21-26, 2015.

26. EPRI, "Technical Advisory Group I&C for Beyond Design Basis Events & Severe Accidents (BDBE &SA) Charter," January 17, 2014.

27. S. Lopez, R. Pedan, S. Beaudry, Pl Bluth, T. Ranson, T. Romero, N. Shore, and D. Vang, "Conceptual Report for Severe Accident Mobile Investigator (SAMI), *9th International Conference on Nuclear Plant Instrumentation, Control & Human–Machine Interface Technologies* (NPIC & HMIT 2015), Charlotte, North Carolina, Feb 21-26, 2015.

28. R. Lutz, R. Linthicum, N. Labarge, S. Pierson and J. Ford., "PWROG Severe Accident Management Technical Support Guidance for Instrumentation," *9th International Conference on Nuclear Plant Instrumentation, Control & Human–Machine Interface Technologies (NPIC & HMIT 2015),* Charlotte, North Carolina, Feb 21-26, 2015.

29. P. Ellison and B. Williamson, "Instrument Challenges Observed at Fukushima Daiichi from an Operations Point of View," *9th International Conference on Nuclear Plant Instrumentation, Control & Human–Machine Interface Technologies (NPIC & HMIT 2015)*, Charlotte, North Carolina, Feb 21-26, 2015.

30. D. L. Rahn, "Criteria for Enhancing Accident Monitoring Instrumentation to Address Lessons Learned from Fukushima," *9th International Conference on Nuclear Plant Instrumentation, Control & Human–Machine Interface Technologies (NPIC & HMIT 2015)*, Charlotte, North Carolina, Feb 21-26, 2015.

31. US NRC, "Recommendations for Enhancing Reactor Safety in the 21st Century," The Near-Term Task Force Review of Insight from the Fukushima Daiichi Accident," ML111861807 (2011).

32. US NRC, "Tier 3 Program Plans and 6-month Status Update in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," SECY-12-0095, July 13, 2012.

33. US NRC, "Fifth 6-month Status Update on Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," SECY-14-0046, April 17, 2014.

34. G. Ducos, P. G. Allinei, C. Roure, C. Rozel, N. Blanc De Lanaute, and G. Musoyan, "On-Line Fission Products Measurements during a PWR Severe Accident: the French DECA-PF Project," *submitted to the 2015 ANIMMA Conference*, Lisbon, Portugal, April 2015.

35. SA Keisou R&D Working Team (Hitachi-GE, Toshiba, and Mitsubishi), "Status of "R&D program for Advanced Instrumentation System for Severe Accidents in Nuclear Power Plants" in Japan" *presentation at the CNWG*, January 27-29, 2015, Argonne, IL.

36. H. Yamana, NDF, "Expectations on the JAEA to Meet Challenges, Consolidation and Concentration/Investigation and Application," Topics Fukushima No. 65, May 1, 2015, http://fukushima.jaea.go.jp/english/topics/pdf/topics-fukushima065e.pdf, last accessed June 28, 2015.

37. Nuclear Damage Compensation and Decommissioning Facilitation Corporation, "Technical Strategic Plant 2015 for Decommissioning of the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company, Towards Amendment of the Mid-and-Long-Term Roadmap in 2015," April 2015.

38. Nuclear Emergency Response Headquarters, Council for the Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station, *Mid-and-Long-Term Roadmap towards the Decommissioning of TEPCO-'s Fukushima Daiichi Nuclear Power Station Units 1-4*, Provisional Translation, June 27, 2013.

39. International Research Institute for Nuclear Decommissioning website, http://irid.or.jp/en, last accessed January 31, 2015.

40. Nuclear Emergency Response Headquarters, Council for the Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station, Individual R&D project (overall project plan) toward the decommissioning etc. of Tokyo Electric Power Company (TEPCO) Fukushima Daiichi NPS Units 1 to 4, June 27, 2013.

41. Progress Status and Future Challenges of the Mid-and-Long-Term Roadmap toward the Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station Units 1-4, Update, dated May 28, 2015.

42. Tokyo Electric Power Company, "Investigation Results of the Inside of Unit 1 PCV at Fukushima Daiichi Nuclear Power Station," October 10, 2012, http://photo.tepco.co.jp/en/date/2012/201210-e/121010-01e.html, last accessed February 22, 2015.

43. Tokyo Electric Power Company, "Result of the second investigation inside of Primary Containment Vessels, Unit 2, Fukushima Daiichi Nuclear Power Plant," March 26, 2012, http://photo.tepco.co.jp/en/date/2012/201203-e/120326-03e.html, last accessed February 22, 2015.

44. K. Tateiwa, "Fukushima Update," *presentation at the first DOE Meeting on Reactor Safety Gap Evaluation and Fukushima Forensics at Argonne National Laboratory*, January 7, 2015.

45. 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.

46. Y. Yamanaka, "Investigation Plan of Inside RPV," *presentation at the first DOE Meeting on Reactor Safety Gap Evaluation and Fukushima Forensics at Argonne National Laboratory*, January 7, 2015.

47. S. Suzuki, "Status of R&D Projects Related to Fuel Debris Retrieval," *presentation at IRID Annual Symposium 2014*, July 18, 2014.

48. K. Tateiwa, email, dated January 8, 2015, to Forensics Expert Panel, "Dose Map of 1F-3 Reactor Building Refueling Floor," (in Japanese), presentation, dated December 25, 2014, http://www.tepco.co.jp/nu/fukushima-np/roadmap/images/d141225_11-j.pdf, last accessed February 10, 2015.

49. K. Tateiwa, "Spent Fuel and Spent Fuel Storage Facilities at Fukushima Daiichi," presentation at the National Academies, "Lessons Learned from the Fukushima Nuclear Accident for Improving Safety and Security of U.S. Nuclear Plants: Phase 2," Meeting # 3, Washington, DC, January 29, 2015.

50. Y. Yamanaka, "Outline of the 3rd Progress Report on the Investigation of Unresolved Issues," *presentation at the second DOE Meeting on Fukushima Forensics*, Washington, DC, May 27, 2015.

51. Y. Yamanaka, "1F-1 PCV inside inspection results," *presentation at the second DOE Meeting on Fukushima Forensics*, Washington, DC, May 27, 2015.

52. D. Yamada, Y. Yamanaka, and S. Mizokami, "Progress for Applying Reactor Imaging Technology by Cosmic Ray Muon in Fukushima Daiichi," *presentation at the second DOE Meeting on Fukushima Forensics*, Washington, DC, May 27, 2015.

53. B. Williamson and P. Ellison, "Instrument Challenges Observed at Fukushima Daiichi – An Operations Point of View" *presentation at the second DOE Meeting on Fukushima Forensics*, Washington, DC, May 27, 2015.

54. K. Tateiwa, TEPCO, email, "Additional materials [Gap Analysis and Fukushima Forensics] TEPCO Presentations," dated January 8, 2015.

55. S. Kraft, NEI, email with attachment, "Industry Proposed Action List ", to J. Rempe, Rempe and Associates, LLC, dated June 15, 2015.

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

57. J. Rempe, L. Stickler, S. Chàvez, G. Thinnes, R. Witt, and M. Corradini, "Margin-to-Failure Calculations for the TMI-2 Vessel," *Nuclear Safety, Special Edition featuring papers from TMI-2 research,* **35**, No. 2, July-December 1994, p 313.

Appendix A

Meeting Agendas and Attendee Lists

January 8-9, 2015 Meeting Agenda

Forensics Meeting to Develop Prioritized Examination Plan

Meeting Agenda
January 8-9, 2015

Bldg. 208, Conference Room A-138
Nuclear Engineering Division
Argonne National Laboratory
9700 S. Cass Avenue
Argonne, IL 60439

Thursday, January 8th

1:00 PM	Welcome and Overview	TAL (UW)
1:15 PM	Background and Proposed Approach for Prioritization - Motivation /Objective of Inspection Program Plan - FY14 Program Plan and Inspection Matrix - Path Forward (Planned/Proposed FY15 & Beyond Activities) Discussion of Proposed Methods for Prioritization	Rempe
2:00 PM	*Break*	
2:15 PM	Discussion and Prioritization of Reactor Building Inspection Worksheet	All
4:00 PM	Discussion and Prioritization of In-Containment Inspection (start)	All
5:00	Adjourn	All

Friday, January 9th

8:30 AM	January 9 Recap	Rempe
8:45 AM	Discussion and Prioritization of In-Containment Inspection (completion)	All
9:45 AM	*Break*	
10:00 AM	Discussion and Prioritization of In-Vessel Inspection Activities	All
12:00 AM	*Lunch*	
12:15 PM	Prioritization of Identified Technology Development Activities to Support/Reduce Cost of Inspections	All
1:30 PM	Closeout/Path Forward: • Prepare draft report and presentation for stakeholders • Submit for review to Expert • Incorporate comments	All
3:00 PM	Adjourn	

January 8-9, 2015 Meeting Attendees

Number	Name	Organization
1	Jordi Roglans - Ribas	ANL
2	Mike Corradini	U. of Wisconsin
3	Joy Rempe	Rempe and Associates, LLC
4	Matthew Francis	ORNL
5	Randy Gauntt	SNL
6	Randy Bunt	Southern Nuclear
7	Jeff Gabor	Erin Engineering
8	Phil Ellison	GE Power and Water
9	Roy Linthicum	Exelon
10	Mitch Farmer	ANL
11	Richard Lee	NRC
12	Cristian Rabiti	INL
13	Chris Henry	Fauske & Associates
14	Wison Luangdilok	Fauske & Associates
15	Damian Peko	DOE
16	Kenji Tateiwa	TEPCO
17	Yasunori Yamanaka	TEPCO

May 27-28, 2015 Meeting Agenda

Reactor Safety Technology Expert Panel Forensics Meeting– 27-28 May 2015

Reactor Safety Technology Expert Panel Forensics Meeting

Meeting Agenda
May 27-28, 2015

DOE Forrestal Building, Conference Room 4A-104
Washington, DC

Wednesday, May 27th

8:30 AM	Welcome and Overview	M. Corradini, UW
8:45 AM	Background and Proposed Agenda	J. Rempe, Rempe and Associates, LLC
9:00 AM	DOE and NRC Insights	D. Peko or T. Miller, DOE R. Lee or S. Basu, NRC
9:15 AM	TEPCO Update	Y. Yamanaka, TEPCO
10:15 AM	*Break*	All
10:30 AM	BWROG Insights from Data Evaluations	B. Williamson, BWROG
11:15 AM	Report Review: Introductory Sections (Sections 1 through 3) Comments	All
12:15	*Lunch*	All
1:30 PM	Report Review: Appendix B Tables and Section 4 Results	All
2:45 PM	*Break*	All
3:00 PM	Report Review: Appendix B Tables and Section 4 Results, Findings, and Proposed Future Activities Discussion	All
5:30 PM	*Adjourn*	

Thursday, May 28th

8:30 AM	May 27th Recap	J. Rempe Rempe and Associates, LLC
8:45 AM	Report Review: Executive Summary Review Discussion and High Priority FY16 Future Activities - POC: High Value Information to Pursue - INL: Website Improvements Closeout/Path Forward for FY15: • Obtain signed co-author forms • Finalize report with Review Comments • Submit for final review to Expert Panel • Issue Final Report – July 2015	All
10:00 AM	*Break*	All

May 27-28, 2015 Meeting Attendees

Number	Name	Organization
1	Sudhamay Basu	US Nuclear Regulatory Commission
2	Randy Bunt	Southern Nuclear Company
3	Mike Corradini	University of Wisconsin
4	Mitch Farmer	Argonne National Laboratory
5	Jeff Gabor	Erin Engineering
6	Phil Ellison	BWR Owner's Group / General Electric
7	Randy Gauntt	Sandia National Laboratory
8	Don Kalinich	Sandia National Laboratory
9	Steve Kraft	Nuclear Energy Institute
10	Richard Lee	US Nuclear Regulatory Commission
11	Wison Luangdilok	Fauske and Associates, LLC
12	Bob Lutz	PWR Owner's Group
13	David Luxat	Erin Engineering
14	Tom Miller	US Department of Energy
15	Chan Paik	Fauske and Associates, LLC
16	Damian Peko	US Department of Energy
17	Cristian Rabiti	Idaho National Laboratory
18	Joy Rempe	Rempe and Associates, LLC
19	Kevin Robb	Oak Ridge National Laboratory
20	Kenji Tateiwa	Tokyo Electric Power Company
21	Rich Wachowiak	Electric Power Research Institute
22	Bill Williamson	BWR Owner’s Group
23	Daichi Yamada	Tokyo Electric Power Company
24	Yasunori Yamanaka	Tokyo Electric Power Company

