CNWG TE-4:
Severe Accident Code Assessment

US-Japan
BLC-CNWG

SAND2017-5181PE

Co-Chairs:

US: Randall Gauntt (Sandia); JAPAN: Masanori Naitoh (IAE)

Fifth Meeting of the

Civil Nuclear Energy Research and Development Working Group

May 16-18, 2017

Idaho Falls, USA

Severe accident code assessment

New items in red

Scope of Work

The scope of activities in this thrust is to mutually exchange information on the Fukushima Daiichi accidents researches developed by each country with the aim of: assessing severe accident code capabilities, identifying gaps in knowledge and supporting decommissioning activities.

Tasks (agreed at the previous CNWG Meeting, Feb. 17-19,2014 and new terms in red)

Task 1: Cooperative Assessment of the Accidents at Fukushima and Shared Development of Uncertainties and Best Modeling Practices

(1-1) MELCOR uncertainty analysis on Fukushima Daiichi NPP

(1-2) MELCOR analysis of Fukushima Daini

(1-3) Experimental activities of emergency equipment (***RCIC Analysis and Testing)***

(1-4) MELCOR-SAMPSON crosswalk

Task 2: Model enhancement of MAAP and SAMPSON, analyses of accident progression of Fukushima Daiichi NPP and Computational Fluid Dynamic (CFD) activities

Task 3: Fundamental experiments on (1) sea water effect and (2) on debris behavior in BWR lower plenum with simulant (complete)

Task 4: Hydrogen explosion analyses for Fukushima Daiichi NPP and experiments for phenomena which were specific to Fukushima Daiichi NPP accident for code validation

(4-1) CFD analysis of hydrogen explosion

(4-2) Experimental activity on accident phenomenology (steam condensation experiments)

(4-3) Experimental activity on lower head penetration failure

Task 1 and 2: Cooperative Assessments of Fukushima Accidents and BSAF Project

- **IAE has served as operating agent for the OECD-NEA BSAF project focused on code analysis of Fukushima accidents and reconstruction of the principal accident signatures.**
- **Phase I of BSAF project was completed in December 2014. Final and summary reports were released, with a main focus on the thermal hydraulics and code damage progression in the accidents**
- **Phase I Best Estimate calculations were presented at the NURETH-16, Sep. 2015, Chicago and selected for special issue publication in Nuclear Technology**
- **Phase II started and will be focused of release and transport of radioactivity throughout the plant (containment and building) and to the environment (3 years activity, 2015 - 2018)**
- **Next BSAF Phase II meeting is being held in Tokyo during July 10-14.**

Task 1-1 MELCOR Uncertainty Analysis of Fukushima Daiichi Unit 1 Damage State

SANDIA REPORT

SAND2016-0232
Unclassified Unlimited Release
Printed January 2016

Fukushima Daiichi Unit 1 Accident Progression Uncertainty Analysis and Implications for Decommissioning of Fukushima Reactors – Volume I

Prepared by
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Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

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Sandia National Laboratories

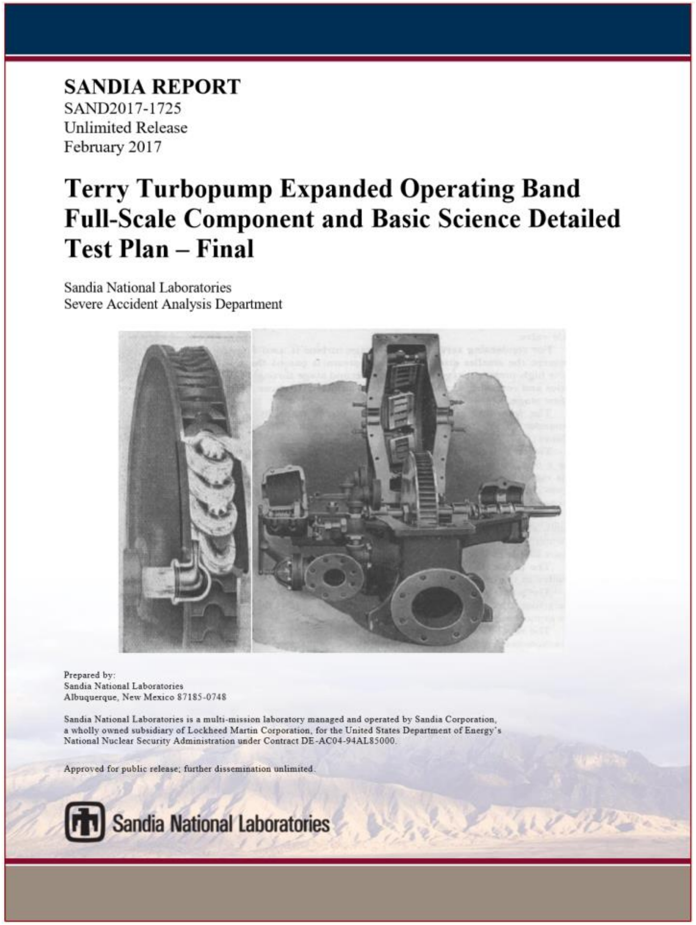
- Work leveraged with OECD/NEA BSAF cooperation
- Parameter uncertainty affecting melt progression
- Monte Carlo sampling
- Focus on providing ranges of core degradation for Unit 1
  - Core damage extent
  - Amount in lower head
  - Debris relocated to reactor cavity

Task 1-3 Experimental Activities of Emergency Equipment (RCIC Analysis and Testing) (1)

- **Investigate realistic operational limits of Reactor Core Isolation Cooling (RCIC) System in BWRs**
  - Supplies makeup/feedwater to reactor under containment isolation events
    - RCIC Turbine powered from reactor steam, which exhausts to the Suppression Pool
    - RCIC Pump draws water from Condensate Storage Tank or the Suppression Pool and sends it to the reactor
  - RCIC Pump is driven directly by a Terry turbine
    - Terry turbines are very rugged and can ingest large amounts of water without damage
    - Performance expected to degrade but poorly understood with two-phase flow in turbine inlet
    - Two-phase ingestion may provide a means for 'self-regulation' of the RCIC System in the absence of electrical power to the governor electronics
- **Research collaboration between US & Japan**
  - Involves US Industry, US DOE, METI/IAE (Japan)
  - Major experimental effort at Texas A&M University
  - Modeling and experimental efforts inform each other
  - Efforts overseen by Turbo-TAG committee
- **Experimental Collaboration: on-site participation at TAMU from multiple organizations**
  - TAMU researchers
  - One SNL researcher on-site at TAMU
  - One IAE researcher on-site at TAMU
- **Modeling efforts at SNL, TAMU, IAE**

Task 1-3 Experimental Activities of Emergency Equipment (RCIC Analysis and Testing) (2)

- **Milestone-based approach**
  - Milestone 1&2: Scoping and planning
  - Milestone 3: Component-level (i.e., valves, bearings) investigations
  - Milestone 4: Turbopump basic science investigations
  - Milestone 5: Full-scale system testing
  - Milestone 6: Full-scale self-regulating feedback testing
  - Milestone 7: Modeling efforts collected from Milestones 3-6
- **Completed Milestone 3 & 4 detailed test plan**
  - SAND2017-1725
- **Finalizing funding contracts, NDA, and Charter amongst all stakeholders with TAMU**
  - DOE/INL-TAMU contract: In Place
  - METI/IAE (Japan)-TAMU contract: In Progress (expected soon)
  - PIM (representing US Industry)-TAMU contract: In Progress (expected soon)
- **Presented efforts at BWROG-EPC, BWROG-RCIC ExOB Committee, and PWROG SAMG core team**
- **Upcoming efforts**
  - Start experimental efforts at TAMU in near future
  - Present efforts to BWROG-RCIC ExOB Committee, EPRI-TTUG, PWROG, and ASME/NRC Pump & Valve Symposium
  - Continue modeling efforts complementary to experimental efforts (system-level and CFD modeling)
  - Start scoping Milestone 5 efforts and domestic testing facility


SANDIA REPORT
SAND2017-1725
Unlimited Release
February 2017

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

Sandia National Laboratories
Severe Accident Analysis Department

Prepared by:
Sandia National Laboratories
Albuquerque, New Mexico 87185-0748

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Task 1-3 Current US Industry Phased RCIC Planning - under development-

Phase	Deliverable	Estimated Duration
Phase 1, Plan Development	Project Plan containing: Detailed benefit, cost, schedule including off-ramps, resource options, project QA/QC aspect	3 months
Phase 2, First Principle Analytical Modeling (Modeling will continue throughout other phases to refine scope in those phases)	Model revisions, analyses, and input to the prototype and small scale testing designs	3-6 months
Decision for next Phase		
Phase 3, Prototype Testing	• Testing results to influence small and large scale testing • Test results to provide anticipated benefits of some enhancements	3 months for setup 6-9 months for test
Decision for next Phase		May include step wise approval of next phase
Phase 4, Small Scale Testing	• Testing results to influence large scale testing • Test results to provide anticipated benefits of some enhancements	4-6 months for setup 9-15 months for tests
Decision for next Phase		Will include step wise approval of next phase
Phase 5, Large Scale Testing	Test results	12-15 months for setup 60 months for tests

Task 1-4 MELCOR-SAMPSON Crosswalk

- **Expansion of MELCOR-MAAP Crosswalk performed with EPRI**
- **Same accident transient**
  - Stylized Fukushima-Daiichi Unit 1
  - SRV failure at 7 hours
- **Highlights different physical treatments**
  - Thermal-hydraulic
  - Phenomenological
  - Relocation transients
  - Molten vs. particulate
- **Benefits**
  - Provide bounding cases for industry partners (SAWA, SAWM)
  - Ensure challenges to containment are appropriately captured
    - Hydrogen production, steam discharge

Task 1-4: MELCOR – SAMPSON CROSSWALK
Main results

Similarities

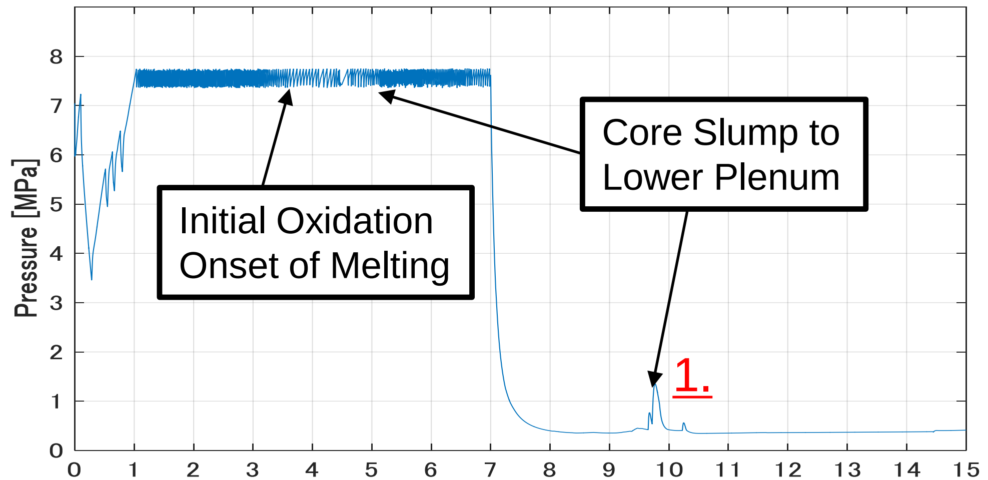
- ✓ Depressurization rate after SRV seizure
- ✓ Pressurization at the timing of core debris relocation
- ✓ Main core slumping occurs at similar timing.
- ✓ Pressurization in main core slumping is similar.

