CNWG TE-4:
Severe Accident Code Assessment

SAND2019-3865PE

Co-Chairs:

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

Presented by: Nathan Andrews (SNL) & Marco Pellegrini (IAE)

Sixth Meeting of the

Civil Nuclear Energy Research and Development Working Group

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Argonne National Laboratory, United States

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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 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 (completed)

~~(1-2) MELCOR analysis of Fukushima Daini~~

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

(1-4) MELCOR-SAMPSON crosswalk (completed)

(1-5) Coupled SA code & MACCS Assessment of Fukushima-Daiichi Accidents

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

(2-1) MCCI experimentation and analysis - ROSAU project

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

Task 4: 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) (completed)

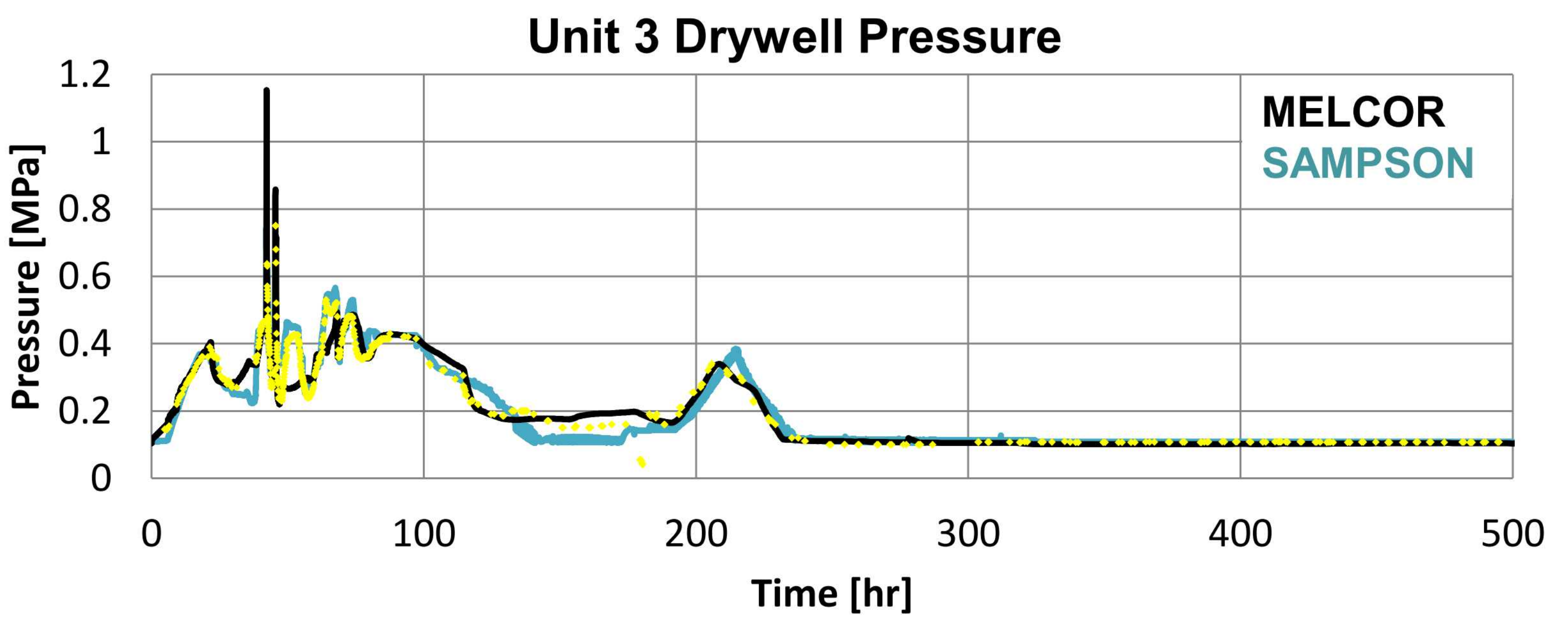
(4-3) Experimental activity on lower head penetration failure