Appendix B

Information Needs

Table B-1. Information Needs from the Reactor Building					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Cost/Level of Effort**
RB-1	Photos/videos[h] of condition of RCIC valve and pump before drain down and after disassembly (U2 and U3)	• 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.).	Currently flooded (requires underwater investigations unless drained down by TEPCO). As noted in Table E-1, inspections could be completed more easily at Daini.	Not currently considered by TEPCO; If torus not drained, requires underwater technology available.
RB-2	Photos/videos of HPCI System after disassembly (U1, U2, and U3)	• Gain insights about degradation.	Impacts AM strategies (equipment utilization).	Currently flooded (requires other alternatives for underwater investigations unless drained down by TEPCO).	Not currently considered by TEPCO; If torus not drained, requires underwater technology available.
RB-3a	Photos/videos of damaged walls and structures (U1)	• Determine mode of explosion in U1 compared to U3.	Understanding what happened. Potential BWR improvements; Impacts BWR AM strategies and code models (venting and interconnection between units); Potential PWR impacts (e.g., modeling, AM strategies, etc.).	When TEPCO goes into U1 and after debris removal.	TEPCO will obtain some information; Labor needed to evaluate and post information.[i]
RB-3b	Photos/videos of damaged walls and structures (U3)	• Determine mode of explosion in U3. • Gain insight about highly energetic explosions in U3 compared to U1.	Understanding what happened. Potential BWR improvements; Impacts BWR AM strategies and code models (venting and interconnection between units); Potential PWR impacts (e.g., modeling, AM strategies, etc.).	When TEPCO goes into U3 and after debris removal.	TEPCO will obtain some information; Labor needed to evaluate and post information.
RB-3c	Photos/videos of damaged walls and structures (U4)	• Determine mode of explosion in U4.	What happened? Potential BWR improvements; Impacts BWR AM strategies and code models (venting and interconnection between units); Potential PWR impacts (e.g., modeling, AM strategies, etc.).	When TEPCO goes into U4 and after debris removal.	TEPCO will obtain some information; Labor needed to evaluate and post information.
RB-4	Photos/videos of damaged walls and components and radionuclide surveys (U2)	• Cause of depressurization. • Cause of H_2 generation.	Impacts BWR AM strategies (equipment utilization and venting); Improved BWR code simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.).	Now.	TEPCO has dose distribution information; Labor needed to evaluate and post information.

[h] 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.

[i] The phrase, 'labor needed to evaluate and post information' is a truncated version of activities described in more detail in Section 4. Namely, labor is needed for cognizant experts to evaluate with respect to the adequacy of provided inspection information to address information need, request additional inspection information (e.g., photos, videos, samples, etc.), and draw insights with respect reactor safety, and post in an easy-to-read format for global use.

Table B-1. Information Needs from the Reactor Building					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Cost/Level of Effort**
RB-5	Radionuclide surveys (U1, U2, and U3)	• Leakage path identification. • Dose code benchmarks.	Understanding what happened. Improved BWR Accident Management (plant robustness, training, SAMG). Improved BWR code simulations and dose code benchmarks, Potential PWR impacts (e.g., modeling, AM strategies, etc.).	Now and later (as debris is removed).	TEPCO will obtain some information; Labor needed to evaluate data for website posting and for benchmark evaluations.
RB-6	Radionuclide surveys and sampling of ventilation ducts (U4)	• 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).	Now, if done externally; Debris and ductwork removal could impact getting data.	TEPCO not planning; Labor needed to obtain, evaluate, and post data.
RB-7	Isotopic evaluations of obtained concrete samples (U2)	• Code assessments. • Possible model improvements for building retention assumptions.	Improved BWR modeling and EP planning; cross check of RN surveys. Potential PWR impacts (e.g., modeling, AM strategies, etc.).	Now.	JAEA labor for evaluations; Labor needed to evaluate data and conduct benchmark evaluations
RB-8	Photos/videos and inspection of seismic susceptible areas (e.g., bellows, penetrations, structures, supports, etc. in U1, U2, U3, and U4)	• To confirm with data that there were no seismic-induced failures.	Improved plant robustness; observed differences between U1 and U3. Potential PWR impacts (e.g., similar penetrations).	Now and later (as debris is removed); Note that debris currently precludes data from being obtained.	Not currently considered by TEPCO; photos may already be available; Labor for seismic experts to identify and evaluate available information.
RB-9	DW Concrete Shield Radionuclide surveys (U1, U2, and U3 - after debris removed in U1 and U3)	• To understand leakage amounts and locations.	Improved AM strategies (Plant improvements, training, and education). Improved codes. Understanding what happened.	Now and later (as debris is removed).	TEPCO has photos and some RN surveys; more will be obtained. Labor needed to identify and evaluate available information.
	Photos/videos around mechanical seals and hatches and electrical penetration seals (as a means to classify whether joints were in compression or tension)	• Potential leakage paths for RN and hydrogen release.[j]	Improved AM strategies (Plant improvements for BWR and PWRS, which have similar seals). Improved codes. Understanding what happened.	Now and later.	TEPCO has photos and some RN surveys; more will be obtained. Labor needed to identify and evaluate available information.

[j] 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.

Table B-1. Information Needs from the Reactor Building					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Cost/Level of Effort**
RB-10	Photos/videos of U1 (vacuum breaker), U1, U2, and U3 PCV leakage points (bellows and other penetrations)	• Potential leakage paths for RN and hydrogen release.	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.	Now and later.	TEPCO has information available; Labor needed to identify and evaluate available information.
RB-11	Photos/videos and available information on U1, U2, and U3 containment hardpipe venting pathway, standby gas treatment system and associated reactor building ventilation system	• To assess performance of seals under high temperature and radiation conditions.[k]	Improved AM strategies (Plant improvements). Improved understanding of events.	Now and later.	Not currently considered by TEPCO; some photos may already be available. Labor needed to identify and interpret available information.
RB-12	Photos/videos at appropriate locations.	• To discern reason for leakage from the reactor building into the turbine building.	Improved BWR AM strategies (Plant improvements); potential PWR impacts, depending on identified leakage path.	Now.	Not currently considered by TEPCO; some photos may already be available. Labor needed to identify and interpret available information.
RB-13	Photos/videos of U1, U2, and U3 main steam lines at locations outside the PCV.	• To determine PCV failure mode.	BWR AM Strategies (plant mods, etc.) and better simulations for training.	Now and later.	TEPCO has not yet considered, but may do photographic exams. TEPCO has some temperatures around MSIV recorded since September 2011 for Units 2 and 3.

[k] Passage of high temperature gas from venting operations at U1 and U3 may have affected seals. The effluent vented from Units 1 and 3 would also have subjected these components to high radiation fields. Note that, at present, available evidence indicates that Unit 2 may not have been successfully vented. The high radiation fields in components of the Unit 2 reactor building ventilation system appears to have been caused by Unit 1 vent effluent bypassing the vent stack shared by Units 1 and 2. Many PWRs have safety grade fan cooler units for post-loss of coolant accident (post-LOCA) containment heat removal; PWRs would be interested if there is anything to learn.

Table B-2. Information Needs from the Primary Containment Vessel					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Cost/Level of Effort**
PC-1	Tension, Torque, and Bolt Length Records (prior and during removal); Photos/videos[l] of head, head seals, and sealing surfaces (U1, U2, and U3).	• Determine how head lifted. • Determine peak temperatures. • Look for indicators of degradation due to high temperature hydrogen, including hydrogen-induced embrittlement.	AM Strategies; What happened with respect to the leak path; better simulations for training.	Now (initial data and photos) and later (if head removed).	TEPCO will obtain some photos and may have last outage tension records; Labor needed to evaluate and post[m].
PC-2	Photos/videos and radionuclide surveys/ sampling of IC (U1).	• 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.	Now.	TEPCO has some photos (and no damage was observed); no RN sampling planned (due to radiation levels). Labor needed to sample, evaluate, and post.
PC-3	a) If vessel failed, photos/videos of debris and crust, debris and crust extraction, hot cell exams, and possible subsequent testing (U1, U2, and/or U3).	• Code assessments • Possible model updates for mass, height, composition, morphology (e.g., coolability), topography of debris, spreading, splashing, and salt effects.[n]	BWR AM Strategies (plant robustness, use of equipment) and better simulations for training. Potential PWR impacts (e.g., modeling.)	Now and > 5 years (per TEPCO roadmap).	In the near term, TEPCO will be obtaining some samples and taking some photos; TEPCO will also obtain and complete some exams; Labor needed to evaluate and post; Labor for screening of technology proposals submitted by US.
	b) If vessel failed, U1, U2, and U3 PCV liner inspections (photos/videos and metallurgical exams).	• Code assessments. • Possible model improvements for predicting liner failure and Molten Core Concrete Interactions (MCCI).	AM Strategies (improved plant robustness); better simulations for training.	Now and > 5 years (per TEPCO roadmap).	TEPCO has some bellows inspection information and will obtain some additional visual exams. TEPCO may do metallurgical exams (if warranted). Labor needed to evaluate and post.
	c) If vessel failed, photos/video, RN surveys, and sampling of U1, U2, and U3 pedestal wall and floor.	• 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.).	Now and later.	TEPCO has some initial inspection information and will obtain some additional information later; Labor needed to evaluate and post.

[l] 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.

[m] The phrase, 'labor needed to evaluate and post information' is a truncated version of activities described in more detail in Section 4. Namely, labor is needed for cognizant experts to evaluate with respect to the adequacy of provided inspection information to address information need, request additional inspection information (e.g., photos, videos, samples, etc.), and draw insights with respect reactor safety, and post in an easy-to-read format for global use.

[n] 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.

Table B-2. Information Needs from the Primary Containment Vessel					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Cost/Level of Effort**
	d) If vessel failed, U1, U2, and U3 concrete erosion profile; photos/videos and sample removal and examination	• 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.).	Now and later.	TEPCO has no plans to obtain at this time. TEPCO may consider in the future.
	e). If vessel failed, photos/videos of structures and penetrations beneath U1, U2, and U3 to determine damage and corium hang-up	• Code assessments. • Possible model improvements.	BWR AM Strategies (plant modifications, etc.) and better simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.).	Now and later.	TEPCO will obtain some information; Labor needed to evaluate and post.
PC-4	Photos/videos of U1, U2, and U3 recirculation lines and pumps	• To determine PCV failure mode and relocation path.	AM Strategies (plant mods, etc.) and better simulations for training.	Now and later.	TEPCO has some pressure and temperature measurements at PLR pump inlet since April 2011. No additional inspections planned. Labor needed to interpret and post data.
PC-5	Photos/videos of U1, U2, and U3 main steam lines and ADS lines to end of SRV tailpipes, including instrument lines	• To determine RPV failure mode.	BWR AM Strategies (plant mods, etc.) and better simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.).	Now and later.	TEPCO has not yet considered, but may do photographic exams. TEPCO has some temperatures around SRV and MSIV recorded since September 2011 for Units 2 and 3.
PC-6	Visual inspections of U1, U2, and U3 SRVs 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 has not yet considered; currently has no plans for any such inspections. Labor needed to obtain, evaluate, and post data.
PC-7	Ex-vessel inspections and operability assessments of U1, 2, and U3 in-vessel sensors and sensor support structures[o]	• Data qualification for code assessment. • Identification of vessel depressurization paths.	Equipment qualification life (U1 at 40 years; underwater cabling); better simulations for training.	Now and later.	TEPCO has completed some examinations and recalibrations and plans to perform more evaluations; Labor needed to evaluate and post data; Possible additional calculations and /or new sensor development efforts.