Differences

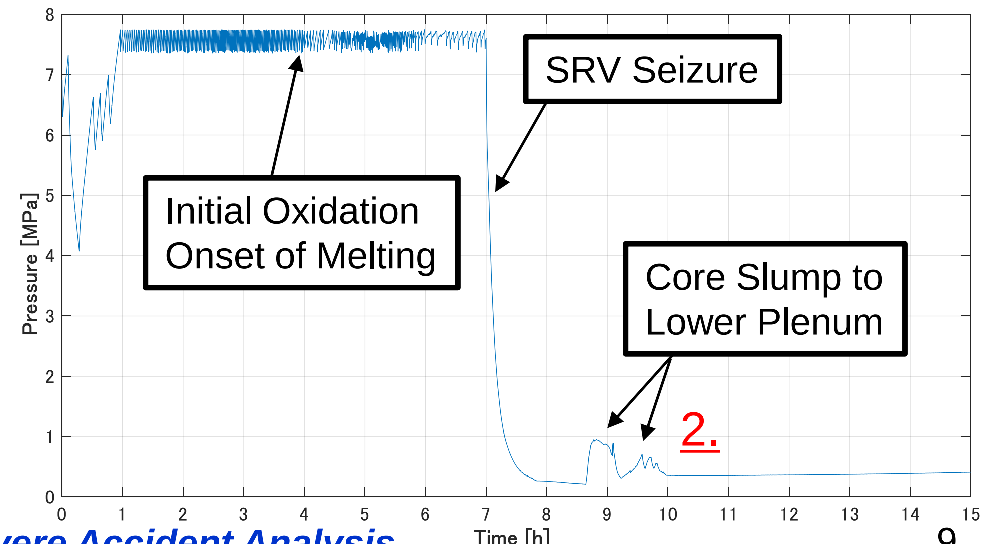
1. MELCOR predicts initial core slumping at 5.1 h due to core support plate failure. Only area below 1st ring fails.

3. SAMPSON simulate main core slumping at 8.6 h due to core support plate failure. Whole area fails.

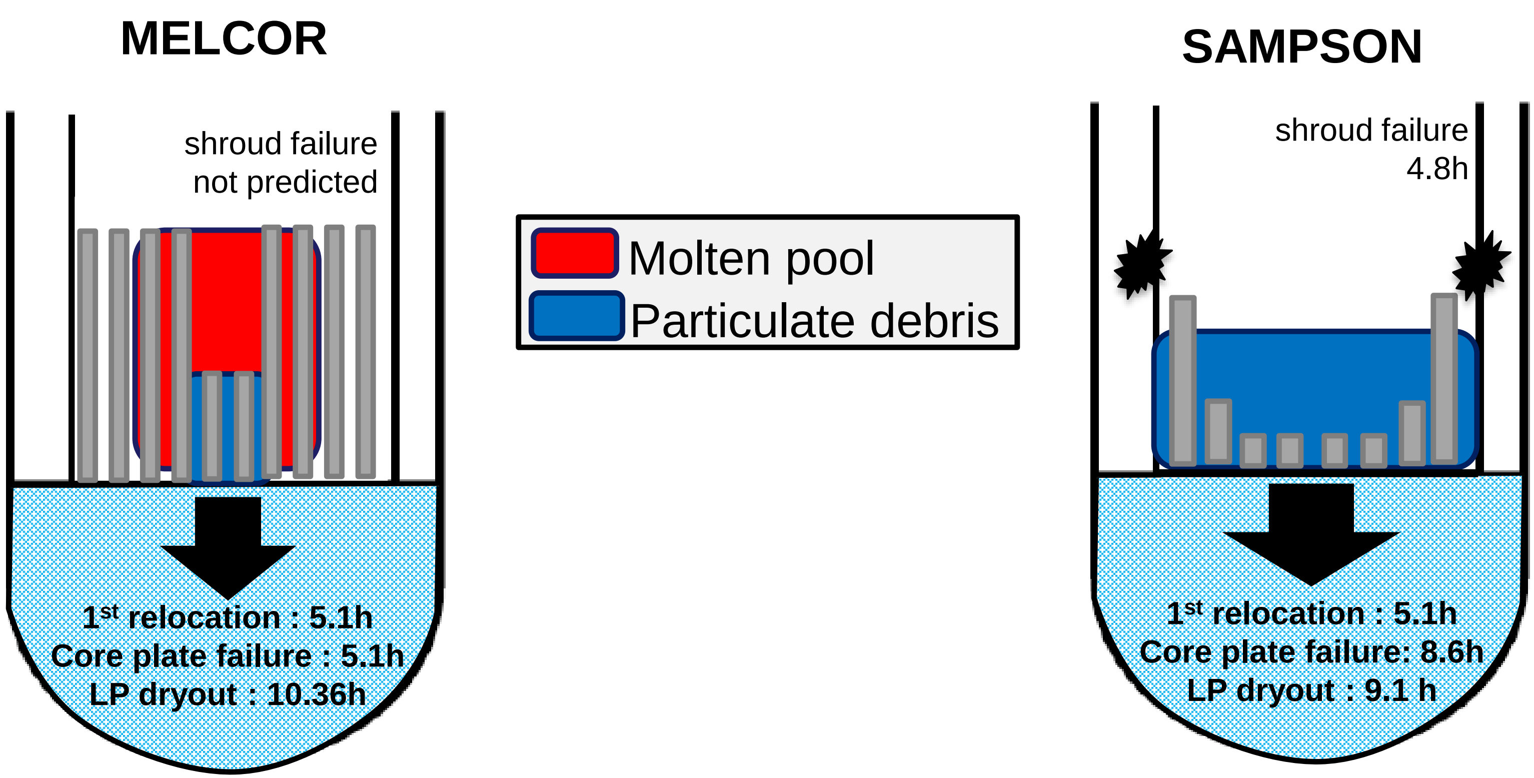
MELCOR



SAMPSON



Task 1-4: MELCOR – SAMPSON CROSSWALK
Debris distribution and technology

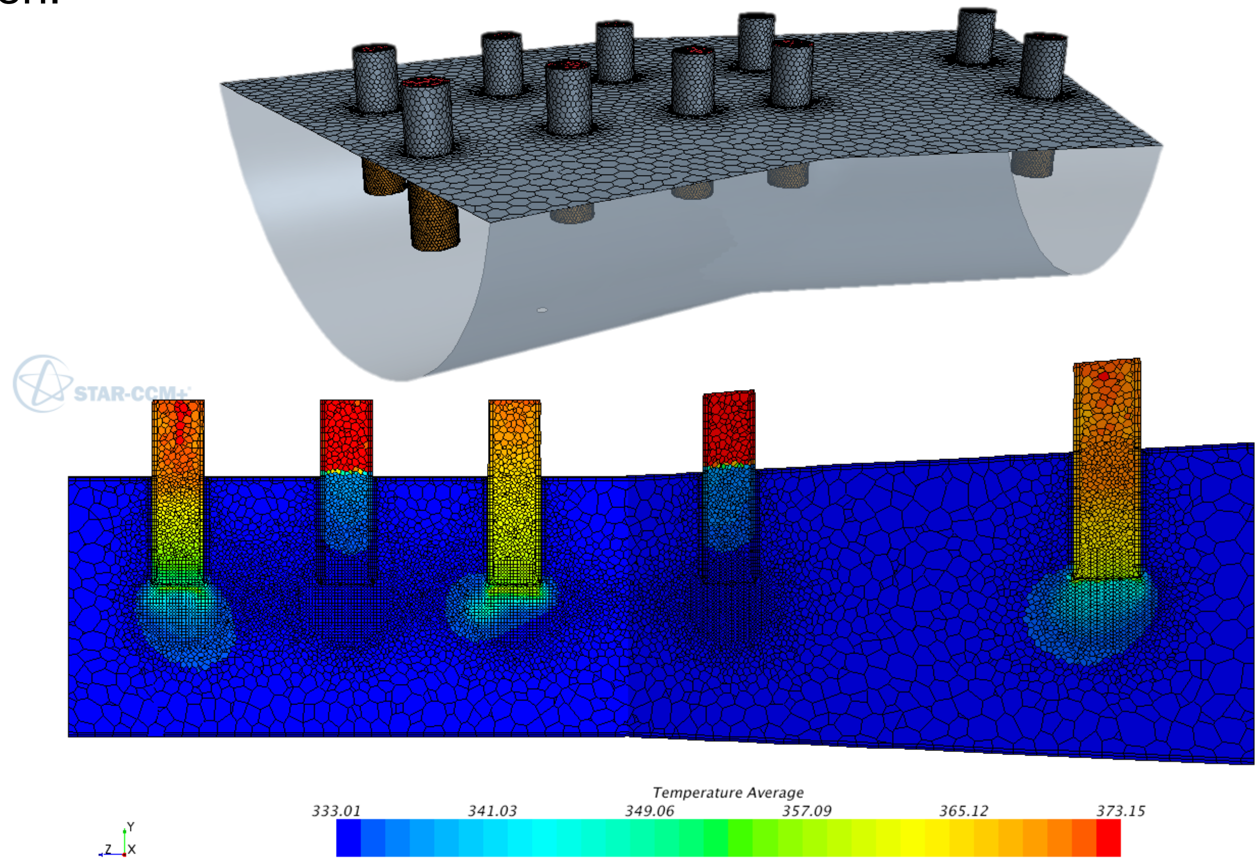


- 10 tons molten pool and 5 tons particulate debris
- molten debris begins to relocate through core plate failure
- molten pool keeps growing until RPV failure at 14.45 h

- 55 tons particulate debris (fuel in 4th ring is exposed without cladding)
- particulate debris begins to relocate through the inlet orifice of the fuel lower structure
- 1 hour to form main particulate bed

Task 2 CFD Analysis of Large Scale S/C

- Application of Multi-phase CFD to the a sector of the whole S/C in order to evaluate temperature stratification and reconsider scrubbing models is ongoing.
- Major concern is thermal-hydraulic behavior in S/C with mixture of steam and hydrogen.