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

- **IAE served as operating agent for the OECD-NEA BSAF project focused on code analysis of Fukushima accidents and reconstruction of the principal accident signatures**
  - The BSAF Phase II addressed three weeks calculations (around 500 h) including thermal-hydraulics, core damage progression, MCCI and fission products transport and release
- **Code assessments for Phase II of BSAF project were completed in 2018**
  - Final report of the benchmark was completed in 2019 summary report will be delivered to OECD/NEA in April 2019
  - Extensive discussion FP treatment, decommissioning implications and lessons learned from the project are included in the final report
  - IAE served as the primary author of the report
- **IAE and SNL cooperatively worked for post-processing the participant data:**
  - Conversion of FP into in containment dose rate (IAE)
  - FP dispersion in environment (SNL)

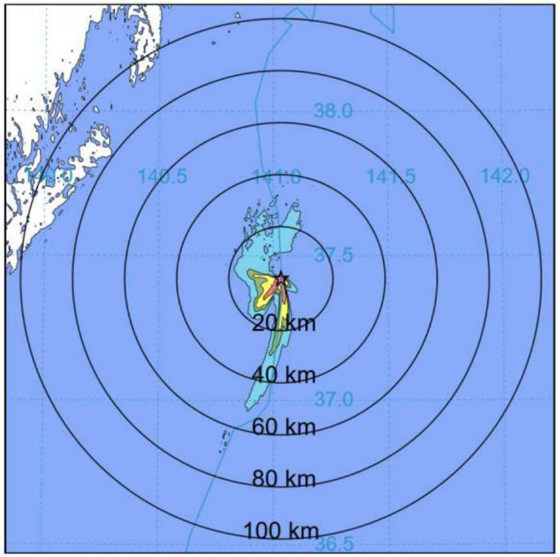
Task 1 and 2: Overview of BSAF Phase II

- **Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (BSAF) Project**
  - Three separate three week long MELCOR and SAMPSON simulations
  - Completed final report
- **Single, combined MACCS simulation of the three MELCOR and MACCS simulations**
  - High-level benchmark of release to the environment from SA codes and subsequent deposition following release



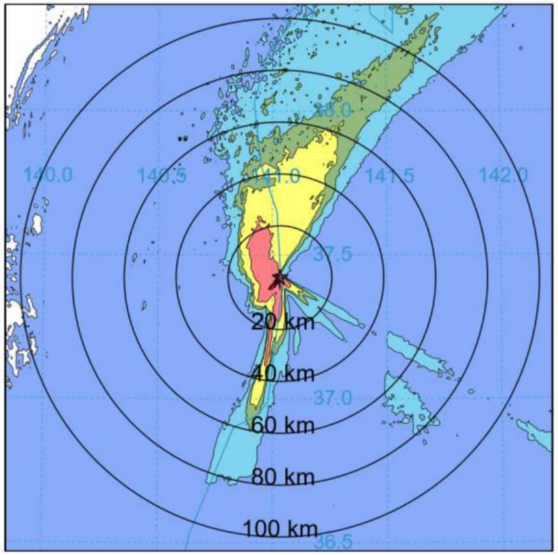
Task 1-5: Coupled SA Code & MACCS Assessment of Fukushima-Daiichi Accidents

1F1



- **Evaluate viability of source terms by reasonably replicating ground deposition patterns**
  - Focus on Cs-137
  - Focus on deposition toward the northwest and overall deposition pattern
- **Provide guidance in release timing and magnitude for source term analysts**
- **Benchmark models against real data**
  - HYSPLIT particle tracking model

1F2



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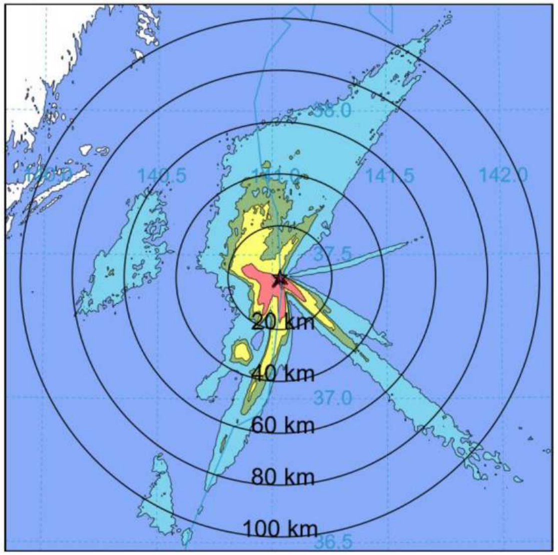