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

Table B-2. Information Needs from the Primary Containment Vessel					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Cost/Level of Effort**
PC-8	Inspections and operability assessments of U1, U2, and U3 ex-vessel sensors and sensor support structures[p]	• Data qualification for code assessment. • Identification of vessel depressurization paths.	BWR and possible PWR equipment qualification life; better qualifications for training.	Now and later.	TEPCO has completed some examinations and recalibrations and plans to perform more evaluations; Labor needed to evaluate and post data; Possible additional calculations and /or new sensor development efforts; Additional funding needed for cable sample removal and evaluation.
PC-9	Photos/videos of U1, U2, and U3 PC (Suppression chamber and DW) coatings	• Assess impact for coating survivability.	BWR and possible PWR maintenance upgrades.	Now and later.	TEPCO has some initial inspection information and will obtain more data later; Labor needed to interpret and post data.
PC-10	U1, U2, and U3 RN surveys in PCV	• Dose code assessments. • Possible model improvements.	BWR and possible PWR AM strategies/Better simulations (plate out).	Now and later.	TEPCO has some initial inspection information and will obtain more data later; Labor needed to evaluate and post data.
PC-11	Photos/videos of U1, U2, and U3 primary system recirculation pump seal failure and its potential discharge to containment	• To assess performance under high temperature/ high pressure conditions.[q]	Improved BWR AM strategies (plant improvements). Improved understanding of events. Potential PWR impacts.[q]	Now and later. As noted in Table E-1, inspections could be completed more easily at Daini.	Not currently considered by TEPCO; some photos may already be available. Labor needed to identify and evaluate.
PC-12	Photos/videos of U1, U2, and U3 Traveling In-Core Probe (TIP) tubes and SRV/Intermediate Range Monitor (IRM) tubes outside the RPV	• To determine if failure of TIP tubes and SRV/IRM tubes outside the RPV led to depressurization.	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.	Although TEPCO has not conducted direct investigation into this issue, an attempt was made to insert a fiber optic scope through the Unit 2 TIP guide tube. The scope was stuck at the TIP indexer and could not get past that location. Labor needed to obtain, evaluate, and post data.
PC-13	Photos/videos of U1, U2, and U3 insulation around piping and the RPV.	• To determine potential for adverse effects on long-term cooling due to insulation debris.	Improved BWR and PWR AM strategies (plant improvements).	Now and later.	Not currently considered by TEPCO; some photos may already be available. Labor needed to identify and evaluate.

[p] 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

[q] 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-2. Information Needs from the Primary Containment Vessel					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Cost/Level of Effort**
PC-14	Samples of conduit cabling, and paint from U1, U2, and U3 for RN surveys.	• Dose code assessments. • Possible model improvements.	BWR and possible PWR AM strategies/Better simulations (plate out).	Now and later.	TEPCO has some initial inspection information and will obtain more data later; Labor needed to evaluate and post data.
PC-15	Samples of water from U1, U2, and U3 for RN surveys.	• Dose code assessments. • Possible model improvements.	BWR and possible PWR AM strategies/Better simulations.	Now and later.	TEPCO has some initial inspection information and will obtain more data later; Labor needed to evaluate and post data
PC-16	Photos/videos of melted, galvanized, or oxidized U1, U2, and U3 structures.	• To provide indications of peak temperatures.	Improved AM strategies (Plant improvements).	Now and later; as noted in Table E-1, this should also be done at Daini.	Not currently considered by TEPCO; some photos may already be available. Labor needed to identify and evaluate in terms of peak temperature and hydrogen production.

Table B-3. Information Needs from Reactor Pressure Vessel					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Cost/Level of Effort**
RPV-1	U1, U2, and U3 dryer integrity and location evaluations (photos/videos[r] with displacement measurements, sample removal and exams for fission product deposition, peak temperature evaluations)	• Code assessments. • Possible model improvements.	Improved AM strategies; Improved simulations for training.	Later (after 2017 based on current roadmap).	TEPCO will conduct visual, some metallurgical and fission product exams; Labor needed to collate and post data.[s]
	Photos/videos, probe inspections, and sample exams of U1, U2, and U3 MSLs; Interior examinations of MSLs at external locations	• Code assessments. • Possible model improvements.	Improved AM strategies; Improved simulations for training.	Later (after 2017 based on current roadmap).	TEPCO has no plans for any such exams; Funding needed for some exams; Labor needed to collate, interpret, and post data.
	Photos/videos and metallurgical examinations of upper internals and upper channel guides	• Code assessments. • Possible model improvements (for predicting peak temperatures, displacement, melting).	Improved AM strategies; Possible plant modifications; Improved simulations for training.	Later (after 2017 based on current roadmap).	TEPCO will conduct visual exams and some metallurgical exams; Labor needed to collate, interpret, and post data.
RPV-2	Photos/videos of U1, U2, and U3 core spray slip fit nozzle connection, sparger & nozzles Photos/videos of U1, U2, and U3 feedwater sparger nozzle and injection points	• Assess operability. • Assess salt water effects (including corrosion). • Applicable to BWRs and PWRs.	Improved AM strategies; Improved simulations for training; Possible use in BWR Vessel and Internals Program (VIP) [depending on plant condition].	Now and Later.	TEPCO has some information (can show TC response if water injected) and will obtain more data; Labor needed to collate, interpret, and post data.
RPV-3	U1, U2, and U3 steam separators' integrity and location (photos/videos with displacement measurements, sample removal and exams for FP deposition, peak temperature evaluations)	• Code assessments. • Possible model improvements.	Improved AM strategies, Improved simulations for training.	Later (after 2017 based on current roadmap).	TEPCO will conduct visual, some metallurgical and fission produce exams; Labor needed to collate, interpret, and post data.

[r] 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)

[s] The phrase, 'labor needed to evaluate and post information' is a truncated version of activities described in more detail in Section 4. Namely, labor is needed for cognizant experts to evaluate with respect to the adequacy of provided inspection information to address information need, request additional inspection information (e.g., photos, videos, samples, etc.), and draw insights with respect reactor safety, and post in an easy-to-read format for global use.

Table B-3. Information Needs from Reactor Pressure Vessel					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Cost/Level of Effort**
RPV-4	U1, U2, and U3 shroud inspection (between shroud and RPV wall); Photos/videos and sample removal and oxidation testing.	• Code assessments. • Possible model improvements.	Improved AM strategies; Improved simulations for training.	Now and later (after 2017 based on current roadmap).	TEPCO has not yet considered; TEPCO has some information (U2 pressure sensor response after water injection) and will conduct visual exams; Labor needed to collate, interpret, and post data.
	U1, U2, and U3 shroud head integrity and location (photos/videos, and metallurgical exams)	• Code assessments. • Possible model improvements.	Improved AM strategies; Improved simulations for training.	Later (after 2017 based on current roadmap).	TEPCO will conduct visual exams and some metallurgical exams; Labor needed to collate, interpret, and post data.
	Photos/videos of U1, U2, and U3 shroud inspection (from core region)	• Code assessments. • Possible model improvements.	Improved AM strategies; Possible plant modifications; Improved simulations for training.	Later (after 2017 based on current roadmap).	TEPCO will conduct visual exams; Labor needed to collate, interpret, and post data.
	Photos/videos of U1, U2, and U3 core plate and associated structures	• Code assessments. • Possible model improvements.	Improved AM strategies; Possible plant modifications; Improved simulations for training.	Later (after 2017 based on current roadmap).	TEPCO will conduct visual exams; Labor needed to collate, interpret, and post data.
RPV-5	Remote mapping of U1, U2, and U3 core through shroud wall from annular gap region (muon tomography and other methods, if needed)	• Code assessments. • Possible model improvements.	Improved BWR and potential PWR AM strategies; Improved simulations for training	Now and later (after 2017 based on current roadmap).	TEPCO is deploying muon tomography; Labor needed to collate, interpret, and post data; US screens proposals for other ideas; Additional funding for out-year muon tomography examinations if successful
	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.	Later (after 2017 based on current roadmap).	TEPCO has not yet considered but will probably perform, as necessary for defueling and D&D. Additional funding may be needed for some sample removal and examinations for data interpretation and posting.

Table B-4. Other Research and Development Tasks					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Cost/Level of Effort**
O-1	Labor for accessing TEPCO information; Interpreting it, assessing adequacy, and posting in an Easy to Use Format	• Data for D&D needs differs from Reactor Safety Needs.	Ensures TEPCO information is available to international community. Code assessments Model improvements Possible plant modifications Improved simulations for training.	Now (data are already available).	~ 1 FTE (split cognizant industry and laboratory experts). ~0.25 FTE (website interface).
O-2	Establish US POC	• Coordinate US inquiries, requests, and responses. • Become familiar with all information available from TEPCO (to minimize O-1 costs). • Assess US proposals and provide input to TEPCO.	Facilitates needed US/Japan interactions	Now.	~1 FTE
O-3	Periodic code 'checkups' and modeling improvements	• Ensures that obtained data are used and knowledge is not lost.	Improved BWR and potential PWR SAMGs; Improved BWR and potential PWR simulations for staff training	> 2 years from now.	May be possible from existing EPRI and NRC code updates.
O-4	Seismic inspections at Daiichi and Daini	• To obtain 'real' comparison data on instrumentation, component, and structure survivability during seismic events.	BWR and potential PWR plant modifications that are important to forensics and with respect to repair, maintenance, and field applications.	Now and after debris removal	~1 FTE
O-5	Poll industry for additional inspections at Daiichi and other plants (see Appendix E, Table E-1 for initial industry poll).	• Impact on concrete. • Impact on instrumentation, components, and structure survivability (water storage tanks, dry storage casks, suppression pools, recirculation pump seals, etc.).	BWR and potential PWR plant modifications that are important to forensics and with respect to repair, maintenance, and field applications.	Now and after debris removal	~0.2 FTE(to poll industry) ~1-2 FTE (to obtain and assimilate data).
O-6	Advanced technologies for U4 external ductwork surveys	• To characterize radionuclides present. • Possible earlier identification if CCI occurred.	Earlier insights on what occurred.	Now	~ 0.5 FTE plus equipment and travel for cognizant persons to train TEPCO on use and assimilate obtained data.

Table B-4. Other Research and Development Tasks					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Cost/Level of Effort**
O-7	Develop procedures to obtained desired data	• Prior US experience could assist in TEPCO preparations.	Provides 'work-in-kind' support Ensured that desired information is obtained.	Now and later.	0.5 (for cognizant expert to work with TEPCO and draft procedures.
O-8	Advanced technologies to assist with obtaining desired data	• Sample removal and post-accident examinations are expensive. It would significantly reduce costs if advanced technologies could be used to obtain desired information without sample removal.	Provides 'work-in-kind' support; Ensured that desired information is obtained. Code assessments; Model improvements	Now and later.	TBD (depends on technology explored).
O-9	Funding for additional muon technology examinations	• Provides early indicator of debris location. • Provides data required for D&D.	Earlier insights on debris insights.	After initial inspections with muon technology are shown to be successful.	TBD
O-10	Instrumentation Survivability Assessments (data, calibration, and operational checks) for U1, U2, U3, and U4 at Daiichi and Daini (see Appendix E, Table E-1 for initial industry input)	• Plant comparison information. • Code assessments • Possible model improvements. • Assess operability • Applicable to BWRs and PWRs.	BWR and PWR plant modifications with respect to repair, maintenance, and field applications; Possible enhancements to Emergency Operating Procedures (EOPs)/SAMGs.	Now (data are already available).	TEPCO has some initial inspection information and will obtain more data later; Labor needed to interpret and post data.
O-11	Aging management inspections (photos, operation checks, tan delta testing, etc.) for cabling at U1, U2, U3, and U4 at Daiichi and Daini (see Appendix E, Table E-1 for initial industry input).	• Plant comparison information. • Assess operability • Applicable to BWRs and PWRs.	BWR and PWR plant modifications with respect to aging management, repair, maintenance, and field applications.	Now (data are already available).	TBD.
O-12	Photos/videos[t] of U1, U2, and U3 vent lines at Daiichi if used for venting during post ELAP response. Determine condition of rupture disc.	• Plant comparison information. • Assess operability • Applicable to BWRs and PWRs.	BWR and PWR plant modifications with respect to aging management, repair, maintenance, and field applications.	Now (data are already available).	TBD.

[t] With the exception of general area views, photos and videos should be obtained with reference length scales at appropriate locations.