Task 2: Model enhancement of SAMPSON, analyses of accident progression – 3D S/C pool

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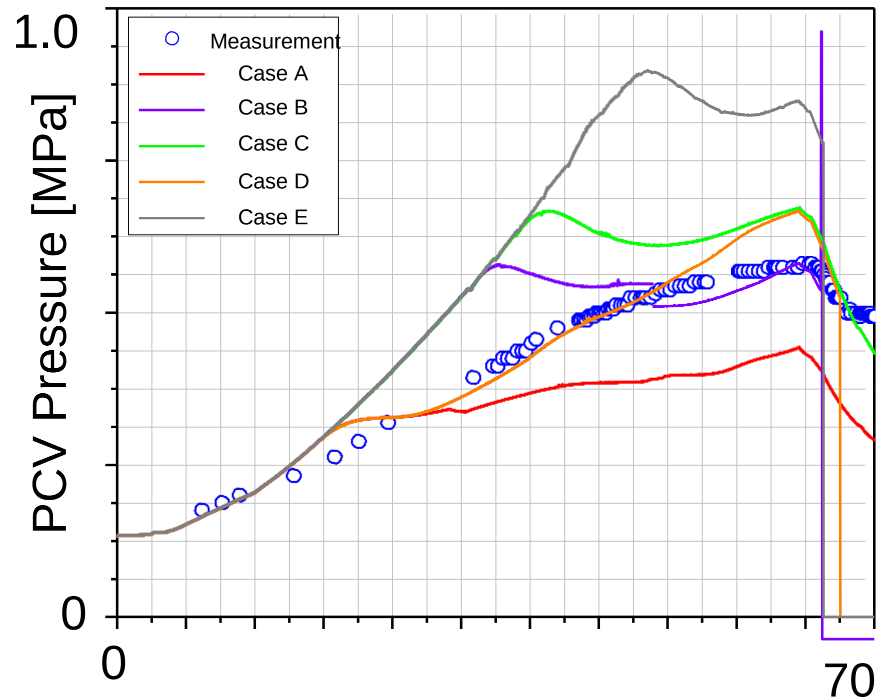
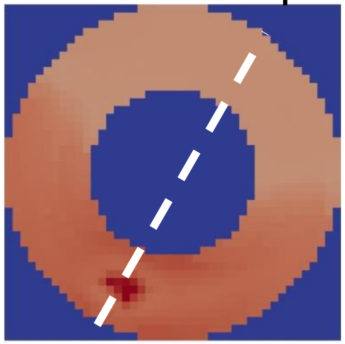
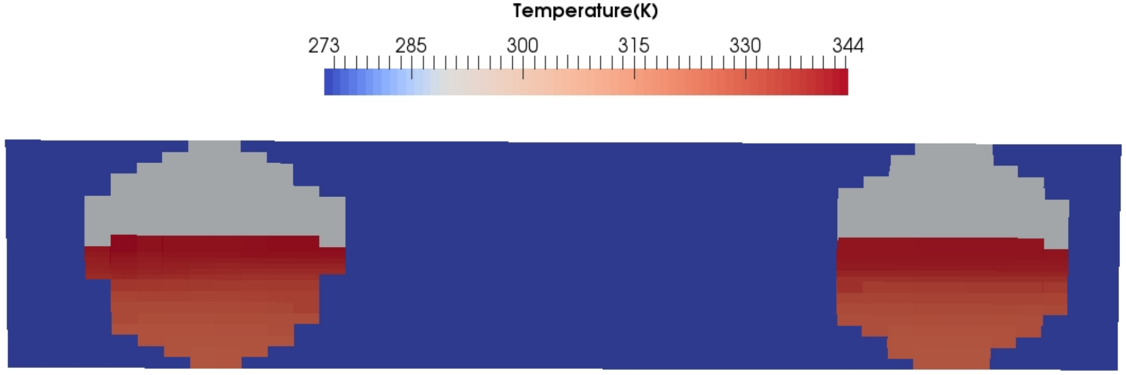
- Sensitivity analysis has been performed to confirm torus room flooding conditions and mixing induced by steam condensation
- Such activity is complementary with the activity to be developed for the RCIC project



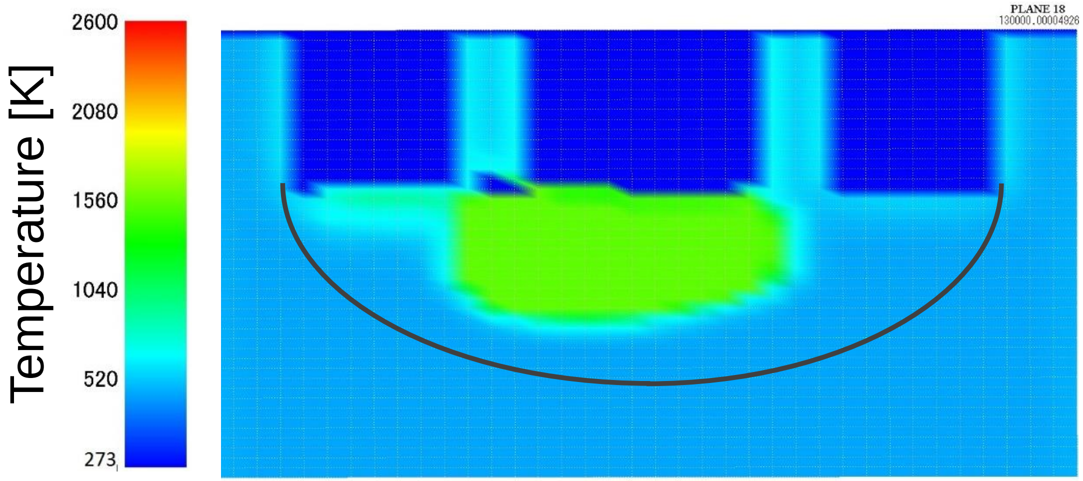
Surface temp.

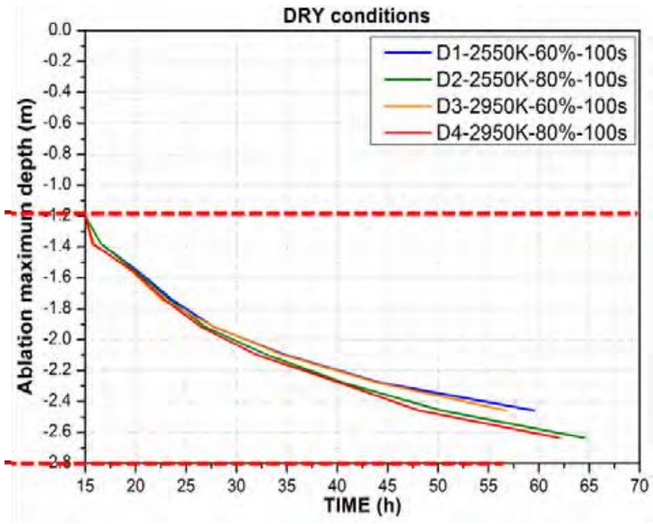


S/C temperature distribution

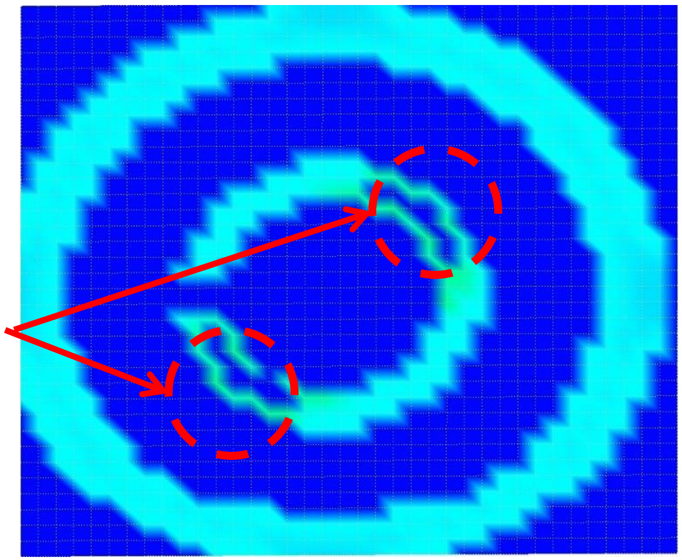


Task 2: Model enhancement of SAMPSON, analyses of accident progression – 3D MCCI





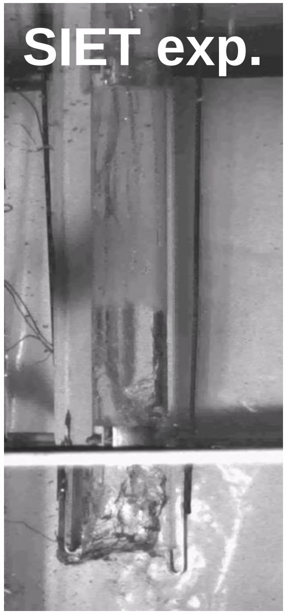
Ablation of pedestal walls

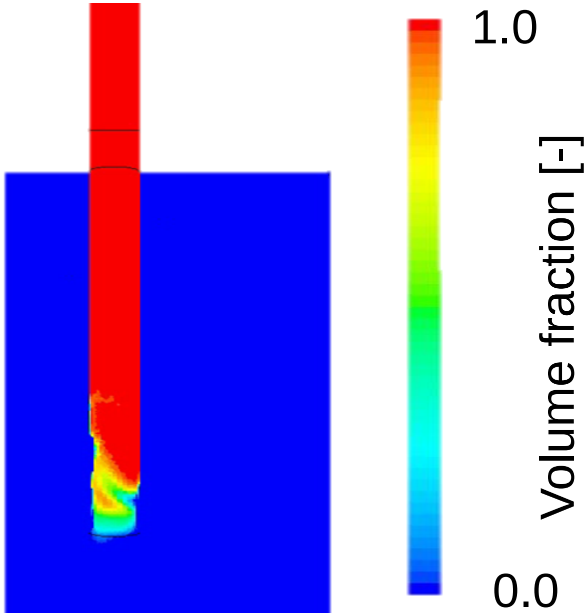


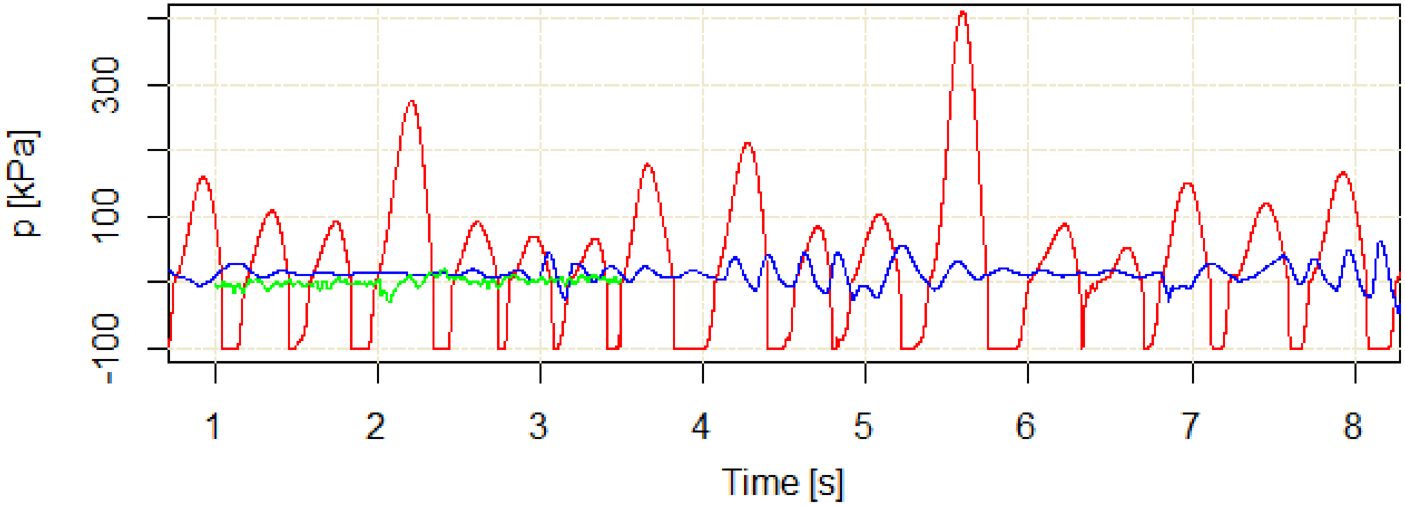
Sensitivity analysis has been performed showing ablation preferential ablation along the sump pits affecting the pedestal walls