Task 1-5: Coupled SA Code & MACCS Assessment of Fukushima-Daiichi Accidents

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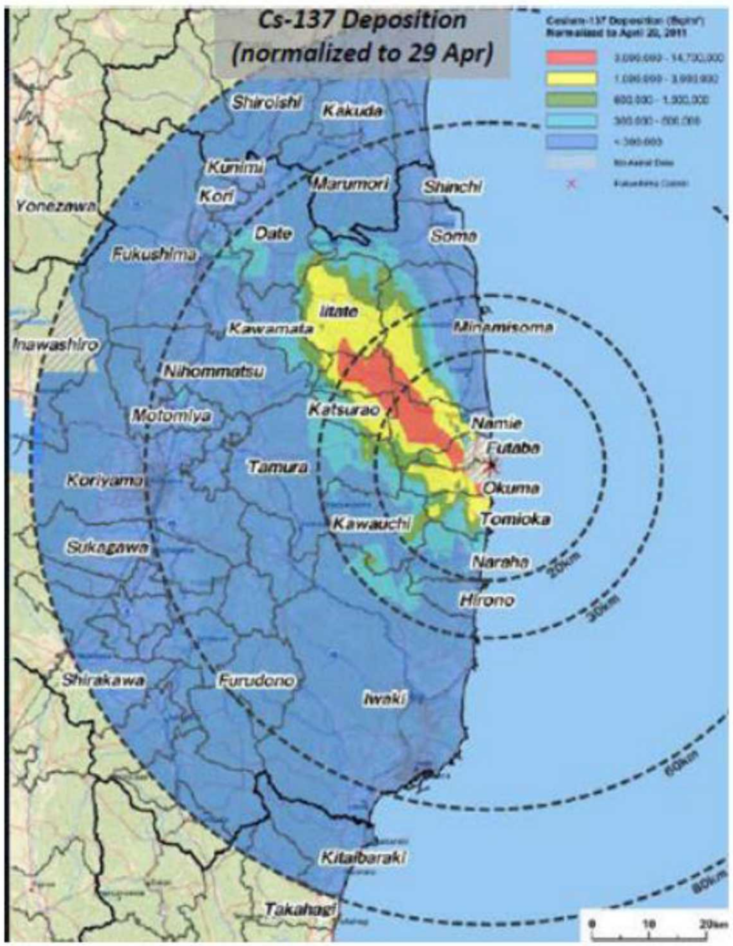
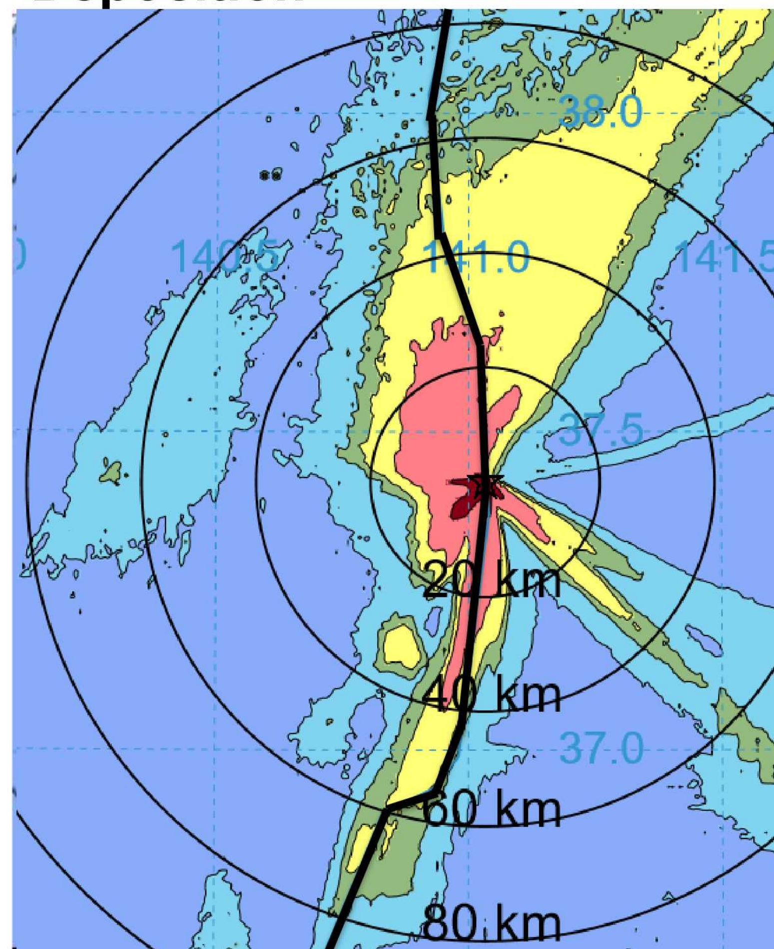
Observed Deposition