Table B-4. Other Research and Development Tasks					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Cost/Level of Effort**
O-13	Photos/videos of primary system recirculation pump seals U1, U2, and U3 at Daiichi and Daini to its potential discharge to containment	• To assess performance under high temperature/ high pressure conditions.[u]	Improved BWR AM strategies (plant improvements). Improved understanding of events. Potential PWR impact.[u]	Now and later. As noted in Table E-1, inspections could be completed more easily at Daini.	Not currently considered by TEPCO; some photos may already be available. Labor needed to identify and evaluate.
O-14	Photos/videos of melted, galvanized, or oxidized U1, U2, and U3 structures at Daini and Daiichi (see PC-16)	• To provide indications of peak temperatures.	Improved AM strategies (Plant improvements).	Now and later.	Not currently considered by TEPCO; some photos may already be available. Labor needed to identify and evaluate in terms of peak temperature and hydrogen production.
O-15	Photos/videos[v] of condition of RCIC valve and pump before drain down and after disassembly (at affected units at Daiichi or Daini)	• 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.).	Currently, locations at U2 and U3 at Daiichi are flooded (requires underwater investigations unless drained down by TEPCO). As noted in Table E-1, inspections could be completed more easily at Daini.	Not currently considered by TEPCO; If torus not drained, requires underwater technology available. Some operational data may be available. Labor needed to identify, evaluate, and post.

[u] 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 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.

[v] 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.

Appendix C

November 22, 2013 Meeting Results

[Only preliminary main report with Attachment A of this document included in this appendix]

Results of Nov. 22, 2013 Experts Meeting on Fukushima Reactor Inspection Planning

A Preliminary Report by

The Department of Energy

Office of Nuclear Energy

Office of Light Water Reactor Technologies

May 1, 2014

INTRODUCTION AND SUMMARY

In September 2013, DOE's Office of Nuclear Energy (DOE-NE) approved a Reactor Safety Technologies Research and Development Program Plan, aimed at enhancing reactor safety in the U.S. The primary focus of the Program Plan was to respond to the accident at the Fukushima Dai-ichi reactor plant in Japan on March 11, 2011. DOE began addressing Fukushima issues almost immediately after the accident as part of its efforts to understand the accidents on March 11, 2011 and subsequent events, and to respond to various requests from within the U.S. Government and from industry and Japanese organizations.

The vision for the DOE Reactor Safety Technologies R&D Program is to develop a fuller understanding of the accident at the Fukushima Dai-ichi reactor plant through analysis of the accidents, their progression and phenomenology; and to apply this understanding and its safety implications to further enhance the safety of existing and future nuclear power plants in the U.S. These efforts are expected to provide useful and timely knowledge to the U.S. nuclear industry in support of its responsibilities (1) for the safe and economic performance of commercial nuclear power plants, and (2) for incorporating lessons learned from the Fukushima accident into its reactor operations. The R&D program includes activities to inform and support the eventual examination of the Fukushima reactors.

The specific objective of the DOE's Fukushima reactor inspection effort is to improve data collection and understanding of severe accidents and to reduce uncertainty around accident progression, for the benefit of U.S. safety analysis capabilities. More specifically, this R&D is intended to provide insights on accident progression and performance of structures, systems, and components through visual examination of in-situ conditions of the damaged Fukushima units, and collection of material samples for future analysis.

BENEFITS

The U.S. has a substantial interest in the results of inspections and equipment sampling that Japan may undertake, because information from the damaged reactors will provide valuable lessons that can further improve reactor safety. There are clear benefits to this research beyond assisting Japan with its own immediate needs. Data, models, and insights from this work will inform many aspects of U.S. reactor safety, including accident analysis tools, severe accident management guidelines (SAMGs), new or revised safety requirements in response to Fukushima, etc. Technologies and lessons-learned developed in the course of this work can be used to prevent and/or reduce the consequences of future accidents.

In particular, today's safety analysis tools, procedures, training, etc. lack a comprehensive underpinning of real operating experience under severe accident conditions, particularly for BWRs – thus the need for these data for the verification and validation (V&V) of current and future safety analysis tools. Even with modern modeling and simulation (M&S) technologies, researchers, regulators, owner/operators and others are forced to rely on a limited database of information to validate these tools. TMI-2 data, in combination with a number of focused and/or large-scale experiments conducted decades ago, constitute the bulk of available data. Fukushima represents an opportunity to vastly increase our experience and data upon which these tools rely. Further, most of the existing data apply primarily to PWRs. Fukushima represents an opportunity to greatly expand our understanding of accidents in BWRs.

STAKEHOLDERS

Primary users of DOE's efforts in developing proposed inspection and sampling priorities for the Fukushima reactor plant are expected to be Japanese organizations responsible for planning and executing the decommissioning of the four reactors at the Fukushima Daiichi site in Japan, as well as any international organizations (governments and industry) that assist Japan in this effort.

The United States – both government and industry – will benefit from the results of these inspections, as described above. Specific U.S. stakeholders include the U.S. Nuclear Regulatory Commission (NRC), other government agencies that may have an interest (State, Commerce, etc.); and the U.S. nuclear industry, including nuclear energy plant owner/operators, reactor vendors, and central organizations responsible for generic industry response to the Fukushima reactor accident, including the Nuclear Energy Institute (NEI), the Electric Power Research Institute (EPRI), the Institute of Nuclear Power Operations (INPO), and the reactor owners groups. Finally, the broader global nuclear energy community would similarly benefit from the data obtained from these inspections.

DOE-NE and NRC's Office of Nuclear Regulatory Research have agreed to collaborate on this reactor inspection and sampling activity. While DOE and NRC will both play roles in planning such an international effort, Japan would ultimately lead any such sampling effort, and Japan would be the lead in determining the scope and parameters of any such sampling plan. Diplomatic interactions among U.S. federal agencies and their Japanese counterparts in proposing and supporting this inspection planning effort are outside the scope of this report, which is focused on technical matters. These interactions could be of a bilateral nature between the U.S. and Japan, or multi-national in nature, via the OECD's NEA, via the IAEA, and/or other multi-national organizations.

DOE-NE EXPERTS MEETING ON FUKUSHIMA REACTOR INSPECTION PLANNING

A meeting was conducted in Washington DC on 22 November, 2013, for the purpose of capturing the ideas and suggestions from a diverse group of U.S. experts regarding future inspection and sampling tasks that Japan may want to consider as it decommissions the Fukushima reactors. A number of key Japanese experts (Mitsubishi, TEPCO) also participated and provided insights into the reactor plants and the accident progression.

The purpose of this meeting was to develop an initial comprehensive listing of inspections/sampling of the Fukushima Daiichi reactors during decommissioning considered beneficial by key US stakeholders. This effort was not intended to be a "U.S. position" on what should be inspected, but rather an input into the process of discussing/determining with Japan, and the broader global nuclear community, what should be inspected, given the many other considerations (such as cost and potential radiation dose to site workers) that would factor into an inspection plan. As such, it is expected that going forward there will be a prioritization/narrowing down of the list that was developed, to the most essential or cost effective inspections. This initial listing was intended to facilitate the U.S. Government's efforts to provide support to Japan in the area of Fukushima reactor inspections. These inspections will provide insights on accident progression and performance of structures, systems, and components for the benefit of U.S. safety analysis capabilities, with parallel benefits to Japan and other nations. These insights will be gained through visual examination of in-situ conditions of damaged Fukushima units, and collection of material samples for future analysis. A key consideration of this meeting was optimization of data collection within the context of the larger cleanup and decommissioning effort, led by Japan with international support.

The results of this meeting, and explanation of the attachments to this report, are explained below under "Results of Expert's Meeting." Note that this report is intended to be a living document. As new information is obtained or further prioritization of inspection recommendations are accomplished, this report will be modified to capture that information.

COMMITMENT TO JAPAN AND ITS LEAD ROLE IN INSPECTION PROGRAM

All of this work will be carried out in cooperation with various Japanese counterpart organizations, which ultimately control the content and pace of inspection and cleanup activities. The efforts of Japanese entities (government and industry), will provide opportunities for the U.S. and the broader nuclear community to obtain vital data and samples for R&D purposes. It is worth noting that there is ample precedent for international collaboration and technical support in response to severe accidents at commercial reactor facilities. DOE provided extensive technical support following the TMI-2 accident, including event analyses, testing and analysis of the damaged reactor core, etc. Japan assisted in this effort, providing resources and gaining access to valuable safety information. Following the catastrophic accident at Chernobyl, DOE also provided extensive technical support, as did Japan.

BACKGROUND

The general approach to be taken in this activity is expected to build on the extensive U.S. experience in managing the cleanup of TMI-2 to inform the recovery, inspection and cleanup effort in Japan, working with Japan's government and industry experts and organizations. In return for sharing this experience base and helping to organize the inspection effort, the U.S. anticipates being able to provide input as to what data should be obtained, and thereby gain access to valuable data, to be used as discussed below. Activities include suggesting a plan for sampling key structures, systems and components, as well as analyses to characterize end states, instrumentation survivability, materials and fuel failure mode and effects analyses (FMEA), etc., and preliminary domestic planning for sample extraction. As new information becomes available, the priorities of this area will be revisited.

Much of the TMI-2 experience is valid today, especially in planning the overall approach and in benchmarking various strategies, procedures, and processes. The TMI-2 cleanup effort, supported by DOE, was well documented. Modern technology could make many aspects of the process easier and more efficient than it was in the 1980s, with lower radiation exposures to the workers involved. In particular, modeling and simulation (M&S) technologies, along with severe accident expert knowledge related to uncertainties in the existing computational models, are expected to be used extensively to predict what will be observed, to aid in planning the initial entry into damaged reactor plants, and to inform what specific structures, systems and components need to be inspected, sampled, and/or recovered. Much of this advanced M&S effort will likely come from a benchmarking effort organized by the NEA, to be augmented by DOE collaborations with U.S. industry (e.g., a "cross-walk" between the NRC code "MELCOR" and the industry code "MAAP").

Many aspects of the Fukushima accident may require new approaches that cannot be informed by TMI-2, and which should provide unique insights and lessons learned. Examples include: seismic response data, impacts of seawater cooling on structures, systems, and components, BWR-unique severe accident progression and phenomenology (especially related to pressure suppression containments, BWR fuel technology, etc.), and late phase accident progression including ex-vessel core-concrete interaction. Note that the Fukushima accident represents the first occasion of a severe accident at a BWR, and the first occasion of the use of salt water to cool a damaged reactor. Given the corrosive nature of salt water as well as the potential impacts of the presence of salt on accident progression and phenomenology due to changed thermohydraulic and chemical environments, this topic deserves attention. (Note that in the U.S., "raw water" is a more likely emergency water source than salt water.)

A preliminary inspection matrix, focused on those structures, systems and components that are likely to be the most important targets for inspection and sampling was developed, based in part from the M&S efforts described above. It served as the foundation for the Nov. 2013 experts meeting, where it was expanded significantly, as discussed below.

RESULTS OF EXPERTS MEETING

A technical meeting among U.S. and Japan experts on the Fukushima reactor design and accident sequence was held on 22 Nov. 2013. The meeting was organized to optimize the efficient capture of inspections and sampling tasks that the participating U.S. experts and stakeholders believe are appropriate during the decommissioning of the Fukushima reactors. In addition to a list of these inspection and sampling needs, the matrix includes the basis or rationale for those needs, since this data collection must be prioritized to minimize impacts on decommissioning. Inspections take time and money, both of which are likely to be limited. Inspections must be planned so they minimize their impact on the reactor cleanup and decommissioning schedule.

Since the cleanup and decommissioning of the Fukushima site is the responsibility of Japan, diplomatic outreach on aspects of this initiative with Japan will be worked out separately within the U.S. Government (USG) (i.e., DOE, NRC, State). Inputs received during the meeting should be viewed as suggestions from the perspective of U.S. nuclear R&D experts. The meeting was a technical brainstorming session, which the USG will draw from in communicating with its counterparts in Japan.

This report contains an **Inspection Matrix,** a five-column table containing inspection ideas, that was reviewed collectively and modified spontaneously during the meeting; and **Inspection Templates**, which were used during and after the meeting by individual participants to provide more information on the details of desired inspections, plus any other information the individual contributors considered important.

Attachment A ("Fukushima Daiichi Inspection to Aid Severe Accident Research") provides the Planning Matrix. Attachment B provides submitted Inspection Templates. Attachment C provides the list of meeting participants. Participants included experts who have been directly involved in Fukushima event reconstruction, experts who have been directly involved in Fukushima accident computer modeling (e.g., MELCOR, MAAP), and BWR technology experts (structures, systems and components, fuel, operations, etc.).

RESULTS:

The results of the meeting are captured in this meeting report and associated attachments. The primary meeting output file (Appendix A) is an Inspection Matrix that has been organized in three sections: one on Primary Containment inspections, one on Reactor Pressure Vessel inspections, and one on Reactor Building inspections. Also attached are key meeting presentation materials (Appendices D thru N).

After this report has been issued, documentation will be prepared to facilitate the USG plan to communicate results to Japan.