Task 4-2: Experimental activity on accident phenomenology (steam condensation)

- We are continuing the CFD validation of the SIET experiment (bottom-opened pipe)
- We have reached most realistic steam pressure signatures due to usage of more realistic physical properties



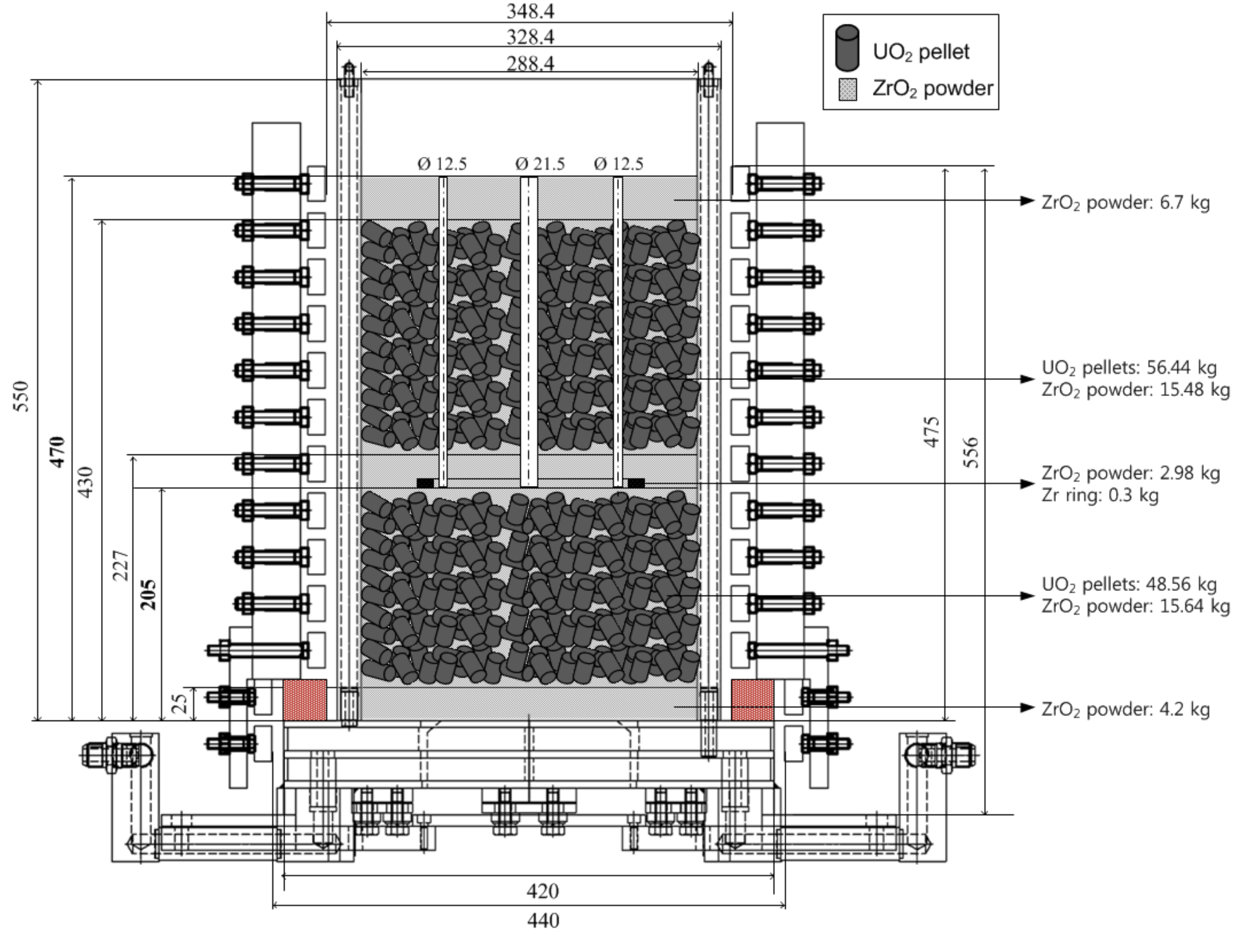




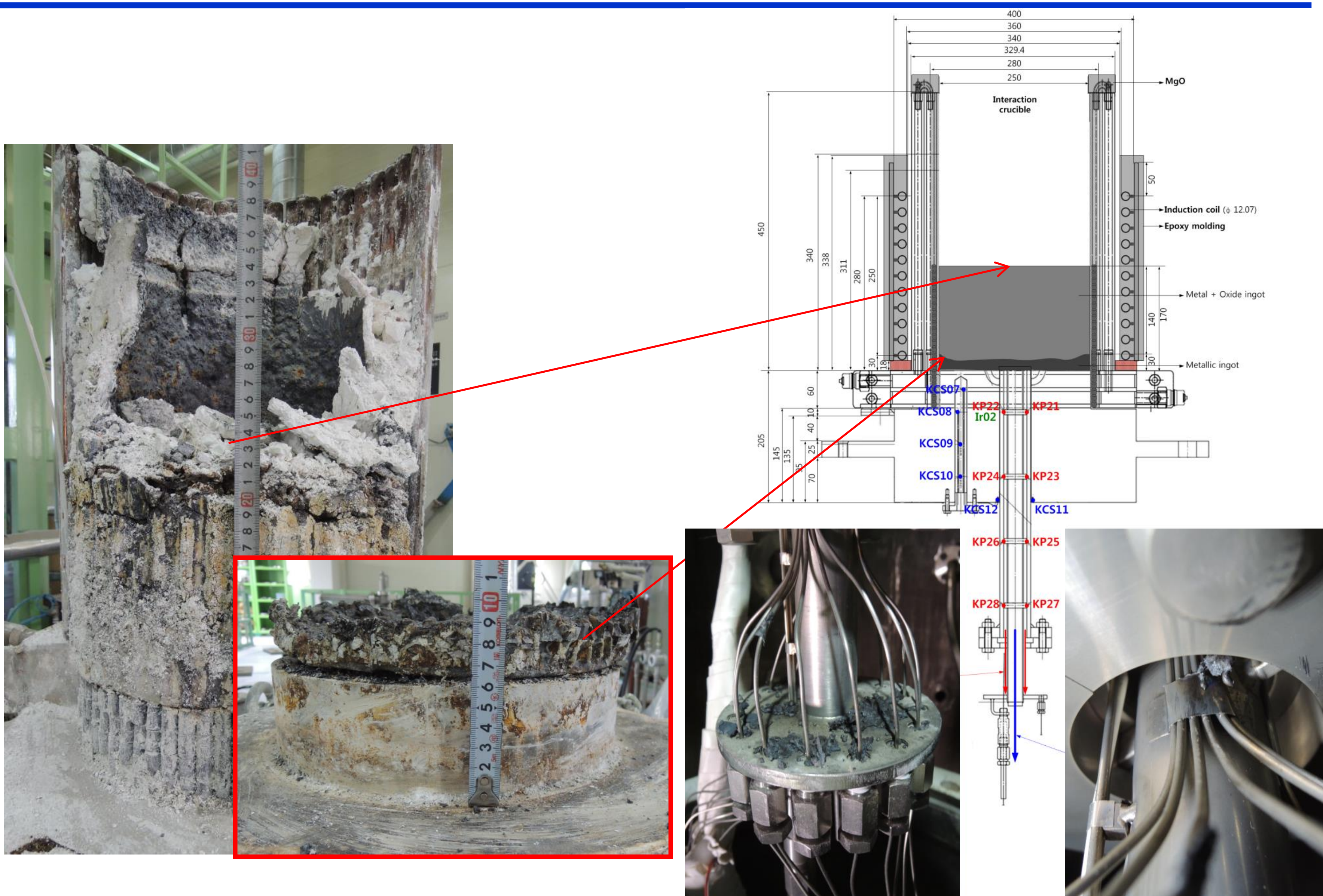
Task 4: (4-3) Experimental activity on lower head penetration failure – KAERI experiment

Material Charging in a Melt Crucible

- Melt composition (Oxidic corium)
  - UO_2 : ZrO_2 : Zr = 69.86 : 29.94 : 0.20 (wt%)
- Total charging mass
  - 150.3 kg
    - UO_2 : 105 kg
    - ZrO_2: 45 kg
    - Zr ring: 0.3 kg

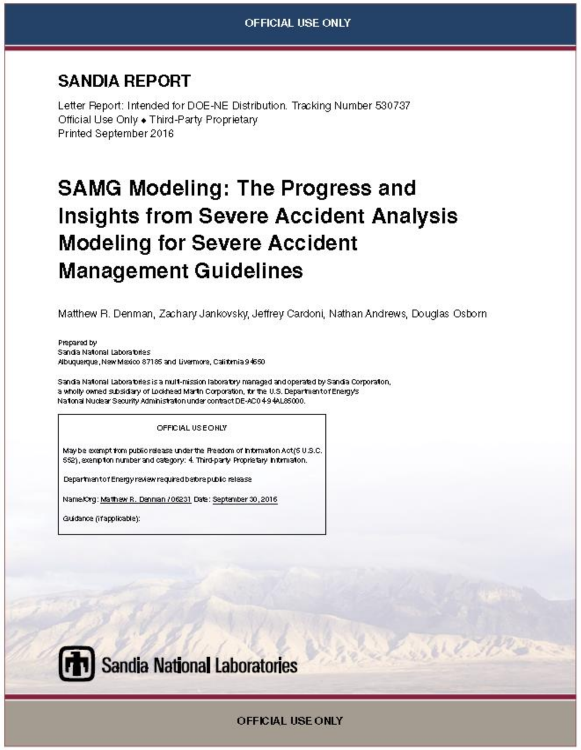


Task 4: (4-3) Experimental activity on lower head penetration failure – SRM/IRM experiment



Reactor Safety Efforts Supported by Industry (Japanese and US): Analysis to Validate SAMG Actions

- **Objective** : Use SA analysis accident signatures to test symptom-based SAMGs and confirm that they can address a wide-range of accident signatures, given model differences and uncertainties, and still achieve a safe stable state.
- **Motivation**: Mitigating Strategies Rule - requiring industry to train on SAMG procedures.
- **Benefits to Industry**:
  - Potentially reduce number of pathways to achieve long-term cooling
  - Potentially reduce number of operator actions and decisions
  - Inform training of Operators and Technical Support Center (TSC)
  - Inform parallel SAWA/SAWM analyses carried out as part of RST efforts.
  - Industry Participation: EPRI, BWROG (Exelon)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Printed September 2016