MELCOR/MACCS Predicted Deposition



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

<u>Reduce and Defer Costs</u>

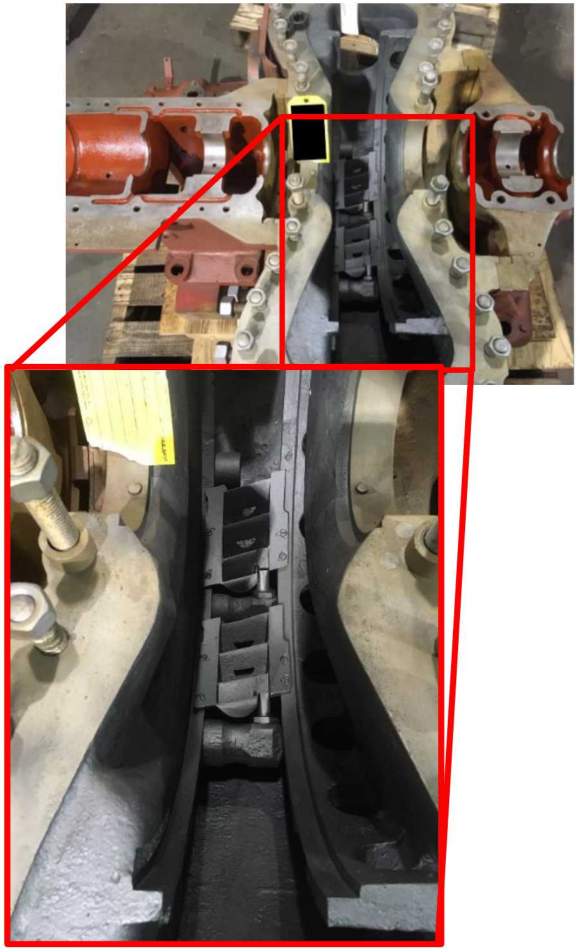
- Extends the intervals between preventive maintenance periods
- Provides improved transition to portable FLEX equipment
  - Deferring the use of ultimate FLEX measures using raw water at one BWR plant saves **~$450M**