Key points made during the meeting regarding a future Fukushima inspection program follow:

- Prioritization process should differentiate between "nice to know" and "need to know" data
- Industry's needs may be different than regulatory needs; U.S. needs may be different than Japan's needs.

- Integrated planning is important: R&D data collection needs to be sequenced and superimposed on Japan's cleanup and decommissioning schedule, such that data collection does not appreciably impact decommissioning.
- TMI-2 experience suggests there will be surprises. Inspection plans must be flexible and responsive to new information.
- State of the art inspection and sampling technologies (e.g., robotics) can improve efficiency, minimize costs, and minimize radiation exposure. Optimizing in-situ versus extraction and off-site inspection strategies will further contribute to efficiency and cost/dose minimization.
- A system will be needed to collect, organize, preserve and share vast amounts of data.
- Severe accident codes such as MAAP and MELCOR have been used to model Fukushima event sequences; these codes were benchmarked against limited available data. They predict plant responses differently in some cases; benchmarking these codes against more detailed data obtained from these inspections will vastly improve our understanding of what actually happened during the accident. Specifically, a recent MAAP-MELCOR "Crosswalk" has highlighted differences in how the codes model corium melt temperatures (predefined values, model abstraction differences, etc.), and the course of the accident progression in general. These differences, in turn, manifest themselves in different views of what component failed first at Unit 1. Inspection results are expected to confirm which predictions are correct.
- Some instrument readings and other raw data from the accident may not be reliable. Inspections will help qualify data and verify correct performance of key instruments.
- Data needs also include spent fuel pools. These data may be the first obtainable, since the pools are the most structurally intact and therefore not nearly as impacted by high radiation conditions.
- Reactor Buildings in Units 3 and 4 look like a H_2 detonations; Unit 1 looks like a H_2 deflagration.
- Inspection results will likely impact BWR operations – forming the basis for reassessing training needs, possible updates to EOPs and SAMGs, etc. Existing EOPs and SAMGs are based largely on MAAP runs on TMI-2 data. If code modeling is changed based on Fukushima findings, modification to EOPs and SAMGs may be needed.
- Aging of components is continuing today, in part due to effects of salt water. Tests that artificially age reactor components in salt water will likely show that degradation is a slow process.
- Reflooding containment to identify leakage paths might present recriticality or seismic issues.
- Inspections should be performed everywhere for seismic damage.
- Sampling and off-site testing may be a less expensive option than in-situ testing, if visual inspection is insufficient.

FUKUSHIMA DAIICHI INSPECTION TO AID SEVERE ACCIDENT RESEARCH

PRIMARY CONTAINMENT

Item	What	Applicable Unit(s)	Why (what question are we trying to answer?)	How
PC-1	Tension (torque) in each bolt on the Primary Containment Vessel (PCV) head ("Drywell head"); map findings. Look at material properties and tensioning details prior to inspection.	Units 1,2, and 3	Determine if head lifted and if so, if it lifted asymmetrically	Visual – signs of asymmetric lift or leakage paths. Look for thermal deformation due to high temperatures over time.
PC-2	Inspect drywell head to determine if leakage from the head occurred. Inspect seal between PCV and head for heat discoloration, scoring, or abnormal changes. Inspect head for possible deformation.	Units 1,2, and 3	Determine if head lifted and if so, if it lifted asymmetrically. The condition of the drywell head is critical to understanding the possible leakage pathways into the reactor building. This understanding impacts the potential benefits of an external filter and will support consideration of possible mitigation strategies.	1. RN Swabbing 2. Visual inspection of seal (steam cuts, etc.) 3. Visual inspection of the head. 4. Inspect shield plug – visual inspection of cracks.
PC-3	Inspect leakage points in PCV (bellows, penetrations, seals, etc.). Inspect at least one example of each type of penetration (electrical, etc.). During flood-up to find and repair leakage pathways, inspect failure points. During cleanup, inspect these sites before plugging, to determine failure location, size, and failure mode during accident, as well as water and/or hydrogen leak path during events.	All units; also identify pathways from unit 3 to unit 4	TEPCO interested in finding leakage points as sources of current leakage. Information on how the penetrations responded to accident conditions will be useful in determining if specific mitigation strategies could be beneficial in preventing such leakage in the future.	1. Visual inspection 2. Capture in photos 3. Use dose rate maps to pick best sites for inspection 4. Look for Cs or Ba indication at cracks
PC-4	Isolation Condenser (IC) system Physical inspection of entire system. Also, sample the radionuclide inventory in the IC. Since the noble gases will track with the hydrogen, the	Unit 1 only	Evidence of water level in IC? Confirmation of small amount of cooling through Leaking valves during event? Evidence of hydrogen in IC? The daughter products from the dominant isotopes	1. Visual – was there seismic damage? 2. Condition of valves, including indication of valve leakage

A-1

Attachment A

Item	What	Applicable Unit(s)	Why (what question are we trying to answer?)	How
	isotope relationship of the deposition material may be weighted more toward the daughter products of the noble gases as opposed to the Cesium and Rubidium fission products.		for the noble gases result in different isotopes than those produced directly from nuclear fission; therefore, regions that had a high concentration of noble gases with lower amounts of volatile fission products should contain a different ratio of isotopes for Cesium and Rubidium. As an example, the deposition within the reactor pressure vessel should be higher in Cesium-137 than that seen in the isolation condenser, which should contain higher amounts of Cesium isotopes resulting from the decay of Xenon. By examining these ratios, an estimate of the location of the hydrogen can be performed.	
PC-5	Extent of corium on the floor (if any) • Overall mass of discharged core debris • Area coverage • Topography of debris surface • Chemical composition • Morphology (i.e. particle bed vs. fractured vs. monolithic material) • Debris splash on walls • Depth of debris • Concrete erosion profile (Note: Much of the requested information will need to be collected anyway in order to inform the teams that are preparing for removal of the core debris from the PCV, and to aid in the design of tools for the debris removal process. Thus, there is expected to be little incremental effort required to carry out this work.)	Unit 1 and possibly Unit 3. Unit 2 – if arrested in vessel, how did contamination get to drywell?	Understand ex-vessel melt progression and debris coolability. Ex-vessel debris coolability is one of the last unresolved issues that arose after TMI-2. The evidence to date suggests that a prototypic CCI was quenched and thermally stabilized by top flooding, thereby demonstrating debris coolability at plant scale. Examination of these reactors thus provides an unprecedented opportunity to gain phenomenological insights into the nature and extent of core debris quenching under prototypic ex-vessel accident conditions. This would constitute a major finding for LWR severe accident management; information gained from examinations would aid in closing out this technical issue that first arose after TMI-2. Results of inspections may impact SAMGs, especially if MELCOR predictions are correct. Will inform recovery and cleanup plan. Explicitly consider the following phenomena (needed for more accurate code predictions of coolability and time for mitigating strategies): • "Regional" (axial and radial) variability • Evidence of peak temperature (e.g., composition,	1. Visual inspection 2. Debris sampling (core 'plug' sample) 3. Hot cell exams (off-site) 4. Some inspections can be done early with cameras/probes 5. Look for indications of concrete dust/debris, (SiO_2) especially at higher elevations 6. After inspections are completed, additional R&D (that includes separate effect laboratory tests) may be needed to interpret observations.

A-2

Attachment A

Item	What	Applicable Unit(s)	Why (what question are we trying to answer?)	How
			morphology indicating previously melted, cracking, debris-to-structure gaps, etc.) • Evidence of salt effects (e.g., U-Cl compounds)	
PC-6	Inspect PCV Liner in the vicinity of the floor to determine if liner melting and failure occurred (only if debris went ex-vessel).	Units 1, 2 & 3; as applicable	Determine potential PCV failure mode Information on the liner will be useful in determining if specific mitigation strategies could be beneficial in preventing such leakage. This may also help determine the source of water to the torus room as observed recently by TEPCO.	1. Visual inspection 2. Metallurgical exam of sample near debris liner contact
PC-7	Pedestal wall and floor integrity: Inspect pedestal and drywell floor to determine the amount of debris located ex-vessel. The inspection should look at the structural steel directly beneath the RPV, the pedestal sumps, and possible debris spread onto the drywell floor (only if debris went ex-vessel).	Units 1, 2 & 3; as applicable	Extent of melt core concrete interaction (MCCI) The determination of the amount and location of core debris ex-vessel will help assess the degree of in-vessel core damage along with providing insights on how the vessel was breached and how debris was spread.	Visual inspection
PC-8	PCV concrete erosion profile (only if debris went ex-vessel)	Units 1, 2 & 3; as applicable	Extent of melt core concrete interaction (MCCI)	Late stage excavating Structural concrete samples
PC-9	Corium hang-up on structures under RPV Currently, both the MAAP and MELCOR analyses relocate debris directly from the lower head of the reactor pressure vessel to the floor of the drywell ignoring the possible retention by control rod drive mechanisms or support structures. It is recommended that MELCOR be used to model the structure beneath the vessel because of its greater flexibility; thereby providing an estimate of the debris material that could be frozen on	Units 1, 2 & 3	Determining RPV failure mechanism and pour conditions Within all three units, the corium is possibly located in one of four potential locations. Each of these locations presents its own unique obstacle to debris removal with some being more difficult to remove than others. As an example, for corium that remains in the core or the lower head, techniques perfected at TMI-2 can be used to remove this debris. For corium that has relocated to the floor, several possible extraction techniques may already exist. Conceivably, the most difficult to remove is the corium debris that has frozen on the control rod drive structures located beneath the reactor pressure vessel	Visual inspection

A-3

Attachment A

Item	What	Applicable Unit(s)	Why (what question are we trying to answer?)	How
	the structures beneath the vessel (only if debris went ex-vessel).		and above the floor. Therefore, an estimate of the amount of material that has solidified on the control rod drive structures would be extremely beneficial in estimating the debris removal cost.	
PC-10	Inspection of structures under RPV	Units 1, 2 & 3; as applicable		Visual inspection
PC-11	Recirculation lines and pumps, including pump seals	Units 1, 2 & 3; as applicable	Determine potential PCV failure mode	Visual inspection
PC-12	Main steam lines and ADS lines to end of SRV tailpipes, including instrument lines.	Unit 3:	Determine potential PCV boundary failure mode (MSLB versus ADS)	Visual inspection
PC-13	SRVs, including standpipes, to determine if there was any failure of the valve or associated piping that could have released gas and fission products directly to the drywell. Inspection should address interior valve mechanisms to assess if the valve had failed open or if any leak paths in the flange or tailpipe had been created during the accident.	Units 1, 2 & 3; as applicable	Determine potential PCV failure mode Look for evidence of SRV cycling Failure of the SRV could have implications on local drywell temperatures that could pose a challenge to containment and could prompt possible consideration of mitigation strategies. Understanding the possible failure modes can help us develop possible mitigation strategies.	Late stage removal of SRV
PC-14	Inspect in-vessel sensors - TIPs (TCs, FCs, etc.) - dP cells - Pressure gauges	All units	Sensor survivability assessment for data qualification. TIP penetrations as possible leak paths	Visual examination to detect end-state and possible failure mechanisms. Survivability testing. Metallurgical exams.
PC-15	Ex-vessel Sensors/Suppression Pool/Containment sensors - TCs - CAMS - Pressure gauges	All units	Sensor survivability assessment for data qualification It is important to evaluate our confidence in the data obtained from various sensors and quantify uncertainties in various data that are being used in on-going forensic efforts before one considers	Visual examination to detect end-state and possible failure mechanisms. Survivability testing.