SAMG Modeling: The Progress and Insights from Severe Accident Analysis Modeling for Severe Accident Management Guidelines

Matthew R. Denman, Zachary Jankovsky, Jeffrey Cardoni, Nathan Andrews, Douglas Osborn

Prepared by
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Albuquerque, New Mexico 87185 and Livermore, California 94550

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Department of Energy review required before public release

Name/Org: Matthew R. Denman / 06231 Date: September 30, 2016

Guidance (if applicable):

Sandia National Laboratories

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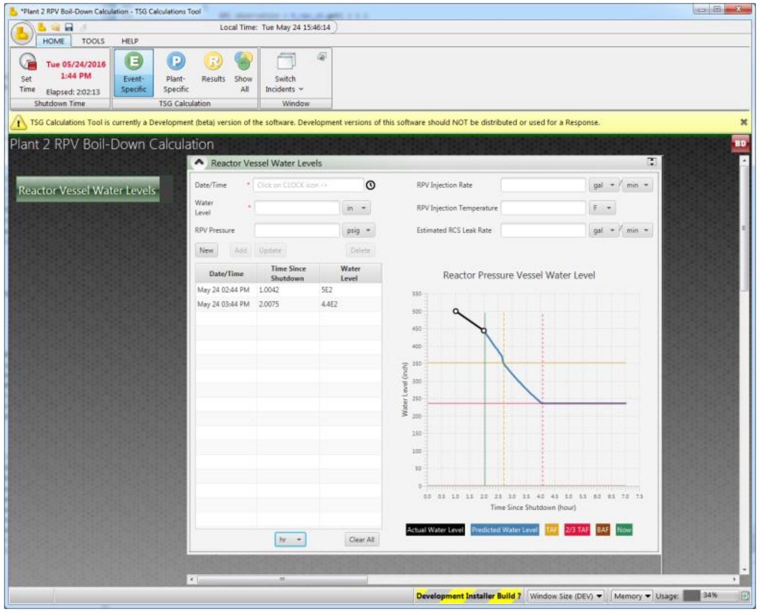
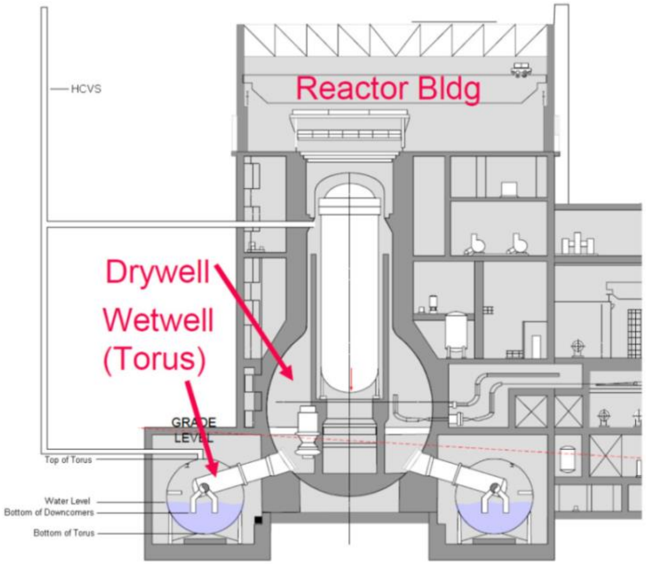
Reactor Safety Efforts Supported by Industry (Japanese and US): TSG Software Tool

- **Background**: BWR industry practice is to develop generic spreadsheet-based calculations in TSG for the TSC to assist operators with EOPs & SAMGs
- **Objective**: Develop user-friendly software tool for TSG calculations to support TSC
- **Motivation**: Generic calculations can be converted for plant specific calculations, but are neither intuitive nor easy to use.
- **Benefit to Industry**: Intuitive tool that can be easily transferred between groups (TSC, operators, EOP/SAMG BWROG support)
- **Industry Participation:** BWROG, PWROG



Reactor Safety Efforts Supported by Industry (Japanese and US): Ex-Vessel SAWM

Mk I BWR Containment



- **Objective:** Enhance, validate, document, and disseminate core melt spreading (MELTSPREAD) and debris coolability (CORQUENCH) models to support industry in SAMG planning.
- **Motivation:** Containment Protection and Risk Reduction
  - Industry is pursuing SAWM as an alternative to installing filters on containment vents.
- **Benefit to Industry:** Providing a validated set of codes that can be used to evaluate optimal containment flooding rate and injection pathway that achieves SAWM objective of maintaining core debris cooling while preserving wetwell vent path.
- **Participation:** EPRI (EdF), BWROG, NRC, IRSN, CEA.

Summary of Activities – 1

- **Significant progress in Fukushima analyses made by SNL and IAE**
- **Task 1-1 MELCOR uncertainty analysis on Fukushima Daiichi NPP**
  - Modeling uncertainties and code model comparisons
  - Supporting improvement of state of knowledge
  - Supporting the informing of decommissioning activities
- **Task 1-2 Daini Analysis of loss of heat sink – deferred to future years**
- **Task 1-3 RCIC Modeling and Experiments**
  - Cooperative evaluation of extended RCIC performance outside of original operating range
  - Development of testing approach
  - Validation of extended operating performance in beyond design basis events
- **Task 1-4 MELCOR-SAMPSON crosswalk**
  - Comparisons of MELCOR and SAMPSON predictions of Fukushima Daiichi 1F accident progression
- **Task 2 SAMPSON Model Enhancements**
  - RCIC mechanistic modeling implemented in SAMPSON
  - CFD validation of fundamental experiments has reached a mature stage
  - CFD application to 1/8 of the S/C is underway

Summary of Activities – 2

- **Task 3 Fundamental effects of sea water and lower plenum debris behavior**
  - Research getting underway to benefit severe accident code modeling
- **Task 4-1 Hydrogen CFD analyses – deferred to future years**
- **Task 4-2 Experimental studies of Suppression Pool condensation effects**
  - Important for realistic modeling of suppression pool response and prediction of containment pressure
  - Test of RCIC spargers for Unit 2 and Unit 3 under various conditions
- **Task 4-3 Experimental studies of lower head penetration failure**
  - Fundamental tests performed
  - Prototypical experiments to be performed before the end of the Japanese 2015 FY (i.e. April 2016)

Overview of Ongoing Activities

- **Fukushima Accident Analysis**
  - Collaboration with IAE analyzing impact of weather data on deposited radionuclides
  - Comparing WSPEEDI to MACCS results for release estimate
  - Calculation of dose in the Fukushima Daiichi containment during the accident
- **MELCOR-SAMPSON xWalk**
- **Reactor Safety Efforts Supported by Industry (Japanese and US)**
  - Technical Support Guidance Software
  - In-Vessel Analysis to Support SAMG Actions
  - Ex-Vessel SAW Analysis
- **Discussions and Informational Meetings with Japanese Collaborators**
  - SNL (and NRC)
  - NRA-J, METI, TEPCO, IAE and JAEA
- **RCIC Testing and Analysis**
  - Investigate realistic operational limits of Reactor Core Isolation Cooling (RCIC) System in BWRs
  - Z-1 and GS-series Terry turbopump tests
  - Schedule and milestone updates

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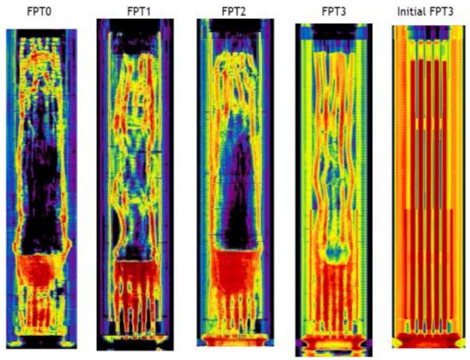
Source: Tokyo Electric Power Company

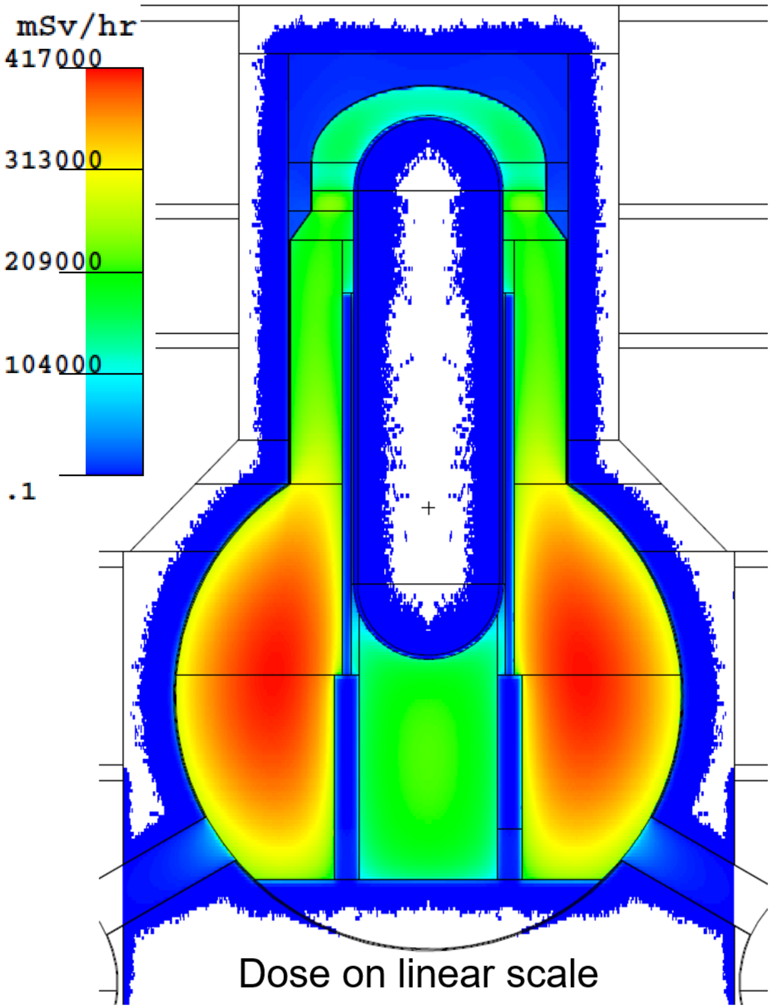


QUESTIONS & COMMENTS?