<u>Reduce Risk of Operations</u>

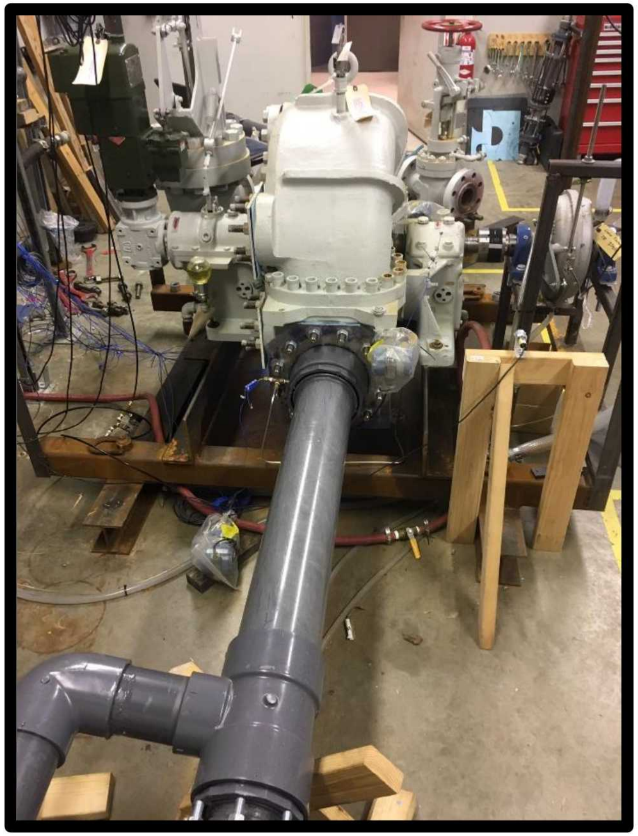
- Update emergency operating procedures (EOPs)
- Establish technical basis for operational changes that prevent progression to core damage and reduce core damage frequency

<u>Simplify Plant Operations</u>

- Add flexibility to respond to event conditions identified in the Fukushima accidents
- Increased time available for implementation of FLEX



Task 1-3: Plans and Goals for 2019 (2)



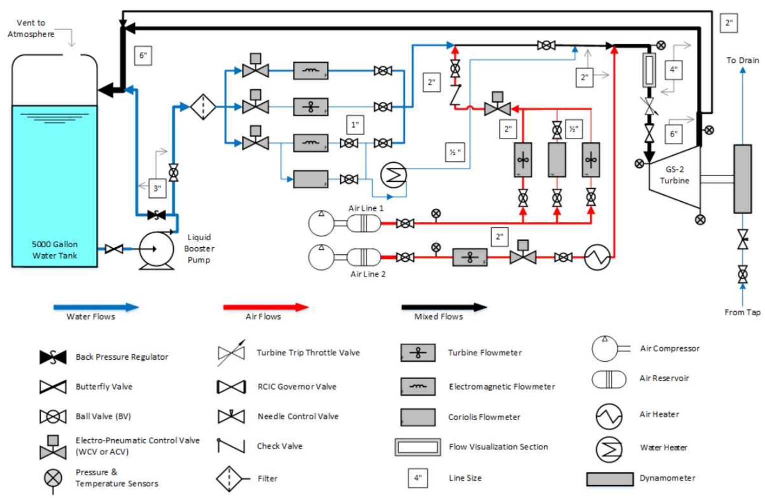
Complete Milestone 3 & 4 efforts:

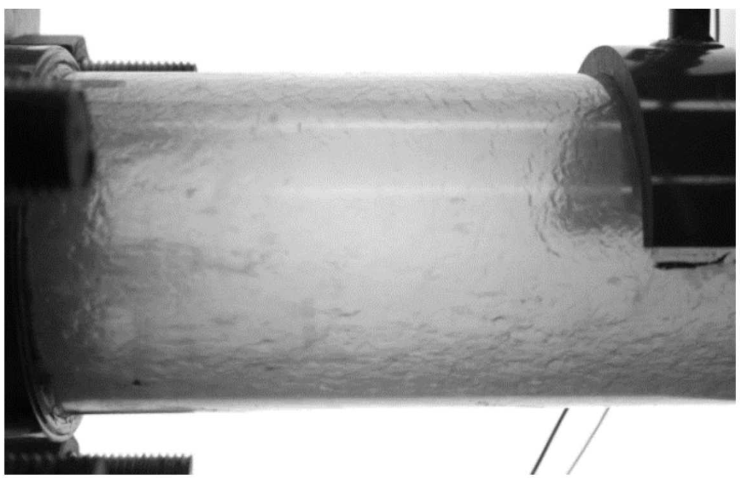
- Modeling
  - CFD critical flow analyses
  - System level analyses
- Experimentation at TAMU
  - ZS-1 Terry turbopump
  - GS-2 Terry turbopump (TD-AFW skid)
  - Steam nozzle free jet visualization
  - Trip/Throttle Valves (GS-1 & GS-2)
  - Governor Valve