A-4

Attachment A

Item	What	Applicable Unit(s)	Why (what question are we trying to answer?)	How
	Instrument survivability and data uncertainty: As researchers continue with code predictions for Units 1, 2, and 3, and comparisons with available data, the program should place greater emphasis on uncertainty in the data and survivability of the sensors giving such data. Instrument seismic support information: How instrument mounting racks and tubing supports held up under the earthquake.		modifications to models in the severe accident codes or changes in plant mitigation strategies. In addition to damage from high temperatures, pressures, and radiation levels, note that instrumentation uncertainties may vary over a particular range (e.g., Type T TCs will have a LOT more uncertainty if they are providing data outside their qualified operating range). Such evaluations can also provide insights related to the use of sensor data as indicators for other effects (e.g., flux monitors for temperature indicators, pressure monitors for IC operation indicators, etc.).	Metallurgical exams. Inspections, sensor removal, and separate effects testing may be needed. Information on how reactor and containment instruments behaved during the event: information could come from chart recordings, data histories, and operator notes and records. Any digital media should be preserved for laboratory examination, even damaged disks and memory devices. Retain instruments and associated valving for later inspection. Thermal and radiation effects on seals and working fluids (e.g., oil inside pressure sensors) are believed to be known from laboratory testing, but Fukushima artifacts could provide additional insights.
PC-16	Investigation into the assumption of symmetric versus asymmetric melting: Both in-vessel and ex-vessel inspections should include an analysis	Units 1, 2 & 3; as applicable	All accident models and analyses to date have assumed perfect axial symmetry of the core heatup and degradation processes. Since in reality small initial disturbances may significantly alter local core	Visual Inspection

Item	What	Applicable Unit(s)	Why (what question are we trying to answer?)	How
	of spatial distributions of the damage/failure zones: (a) RPV penetrations around the vessel, (b) degree of instrumentation damage at different locations (c) azimuthal distribution of relocated in-core materials, (d) local debris distribution inside lower plenum and on lower head, (e) Signs of the impact of single RCIC loop operation on in-vessel phase distribution and non-uniform core cooling.		coolability and, thus, corium heatup, melting, melt accumulation and discharge, and melt/structure interaction, it is important to assess the effect of such factors. It will also help to distinguish between deterministic and stochastic characteristics of accident progression.	
PC-17	Condition of primary containment coatings	Units 1, 2, 3	Potential impact of debris on sumps and screens	Visual inspection
PC-18	Quantify differences in radioisotope concentrations and quantities in the drywell and what is left of the reactor buildings. Try to find evidence that provide insights about various depressurization assumptions (e.g., SRV seizure, MSL rupture, etc.) and the potential for releases to be reduced by venting through the wetwell vent.	Units 1, 2, 3	Degree of iodine plate out on walls, etc., might give some insight into that process and the degree to which it might be credited for dose reduction.	Visual Inspection
PC-19	Inspection plan flexibility and visual examination importance: Greater emphasis is needed on inspection plan flexibility with hold points that allow initial inspections to inform subsequent inspections. In particular, it will allow inspections to be informed by initial visual	All units	Initial visual inspections after the TMI-2 experience considerably altered the focus of post-accident examinations and the development of subsequent improvements in plant operations. • Conduct visual observations and/or measurements that indicate melt progression phenomena and debris end state morphology (solid or refrozen liquid, cracked, particulate, etc.) and location. Subsequent	Visual examination

Item	What	Applicable Unit(s)	Why (what question are we trying to answer?)	How
	inspections and other non-destructive examination results. Plans should explicitly recognize the need for hold points that allow some reconsideration of options after various types of data are obtained. Greater emphasis should be placed on the importance of initial results from visual examinations and non-destructive evaluations.		inspections should focus on indications such as radioisotopes, which suggest the degree of melting and occurrence of core-concrete interactions. • Initial visual inspections should concentrate on identifying differences in melt progression due to accident-specific phenomena that differs from what was observed at TMI-2, such as dry event progression and the impact of salt, and that can provide insights about code assumptions such as holdup on the core plate. The discussion on visual observation should be expanded to emphasize the importance of any evidence of interactions between structures and relocated melt (e.g., attack by melt with higher metallic composition in a steam environment, possible corrosion by salt, and evidence of any remaining salt deposits--- sufficient detail should be obtained that we can map out the locations of such deposits).	

REACTOR PRESSURE VESSEL

Item	What	Applicable Unit(s)	Why	How
RPV-1	Inspections of dryer integrity and location	Units 1, 2 & 3; as applicable	Determine the temperatures that were experienced and end-state mechanical condition (sagging, melting, etc.)	1. Visual inspection 2. Metallurgical exam 3. Displacement meas. 4. FP deposition
RPV-2	Core spray slip fit nozzle connection, sparger & nozzles	Units 1, 2 & 3; as applicable	Functionality, thermal damage, salt clogging present; slip fit of nozzle. Determine if high temperatures diminished the ability to inject water.	Visual inspection
RPV-3	Feedwater sparger nozzle and injection points	Units 1, 2 & 3; as applicable	Determine if high temperatures diminished the ability to inject water.	Visual inspection
RPV-4	Send probe down all main steam lines (or through failure location). Visual inspection of main steam lines to determine if there was any failure of the pipe that could have released gas and fission products directly to the drywell. This inspection should include the MSL nozzle at the RPV wall, the piping, and instrument and drain lines coming off of the main steam line. A sample from the pipe surface could be used to determine temperature history of the pipe.	Units 1, 2 & 3; as applicable	Determine potential for various RPV failure mode(s). Failure of the main steam line has implications on local drywell temperatures that could pose a challenge to containment and could prompt possible consideration of new mitigation strategies. Understanding the possible failure modes can help us develop possible mitigation strategies.	Visual inspection
RPV-5	Determine whether there were multiple failures of the main steam lines or at other locations.	Units 1, 2 & 3; as applicable	Past MELCOR analysis show that after one failure occurs in the primary pressure boundary, successive failures may occur within a short time period (as short as a few minutes). Unless the first failure is a large one capable of depressurizing the primary side within a matter of a few minutes, the system pressure will remain high enough that other weak points give. Energy and mass release though subsequent failures will aid in understanding the accident progression.	Visual inspection

Item	What	Applicable Unit(s)	Why	How
RPV-6	Inspection of steam separators' integrity and location	Units 1, 2 & 3; as applicable	Determine the temperatures that were experienced and end-state mechanical condition (sagging, melting, etc.)	1. Visual inspection 2. Metallurgical exam 3. Displacement measurements 4. FP deposition
RPV-7	Shroud inspection (between shroud and RPV wall)	Units 1, 2 & 3; as applicable	Possible melt flow path	Visual inspection Testing for oxidation to determine if dryout of the jet pump region occurred
RPV-8	Remote core mapping through shroud wall from annular gap region	Units 1, 2 & 3; as applicable	Extent of core melt and inform D&D	1. Ultrasonic 2. X-rays
RPV-9	Inspection of shroud head integrity and location	Units 1, 2 & 3; as applicable	Determine the temperatures that were experienced and end-state mechanical condition (sagging, melting, etc.)	1. Visual inspection 2. Metallurgical exam 3. Displacement measurements
RPV-10	Map core remnants (if present)	Units 1, 2 & 3; as applicable	Extent of core melt progression and potential core crust formation. Look at Zircaloy channel boxes to gain insights about melting. Provides insights regarding TMI-2 melt progression (in MAAP) versus BWRSAR melt progression (in MELCOR) (i.e., investigation into the formation of eutectics). Examine asymmetry of material distribution. Issue is important for the understanding of local heatup/melting modes and melt relocation/discharge.	1. Visual inspection 2. Radiochemical sampling 3. Chemical
RPV-11	Inspect upper internals and upper channel guide to determine if high temperatures during the accident resulted in sagging or melting of this structure.	Units 1, 2 & 3; as applicable	Determine the temperatures that were experienced and end-state mechanical condition (sagging, melting, etc.) The structural condition of these internal components will provide valuable data as to the conditions created during the accident. Knowing these conditions supports future development of analytical tools for assessing core damage events and allows insights on possible mitigation strategies.	1. Visual inspection 2. Metallurgical exam 3. Displacement measurements

Item	What	Applicable Unit(s)	Why	How
RPV-12	Shroud inspection (from core region)	Units 1, 2 & 3; as applicable	Possible effects of melt relocation	Visual inspection
RPV-13	Inspect core plate and associated structures	Units 1, 2 & 3; as applicable	Extent of core melt progression, melt flow path, and failure mechanism	Visual inspection
RPV-14	Inspect control drive and instrument guide tubes (if present), as they exit the lower plenum, to assess if core debris could have entered the tubes from within the RPV. Tubes may need to be cut out and examined to determine if core debris had relocated within the tube.	Units 1, 2 & 3; as applicable	Extent of core melt progression, melt flow path, and failure mechanism Core debris within the instrument tubes provides insights on how the vessel may have been breached.	Visual inspection Pressure boundary inspection for both hydraulic lines and guide tubes Look for TIP failures.
RPV-15	Extent of debris in lower head (if present) • Area coverage • Topography of debris surface • Chemical composition • Morphology • Depth of debris	Units 1, 2 & 3; as applicable	Understand in-vessel melt progression Explicitly consider the following phenomena (needed for more accurate code predictions of coolability and time for mitigating strategies): • "Regional" (axial and radial) variability • Evidence of peak temperature (e.g., composition, morphology indicating previously melted, cracking, debris-to-structure gaps, etc.) • Evidence of salt effects (e.g., U-Cl compounds)	1. Visual inspection (camera/probe) 2. Debris sampling (core "plug" or "bore" samples) 3. After inspections are completed, additional R&D (that includes separate effect laboratory tests) may be needed to interpret observations.
RPV-16	Inspect lower head and lower head penetrations. The BWR drain line and associated structures should be examined. Look for evidence of enhanced heat transfer associated with external lower head penetration structures. If the drain line is intact, look at the	Units 1, 2 & 3; as applicable	Extent of core melt progression, melt flow path, and failure mechanism Material interactions with penetrations and vessel head materials The BWR drain line is the most logical point of failure for several reasons: • It is located at the bottom and center of the lower head	1. Visual inspection 2. Metallurgical sampling 3. After inspections are completed, additional R&D, that includes separate effects laboratory tests, may be needed to interpret

Item	What	Applicable Unit(s)	Why	How
	debris near it to understand why it didn't fail.		• There are no in-vessel structures associated with it (that would require melt though before relocated core materials would travel ex-vessel) • It has an open geometry for relocated melt to travel ex-vessel • It is composed of lower strength, SA105/SA 106 steel (as opposed to stainless steel) Although some hand calculations were performed to assess the effects of enhanced heat transfer associated with these external penetration structures, the radiation and convective heat transfer associated with such enhanced heat transfer is not included in severe accident analysis codes. It would be worthwhile to look for evidence suggesting that could help quantify this heat transfer. Post-TMI-2 evaluations found that in-vessel instrumentation nozzles embedded in debris were intact, whereas remaining stubs of nozzles were located in areas with lower debris height (but in the hot spot region of the vessel lower head). Code assumptions that relocated debris is uniform and axi-symmetric are not accurate but there are insufficient data for improved characterization. Increased knowledge on this area would be useful in future predictions of vessel failure and in formulating accident management strategies.	observations.

A-11

Attachment A

REACTOR BUILDING

Item	What	Applicable Unit(s)	Why (what question are we trying to answer?)	How
RB-1	RCIC Systems for both Units 2 and 3 Position/condition of RCIC valves, including check valves (unit 2), and turbine governor valves. Which valves required DC power? Where do RCIC turbines exhaust to? Visual inspection should include an identification of the possible failure mode and any leak pathways from the RCIC system into the RCIC room. It is supposed that unregulated operation of the RCIC System in Unit 2 was possible because the turbine governor valve froze in a position favorable for removal of steam at a rate corresponding to two-to-three days along the decay heat curve. Confirmation of the valve position would enable us to know whether this situation occurred, and whether the RCIC turbine somehow tolerated ingestion of two-phase flow for a large portion of its operating position. A closed valve would be inconclusive in this regard but would inform us that the RCIC System shut down due to valve closure at about 20 hours (Unit	Units 2 & 3: Hard to soft vent path for unit 3 Unit 2: rupture disc on hard vent	Failure mode; uncontrolled operating history. Why did Unit 2 RCIC perform much better than Unit 3 RCIC? How can we maximize RCIC reliability under accident conditions? Obvious damage to RCIC turbine in Unit 3 vs. mere trip (underwater now, may be hard to inspect) The RCIC system operated well beyond design on Unit 2. This inspection data will help to better understand the capabilities and possible limitations of the system and inform possible system changes or operating strategies. Inspection data can be compared to manufacturer specifications on each component to better understand system and component survivability under accident conditions. Operation of RCIC under beyond design basis conditions is a potential R&D topic. The current BWR integrated plans in response to NRC Order EA-12-049 for Mitigating Strategies involves the use of RCIC for extended periods of time. Information on the long term response of the system would support those strategies. 1) Barrett [1973][1] reports an experimental program in which the Terry turbines used in RCIC Systems are remarkably robust in situations where water slugs enter the turbine. The reason for these conclusions and whether severe accident conditions were within the scope of this report are unclear. 2) The location of the pump suction with respect to the turbine discharge may be close. Plant data are	Visual inspection