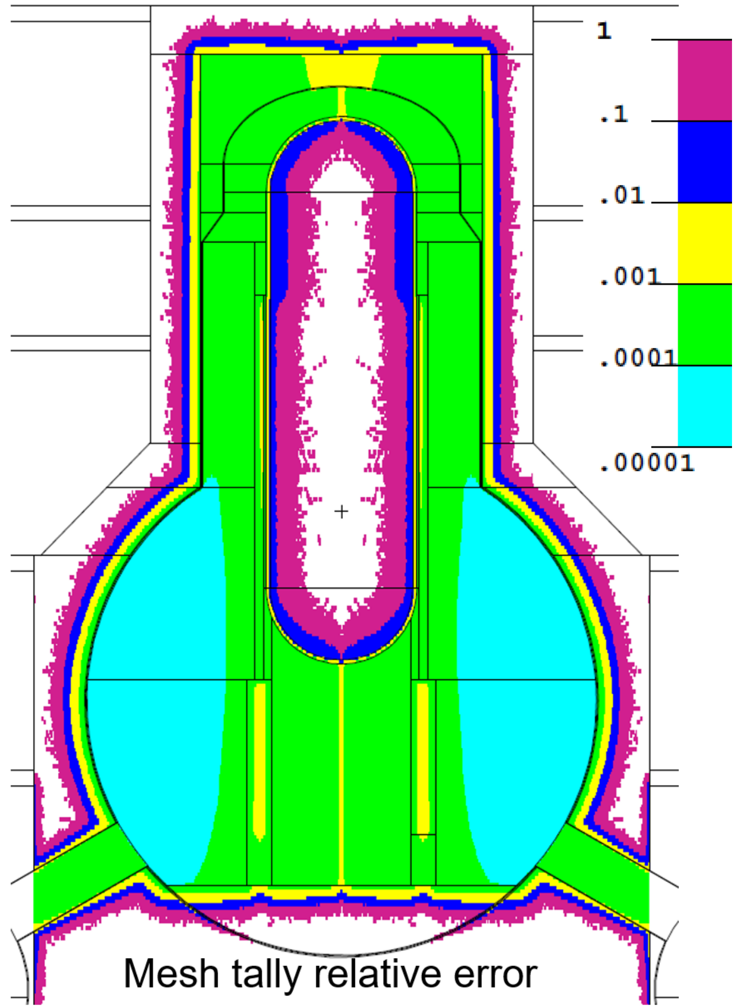
Objectives of Accident Analysis

- **Reasonably replicate ground deposition pattern based on MELCOR source terms**
  - Major focus on Cs-137
  - Primary focus on deposition toward the northwest
- **Provide guidance in release timing and magnitude for severe accident code analysts**
- **MCNP an WSPEEDI Analysis of Dose**
  - Primary containment during the accident
- **Inform Japanese Partners**
  - Locations of damaged fuel
  - Fission product species present in containment, reactor buildings and in the countryside
  - Reduce cost and uncertainty in decommissioning

Primary Containment Dose Estimates – 1F3



Dose on linear scale



Mesh tally relative error

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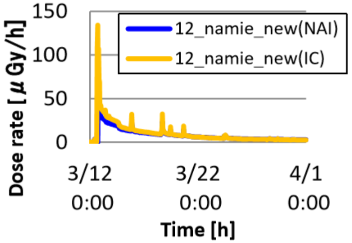
IAE – Connection between measurements and plant releases

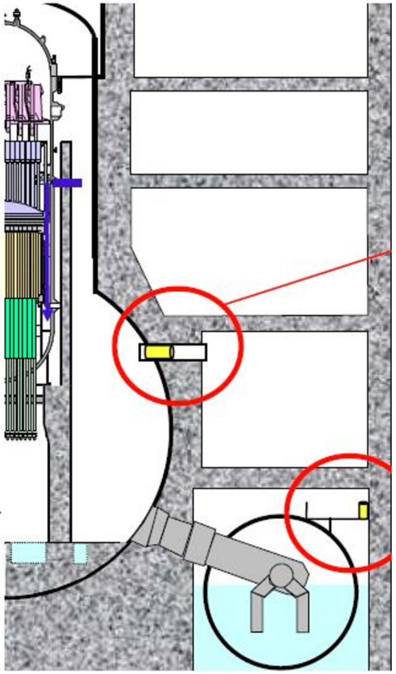
Measurements and data



WSPEEDI and SEA-GEARN-FDM inverse calculation

G. Katata, M. Chino, T. Kobayashi, H. Terada, M. Ota, H. Nagai, M. Kajino, R. Draxler, M. C. Hort, A. Malo, T. Torii, and Y. Sanada, "Detailed source term estimation of the atmospheric release for the Fukushima Daiichi Nuclear Power Station accident by coupling simulations of an atmospheric dispersion model with an improved deposition scheme and oceanic dispersion model", Atmos. Chem. Phys., 15, 1029–1070, 2015

Plant releases



Output from WSPEEDI calculation represent a perfect output to compare results of SA codes and add information for accident interpretation.

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Collaborative Meetings

- **NRA-J**
  - Decommissioning and decontamination
  - Water infiltration and treatment
  - Combustible gas issues
- **TEPCO/IRID**
  - Decommissioning and decontamination
  - Modeling dose around Fukushima Daiichi site
- **IAE**
  - Terry Turbine (RCIC)
  - MELCOR-SAMPSON xWalk
  - Fukushima Modeling
- **METI**
  - TMI Knowledge Transfer (with NRC)
- **JAEA**
  - Fukushima release characteristics
  - Future international Fukushima analysis activities
  - Ongoing experimental activities