Finalize plans for Milestone 5 testing:

- Develop path forward for project completion

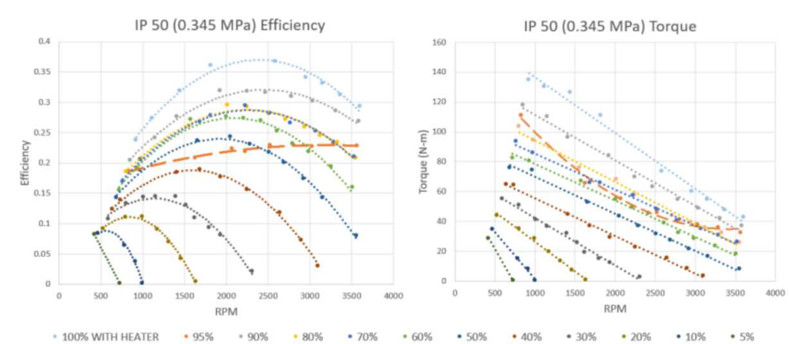
Task 1-3: Accomplishments (Experimental) (3)





b) 60% Air Mass Fraction

- Terry GS-2 testing
  - Test Ranges
    - 20 to 70 psia (0.14 to 0.517 MPa)
    - 5% to 100% air mass fraction
    - Below 4000 RPM
  - Dynamometer and needle control valve to regulate turbine speed
  - Air heater implemented to warm exit temperature above freezing due to expansion cooling
- Two-Phase flow Visualization
  - Located at TTV inlet
  - High speed camera
  - 400 fps
  - Backlit board

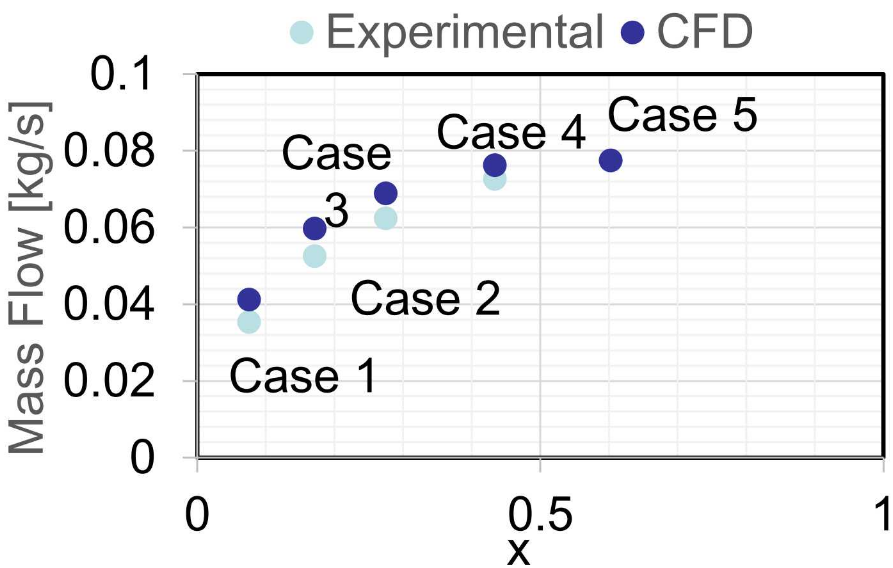


Task 1-3: Accomplishments (Experimental) (4)