[1] Barrett, K. E. "Terry Wheel Water Slug Test", Terry Corp. Engin. Library Log No. 20,106, Mar. 1, 1973

A-12

Attachment A

Item	What	Applicable Unit(s)	Why (what question are we trying to answer?)	How
	3) and near 70 hours (Unit 2) and termination of steam into the turbine.		not publicly available. 3) The RCIC System was designed for isolation events, not accident scenarios. However, in Units 2 and 3, it played in important role in decay heat removal. 4) The SP temperature may have reached 302°F, causing some of the RCIC turbine bearings to "wipe" and therefore the turbine destroyed itself.	
RB-2	HPCI	Units 2 & 3		Visual inspection
RB-3	Look for evidence that could provide insights related to the observed explosion and reduce uncertainties in code predictions for hydrogen generation and transport from Unit 3. Look for visual indicators on the walls, radionuclide deposits in the ductwork, etc., that could help reduce late phase hydrogen generation predictions.	Unit 4	Post-TMI-2 evaluations found damage in non-related components, such as a melted phone, that provided indicators of peak containment temperature. Look for the unexpected.	
RB-4	Look in the reactor building for evidence (damage outside the containment) that could provide insights related to depressurization mechanism and hydrogen generation.	Unit 2		
RB-5	Reactor Building Cesium Distribution: Samples within the reactor buildings should be taken to determine the fission product leak paths from containment. Specifically, sample at the following locations: 1. Within vent piping and interfacing systems to determine leakage from hard vent to the softer vent piping. 2. Drywell head area and shield plugs to determine drywell head leakage.	Units 1, 2 & 3; as applicable	Leakage from the hard pipe system into lower pressure piping is important to understand how well the existing vent system functioned. These systems are being updated around the world and this information will inform what designs work best to isolate the leakage from the hard vent system. This information will also support leak rate analysis from proposed filtered venting systems.	

Item	What	Applicable Unit(s)	Why (what question are we trying to answer?)	How
RB-6	Radionuclide Sampling of Ventilation Ducts in Unit 4: Perform sampling of the ventilation ducts that connect Unit 4 to the common header to determine the existence of lanthanides. The presence of lanthanides in the ventilation ducts could possibly indicate that carbon monoxide could have been pushed from unit 3 into unit 4 prior to the explosion event.	Units 1, 2 & 3; as applicable	The daughter products from the dominant isotopes for the noble gases result in different isotopes than those produced directly from nuclear fission; therefore, regions that had a high concentration of noble gases with lower amounts of volatile fission products should contain a different ratio of isotopes for Cesium and Rubidium, possibly indicating the previous existence of hydrogen. Likewise, carbon monoxide would have carried fission products that are released as a result of MCCI. Therefore, a significant deposition of lanthanides could indicate the existence of a large amount of carbon monoxide.	

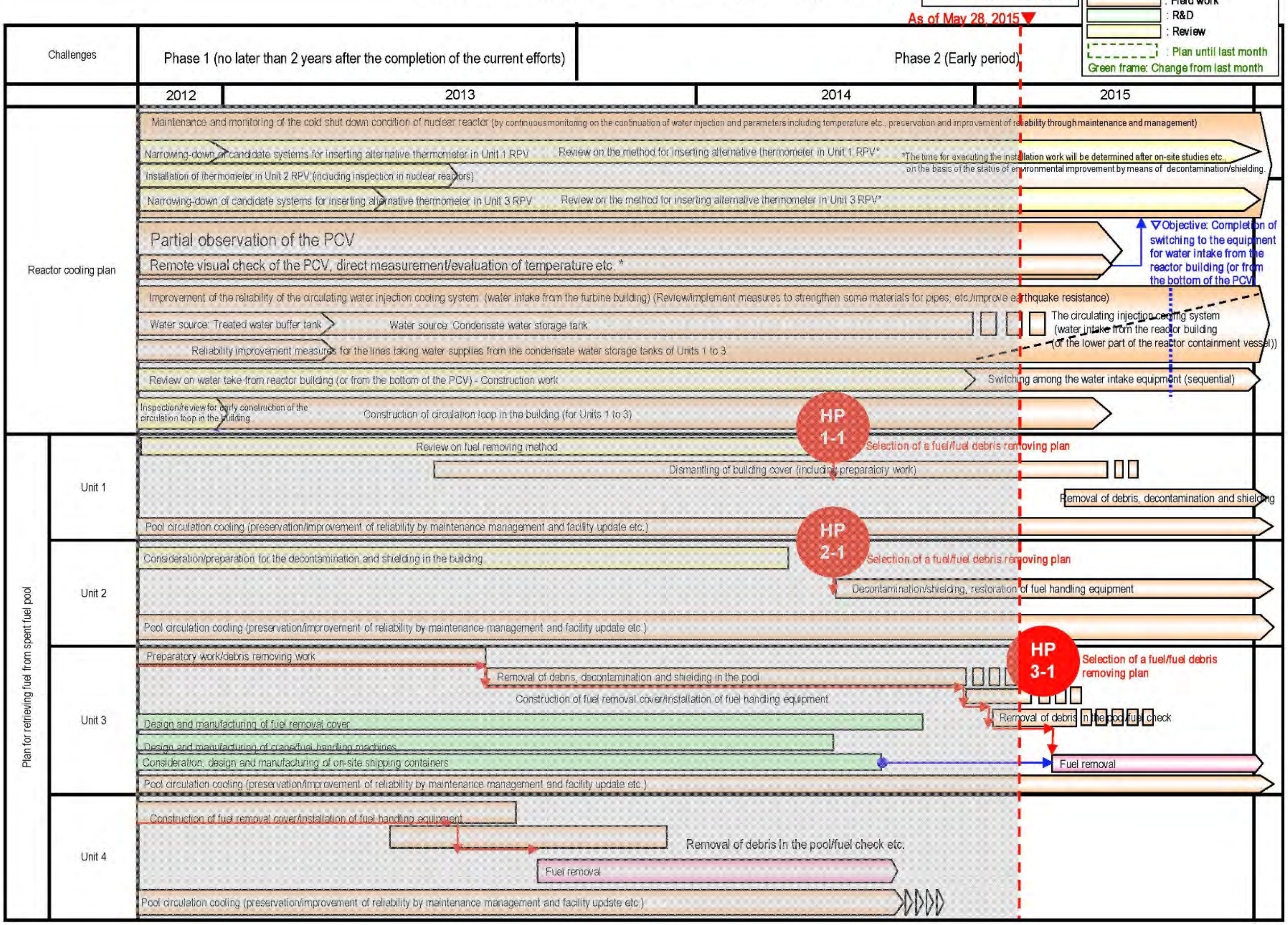
Appendix D

Mid-and-Long-Term Roadmap

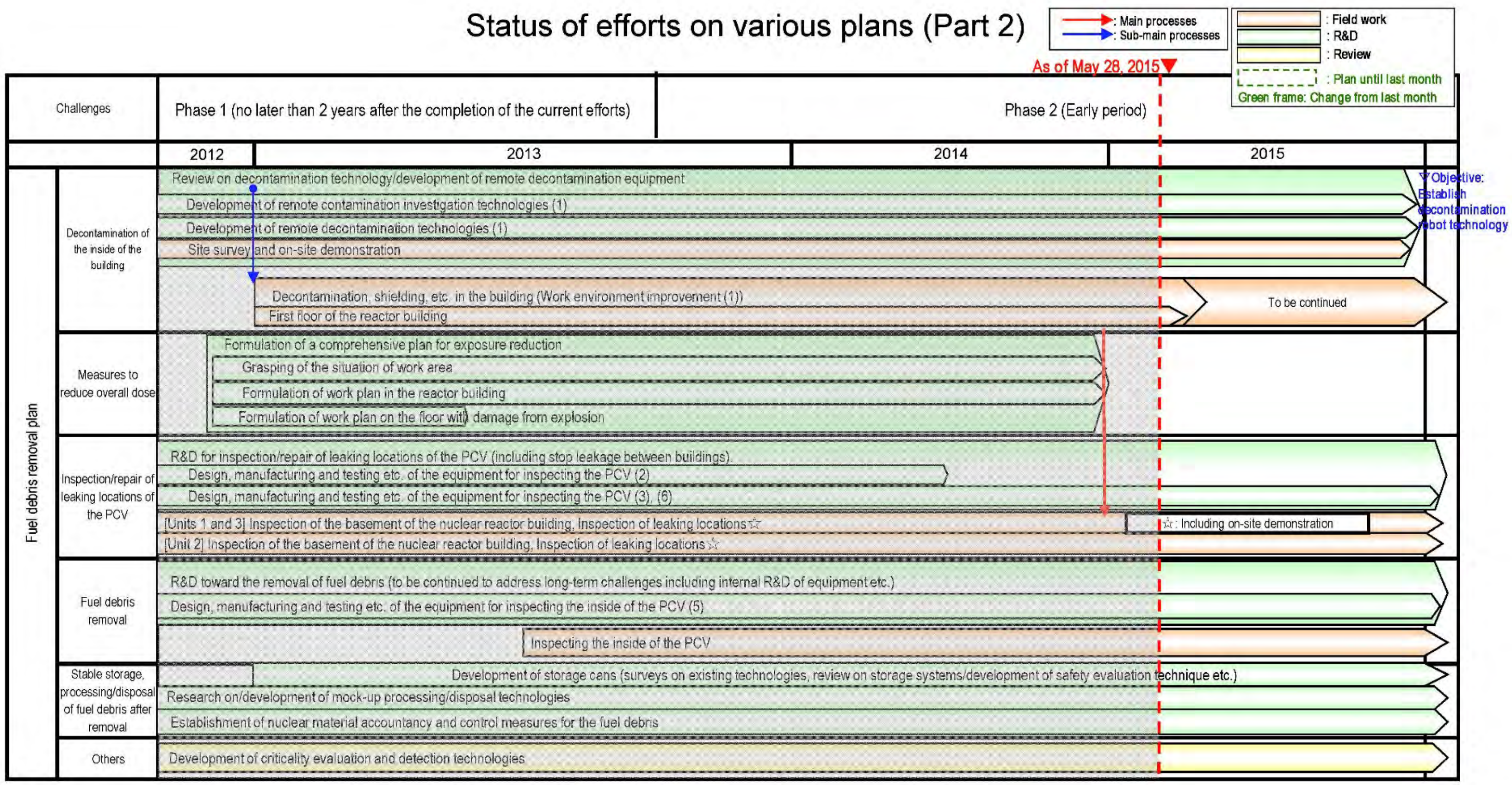
[Roadmap provided courtesy of TEPCO; Reference 40]