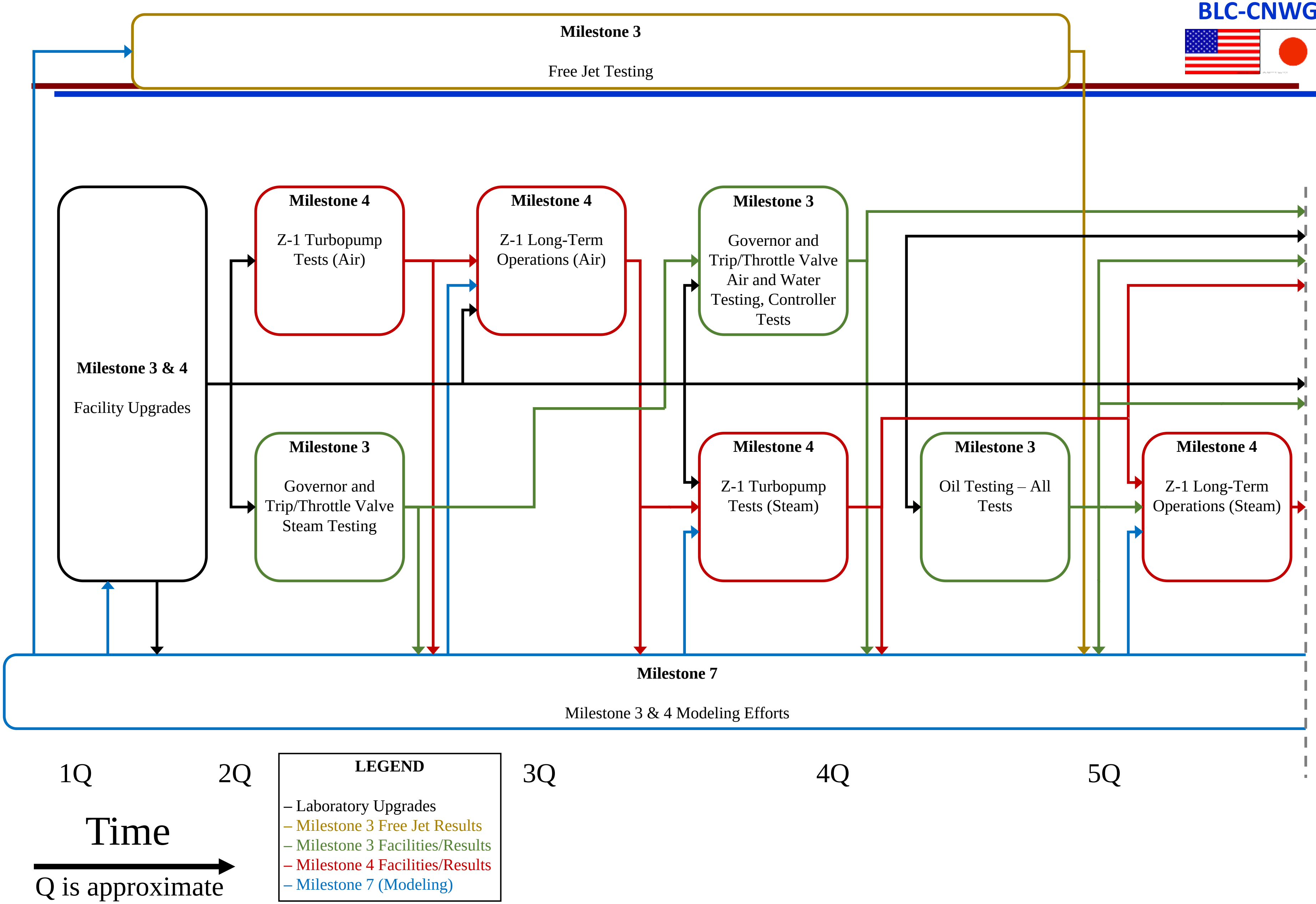
Task 1-3 RCIC EXOB Next Steps

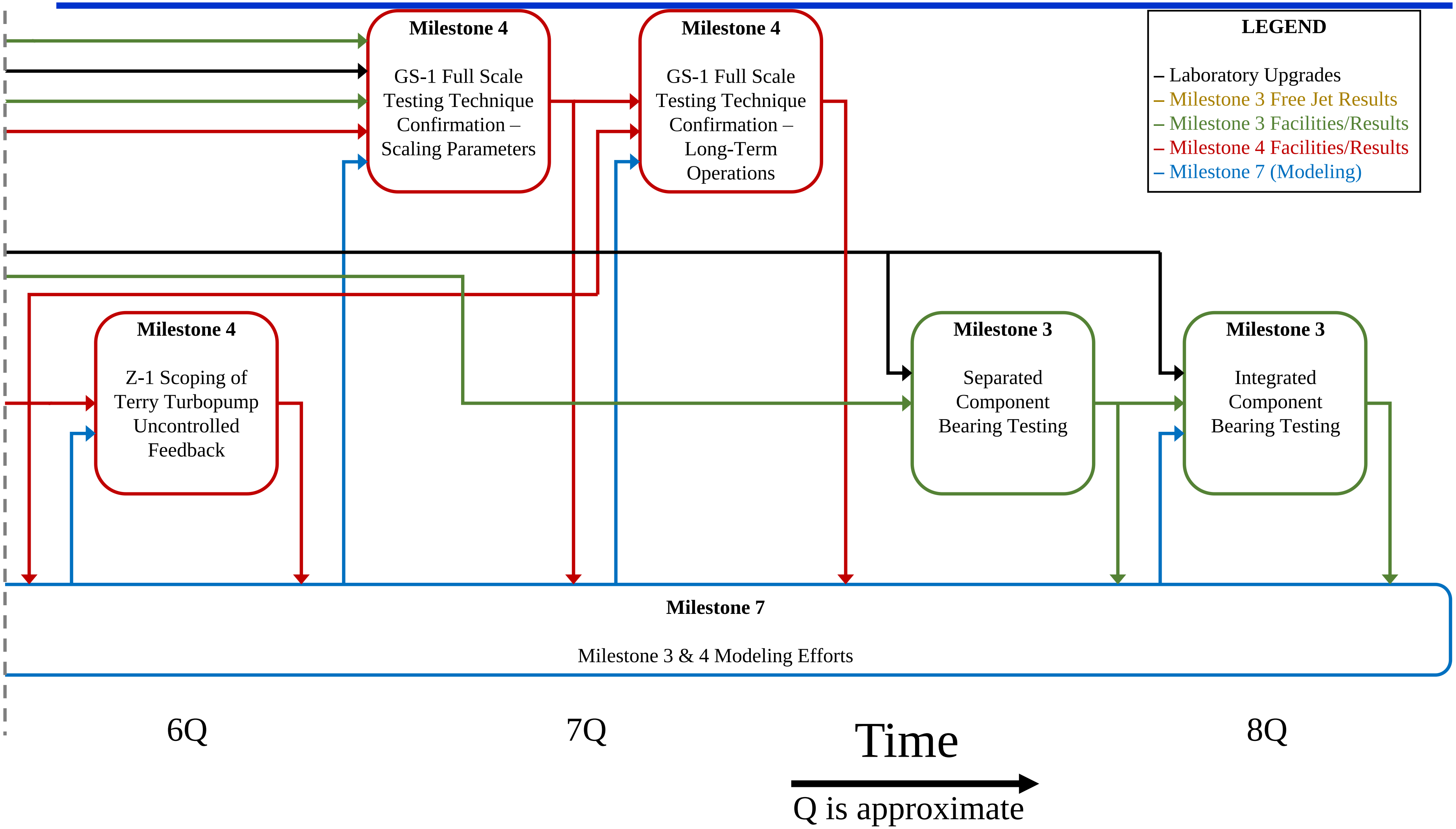
Project Plan Timeline

01-08-2016	Draft
02-01-2016	Core Team Review
03-01-2016	Committee Approval

Project Plan Logistics

February	Committee meeting and visit to Texas A&M (Core Team to visit Texas A&M and Sandia on Proto-type and Small Scale testing)
February	Review Test Facilities (NTS, Dresser-Rand, AREVA Karlstein, DOE) Note: Meet at a Video Conference location
February	Select Project Management (PM Options PIM, EPRI, OG (GEH), Independent)
March	Update BWROG EOC
April	Update NSIAC
May	Update BWROG Exec & EPRI
July	Update TTUG

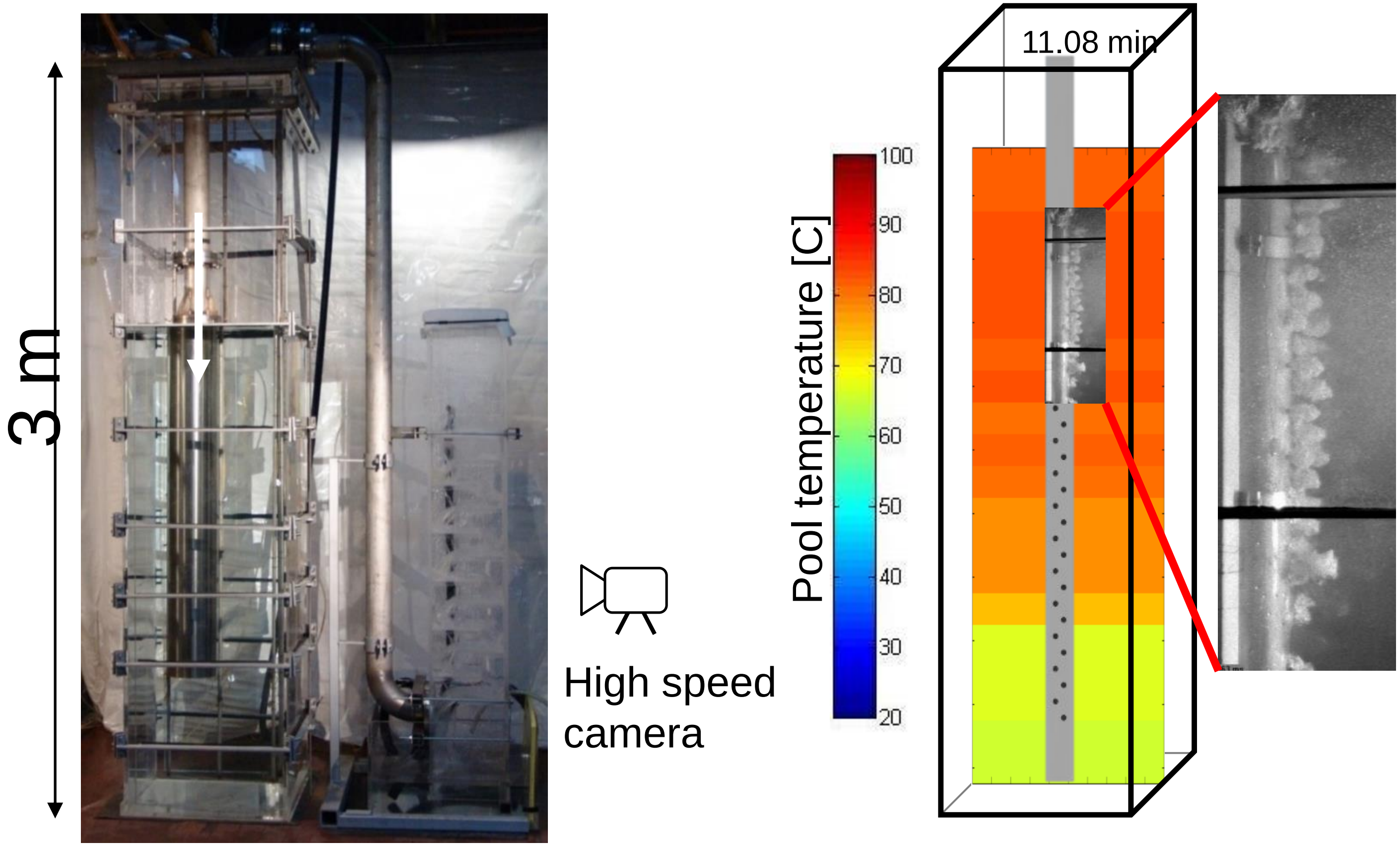

Milestone 3
Free Jet Testing
Milestone 3 & 4
Facility Upgrades
Milestone 4
Z-1 Turbopump Tests (Air)
Milestone 4
Z-1 Long-Term Operations (Air)
Milestone 3
Governor and Trip/Throttle Valve Air and Water Testing, Controller Tests
Milestone 3
Governor and Trip/Throttle Valve Steam Testing
Milestone 4
Z-1 Turbopump Tests (Steam)
Milestone 3
Oil Testing – All Tests
Milestone 4
Z-1 Long-Term Operations (Steam)
Milestone 7
Milestone 3 & 4 Modeling Efforts
1Q
2Q
3Q
4Q
5Q
LEGEND
– Laboratory Upgrades
– Milestone 3 Free Jet Results
– Milestone 3 Facilities/Results
– Milestone 4 Facilities/Results
– Milestone 7 (Modeling)
Time
Q is approximate


Milestone 4
GS-1 Full Scale Testing Technique Confirmation – Scaling Parameters
Milestone 4
GS-1 Full Scale Testing Technique Confirmation – Long-Term Operations
LEGEND
– Laboratory Upgrades
– Milestone 3 Free Jet Results
– Milestone 3 Facilities/Results
– Milestone 4 Facilities/Results
– Milestone 7 (Modeling)
Milestone 4
Z-1 Scoping of Terry Turbopump Uncontrolled Feedback
Milestone 3
Separated Component Bearing Testing
Milestone 3
Integrated Component Bearing Testing
Milestone 7
Milestone 3 & 4 Modeling Efforts
6Q
7Q
Time
8Q
Q is approximate

Task 4-2 Experimental activity – Steam Condensation in S/C

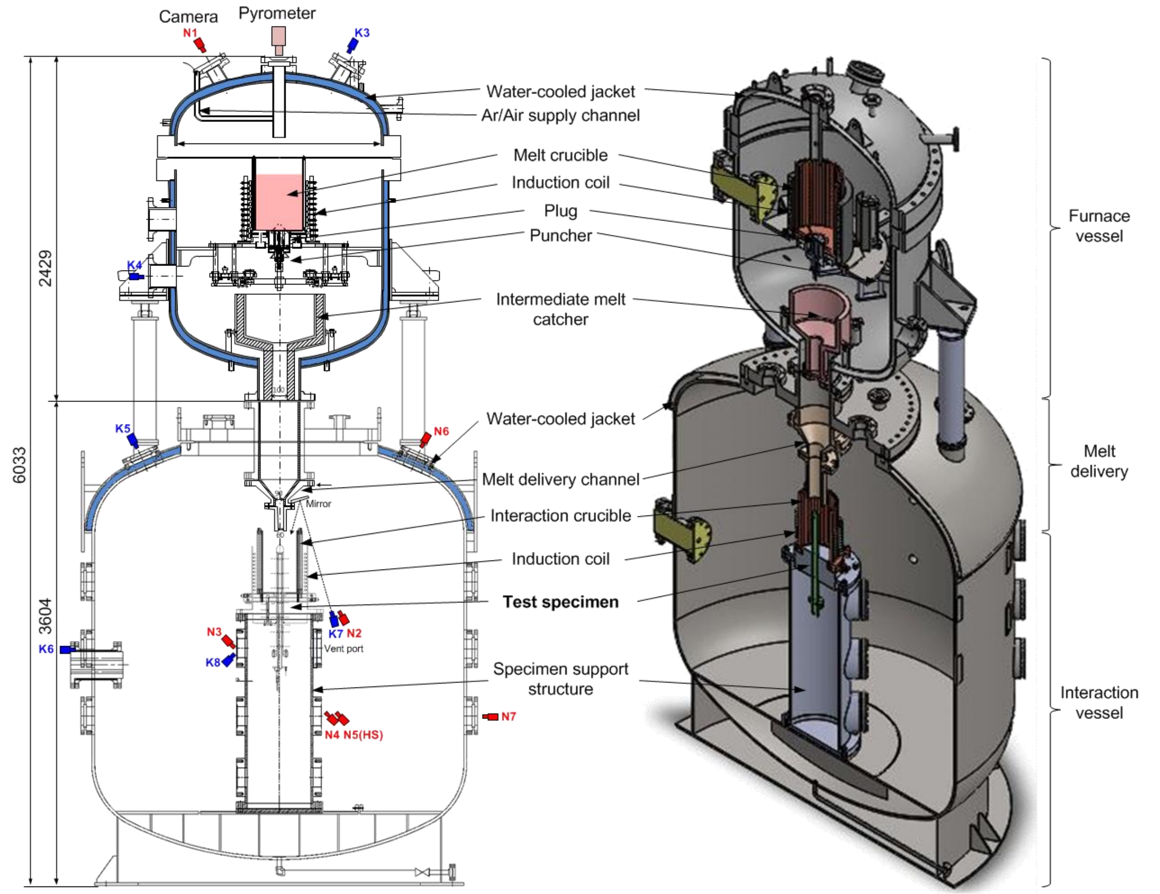
- Real scale sparger geometry and height was tested in the SWAM facility in the SIET laboratories to confirm S/C temperature distribution during RCIC operation.