Attachment 3

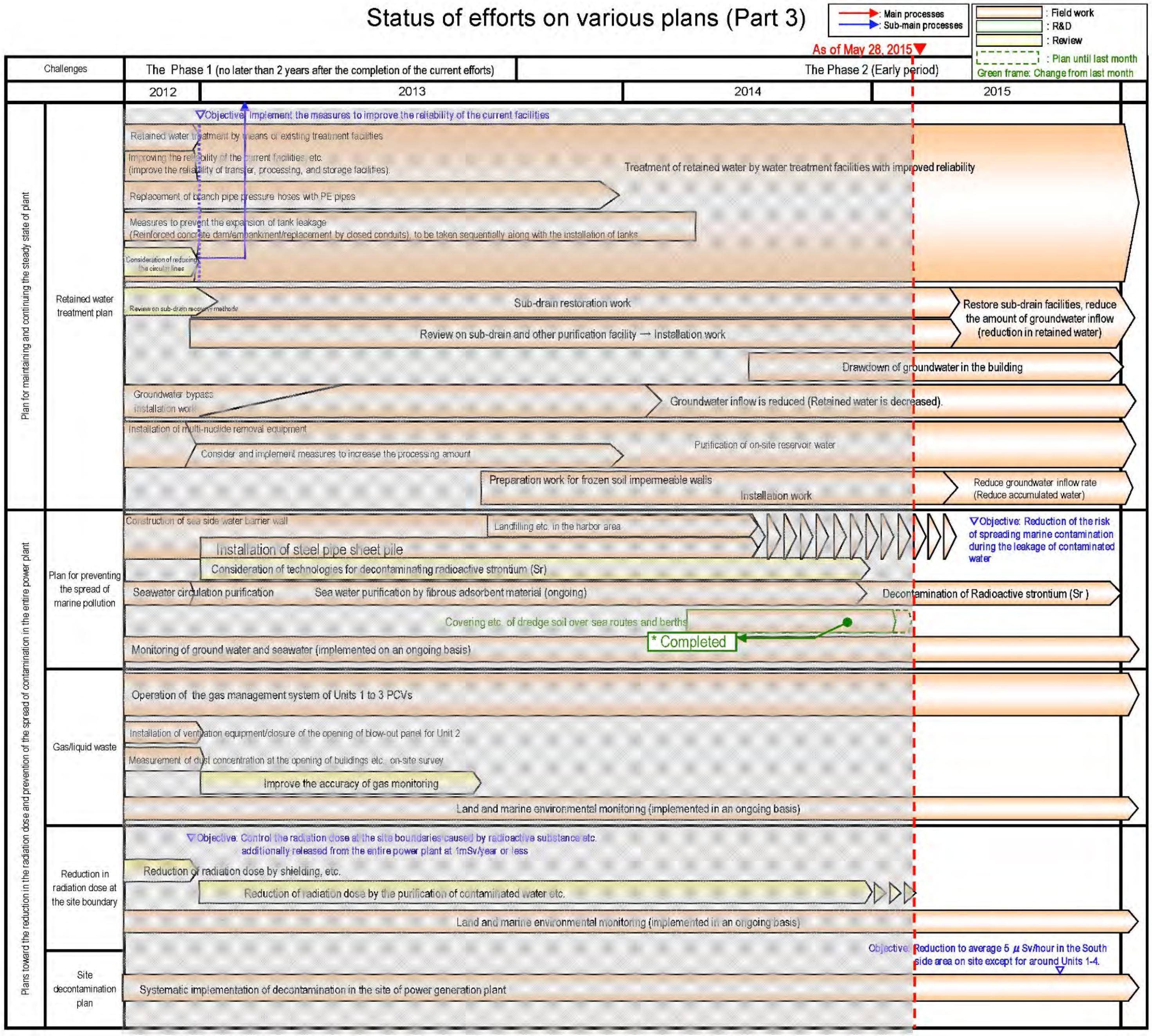
Status of efforts on various plans (Part 1)



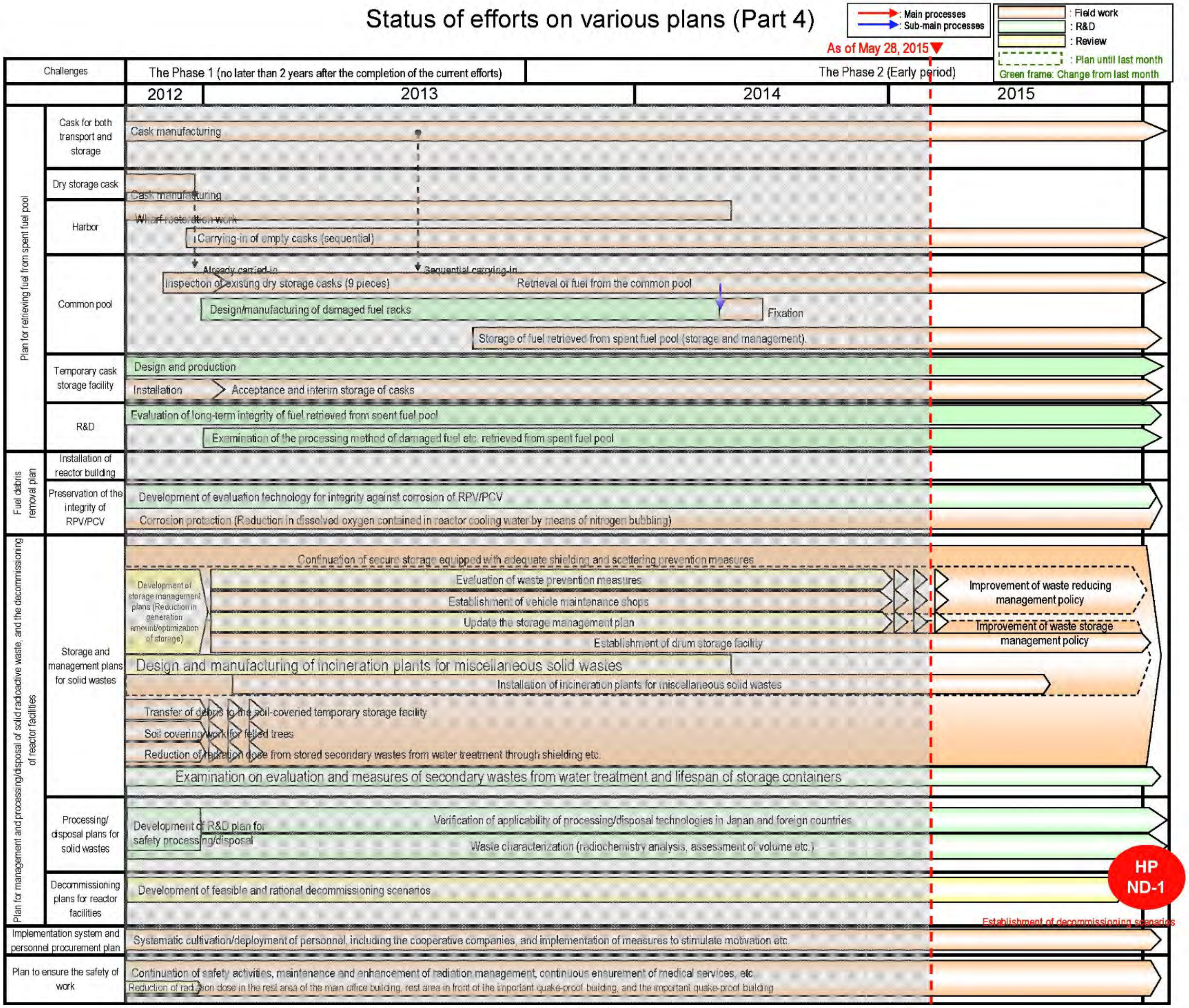
Status of efforts on various plans (Part 2)



Status of efforts on various plans (Part 3)



Status of efforts on various plans (Part 4)



Appendix E

Input from US Industry

<table>
<tr><th colspan="2">Table E-1. Initial Industry Input for Daiichi/Daini Data Needs[w]</th></tr>
<tr><th>Fukushima Daini</th><th>Fukushima Daiichi</th></tr>
<tr><td colspan="2">Reactor Core Isolation Cooling</td></tr>
<tr><td>[See Table B-4, Item O-15]
• Complete forensic disassembly of the RCIC turbine and pump with measurements of clearances and visual conditions via pictures.
- Inspection of RCIC pump impeller/wearing rings for cavitation related damage
- Oil sampling from RCIC turbines operated under event conditions or sample results if sampling and analysis is already completed
- Pump mechanical seal condition
• Document RCIC operational history during the event to include:
- Length of time RCIC operated during post event response
- Trips/restarts of RCIC
- Actions taken to preserve RCIC operation
- Local operation of RCIC performed during event response
- Temperature history of RCIC suction during post-ELAP operation</td><td>[See Table B-4, Item O-15 and Table B-1, Item RB-1]
• With similar information available from Fukushima Daini, most work can be done on components with minimal radioactive contamination. Examinations may not need to be repeated for Fukushima Daiichi except as noted below:
- Check for evidence of damage to the RCIC turbine caused by extended operation in the self-limiting mode
• Determine RCIC failure mode(s) for RCIC on U2 [see Table B-1, Item RB-1]</td></tr>
<tr><td colspan="2">Core Debris and Reactor Fuel Performance</td></tr>
<tr><td></td><td>• Characterize core debris spreading in the primary containment. [See Table B-2, Item PC-3]
• Check for evidence of molten core debris intrusion into penetrations at the bottom of the reactor vessel (e.g., instrument tubes, control rod guide tubes) and conditions of the penetrations.) [See Table B-2 Item PC-3]
• Characterize final core debris configuration (e.g., fragmented debris bed, re-frozen large metallic/oxide mass, etc.) [See Table B-2, Item PC-3 and Table B-3, Item RPV-5]
• Evidence of final disposition of salt from seawater injection (finely dispersed or concentrated masses). [See Table B-2 Item PC-3 and Table B-3, RPV-5]
• Inspect a sampling of Unit 5/6 irradiated fuel for impact of loss of cooling for short durations under various temperature/pressure conditions. [See Table B-4, Item O-14]</td></tr>
</table>

[w] Brackets denote locations where industry needs have been incorporated into Appendix B tables.

<table>
<tr><th colspan="2">Table E-1. Initial Industry Input for Daiichi/Daini Data Needs[w]</th></tr>
<tr><th>Fukushima Daini</th><th>Fukushima Daiichi</th></tr>
<tr><td colspan="2">Reactor Recirculation Pump Seals</td></tr>
<tr><td>[See Table B-4, Item O-13]
Inspect Reactor Recirculation pump seal condition to evaluate integrity for high pressure and temperature impacts.</td><td>[See Table B-4, Item O-13 and Table B-2, Item PC-11]
Fukushima Daini should have experienced similar pump seal transient conditions as Fukushima Daiichi. Should not need to be repeated for Fukushima Daiichi, which will have extreme levels of contamination and are not currently accessible.</td></tr>
<tr><td colspan="2">Containment/RPV Performance</td></tr>
<tr><td>• Determine degree of galvanized and aluminum oxidation in the primary containment and evaluate in terms of a source for hydrogen production. [See Table B-2, Item PC-16].</td><td>• Vent piping and rupture discs are not accessible in the foreseeable future. [see Table B-1, Item 11; Table B-4, Item 12].
• Check for evidence of degradation of primary containment penetration seals due to accident conditions (elevated containment temperature and pressure). [see Table B-1, Item RB-10; Table B-2, Item PC-1].
• Check for degree of core concrete interaction, significance of concrete oblation. (See Table B-2, PC-3d).
• Determine degree of galvanized and aluminum oxidation in the primary containment and evaluate in terms of a source for hydrogen production. [See Table B-2, Item PC-16].
• Check for evidence of SRV degradation due to passing highly superheated fluid [See Table B-2, Items PC-5, PC-12, and PC-13].
• Determine location of RPV lower head failure (penetration, tear, oblation, and catastrophic global failure) and mode of failure (e.g., creep, weldment melting, etc.). [See Table B-2, Item PC-3d].
• Check for evidence and degree of melting or change on properties of equipment and structures within the primary containment. [See Table B-2, PC-16].</td></tr>
<tr><td colspan="2">Aging Management</td></tr>
<tr><td>[See Table B-4, Item O-11].

• Inspect cable routing and splice conditions four years post event to determine if field quality work has a high success rate.
• Perform tan delta testing of cables for water intrusion benchmarking.
• Inspect buried piping for seismic and water intrusion impact (action could be done at either site but does not need to be done for both sites).</td><td>[See Table B-2, Item PC-7 and PC-14 and Table B-4, Item O-11].

• Inspect accessible cables for radiation life impacts and for water intrusion.
• Inspect buried piping for seismic and water intrusion impact.
• Inspect condition of cabling (e.g., insulation damage, electrical conductivity, evidence of heat induced transmission changes, etc.) in the reactor building and in the region of the hydrogen explosion.</td></tr>
</table>

Table E-1. Initial Industry Input for Daiichi/Daini Data Needs[w]	
Fukushima Daini	**Fukushima Daiichi**
Monitoring and Instrumentation	
[See Table B-4, Item O-10]. • To the extent possible, plot containment pressure and temperature response during the event from start of the event until plant cool down or cold shutdown condition. • Calibration with emphasis on instrument drift for containment instrumentation for qualification benchmark.	[See Table B-2, Items PC-7 and PC-8 and Table B-4, Item O-10]. • Determine failure modes for any significant instrumentation (e.g., instruments beneficial for event response) that failed during the event. • Review significant instrumentation (instruments beneficial for event response) fidelity to identify any response of parameter values that were possibly influenced by beyond design basis instrument conditions created by the accident progression (e.g., the RPV level instrumentation impact due to loss of water in the reference leg caused by high primary containment temperature conditions)Review operational records, available instrumentation records and plant conditions for possible enhancements to Emergency Operating Procedure/Severe Accident Guidance bases document.
Coatings	
Photos/videos of drywell and wetwell coating integrity for steam condition impacts inside containment during the event.	[See Table B-2, PC-9 and PC-14]. Photos/videos and samples of U1, U2, and U3 PC (Suppression chamber and DW) coatings.