- The test results have been applied for validation of CFD analysis.



Task 4-3 IAE/KAERI Penetration Tube Experiment

- The melt behavior of In-Core Monitor Housing and Control Rod Housing Tubes, which penetrate the RPV bottom head, and corium ingression into the Tubes were investigated with KAERI VESTA facility.
- The test tube specimens were the full scale horizontal cross sections, but short length.
- The tubes investigated were the Source-Range Monitor (SRM) and CR housing.
  SRM: Double tube with open bottom end
  CR housing: Quadruple tube with closed bottom end
- The corium used was a mixture of 70% UO_2 + 30% ZrO_2.



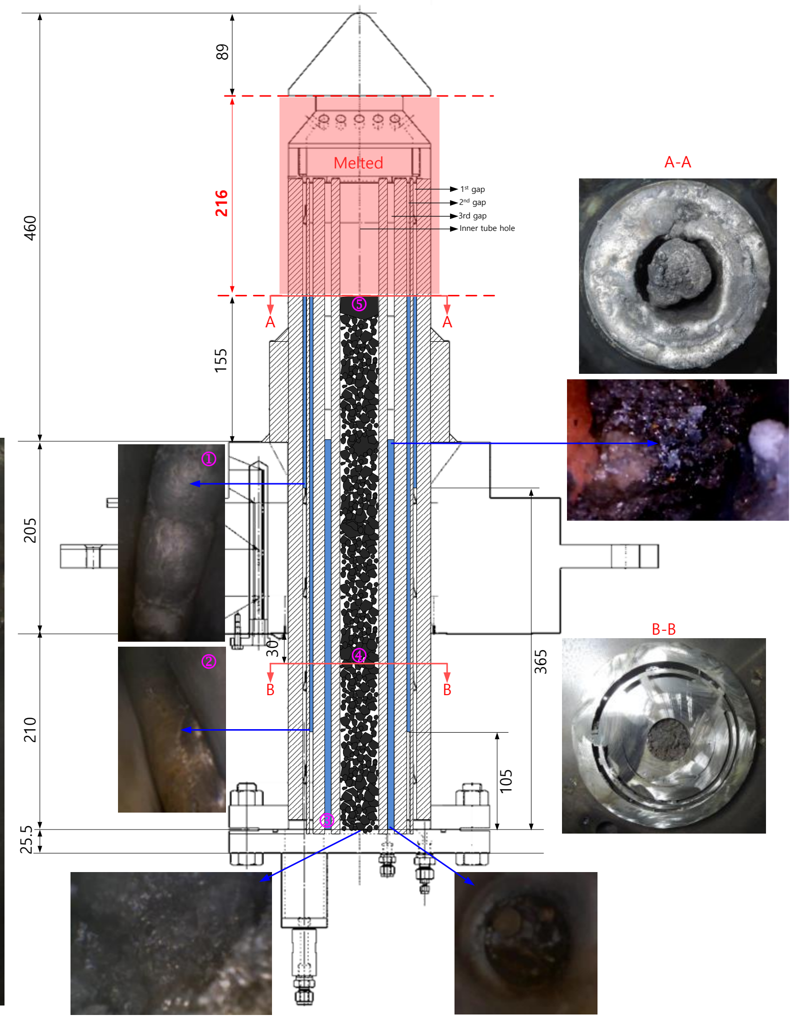
Task 4-3 IAE/KAERI Experiment – Post Test Inspection (Ingression of Corium into the tubes)

- SRM double tube with open bottom end:
  - ✓ Very small amount of corium had fallen out from the bottom end, but
  - ✓ The annular and inner path were blocked by SS crust.
- CR housing quadruple tube with closed bottom end:
  - ✓ The most inside tube was filled by the oxide crust, and
  - ✓ The annulus flow path were partially blocked by SS crust.

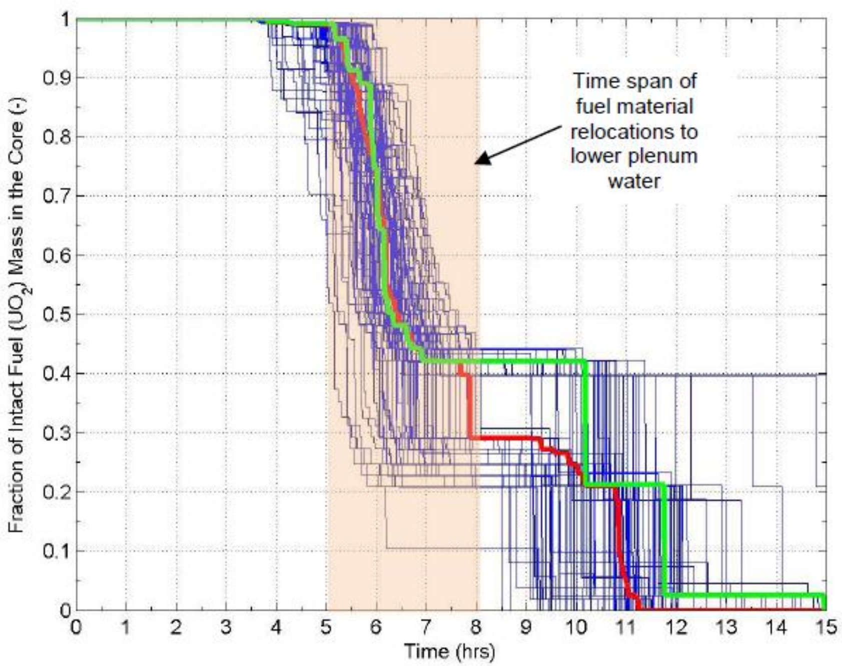
Application of the test results to an actual plant is now being considered.

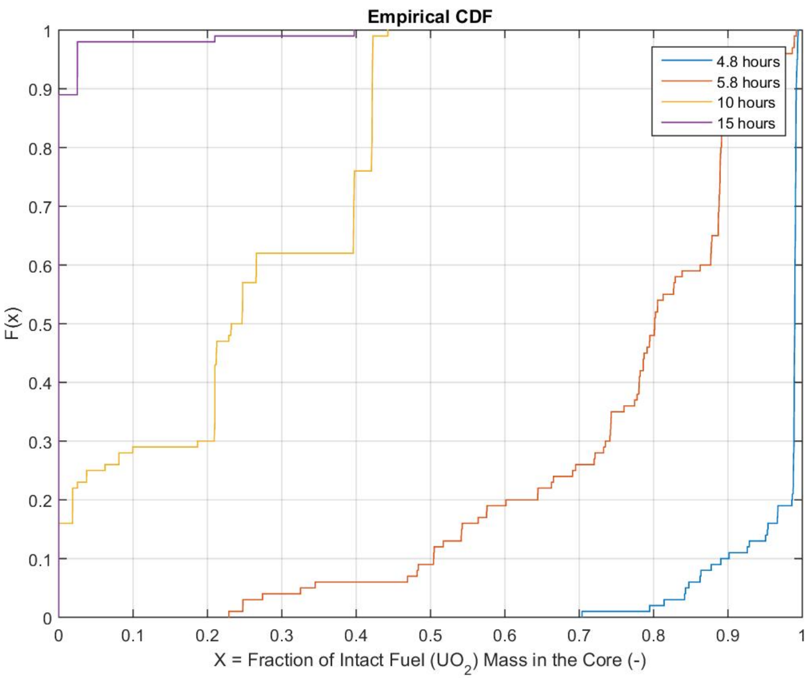
Reference: M. Naitoh, et al., ERMSAR2017 Paper No. 518, Warsaw, Poland, May 2017

B-B cutting surface



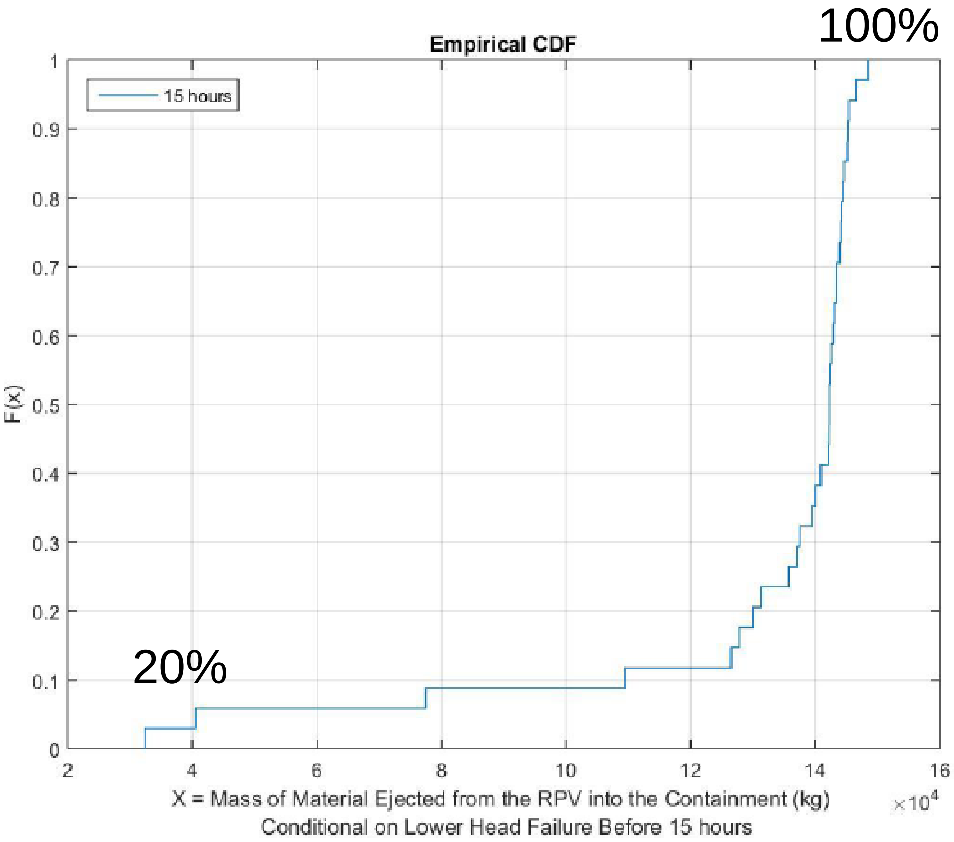
Task 1-1 MELCOR Uncertainty Analysis of Fukushima Daiichi Unit 1 Damage State (2)





- Fraction of fuel remaining in core region
- 100 sampled realizations (red: median; green: mean
- Cumulative distribution representation for 5 select times
- 15 hour cdf suggests some residual fuel in core region is possible – peripheral elements

Task 1-1 MELCOR Uncertainty Analysis of Fukushima Daiichi Unit 1 Damage State (3)



- Cumulative distribution at 15 hours for core mass relocated to reactor cavity
- Spectrum of results suggests that:
  - A few percent of realizations showed only 20% fuel debris relocated to cavity
  - 90% of realizations ejected between 85% and 100% to cavity
- Results can inform decommissioning planning for fuel removal