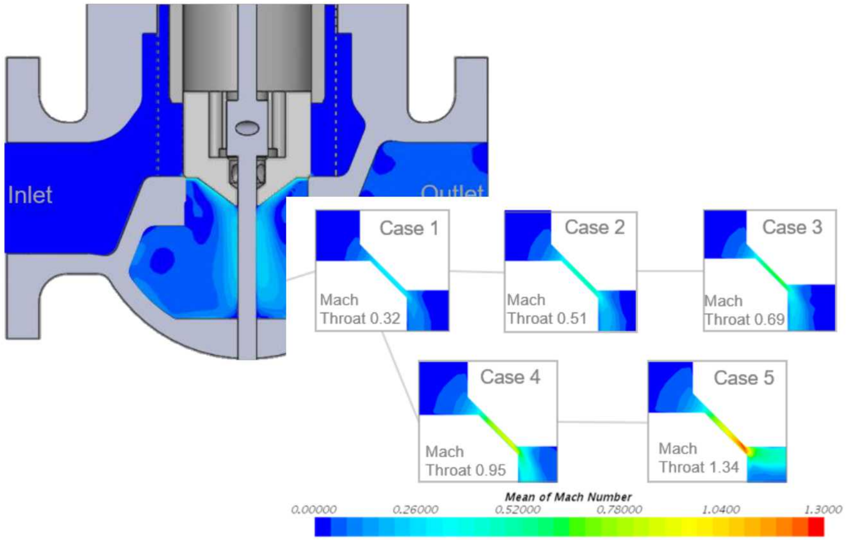
- Conclusions
  - Turbine follows established affinity laws
    - Quadratic relationship between RPM and efficiency
    - Linear relationship between RPM and torque
  - When water is present at high qualities…
    - Suppressed output torque and efficiency
    - Temperature drop smaller than expected
    - Not observed at lower qualities (<80%) because of heat capacity of water
    - May be due to presence of 3rd phase (ice) at low temperatures or non-equilibrium interactions
- Bearing Friction
  - Oil degradation
  - Representative temperatures
    - Prove survivability

Task 1-3: IAE CFD modeling (VALVE)

At IAE we have established a method for evaluation of the valve Cv based on CFD calculations in case of chocked flow conditions for single phase flow. Results show excellent agreement with the experimental activity performed at Texas A&M



Example of Trip/Throttle valve simulation in STAR-CCM+



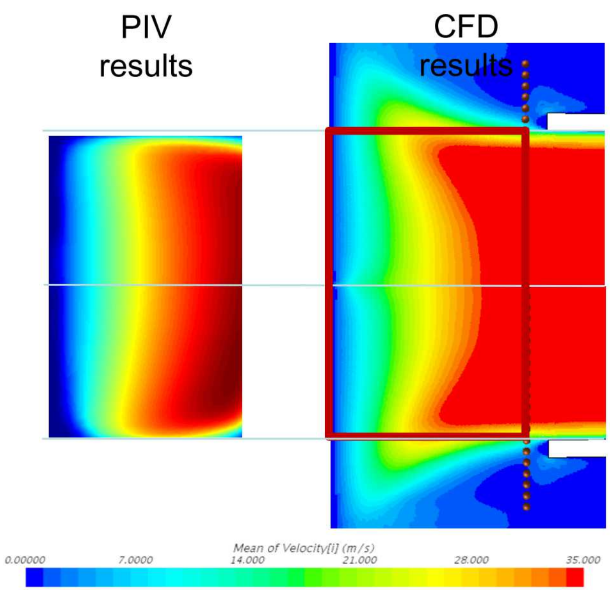
- Current activity is the extension to T/T GS-1, GS-2 and governor type valve in single phase flow
- 2019 activity will deal with two phase flow validation against TAMU experiments

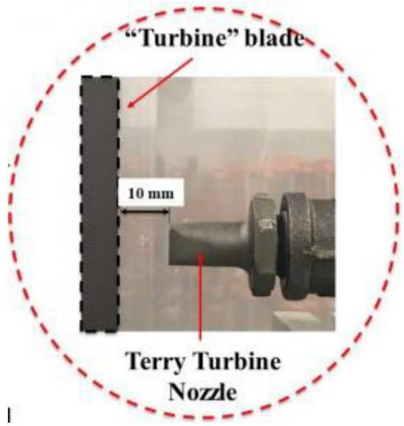
Task 1-3: IAE CFD modeling (Super-sonic JET)

Comparison between CFD and TAMU experiment

x-component velocity

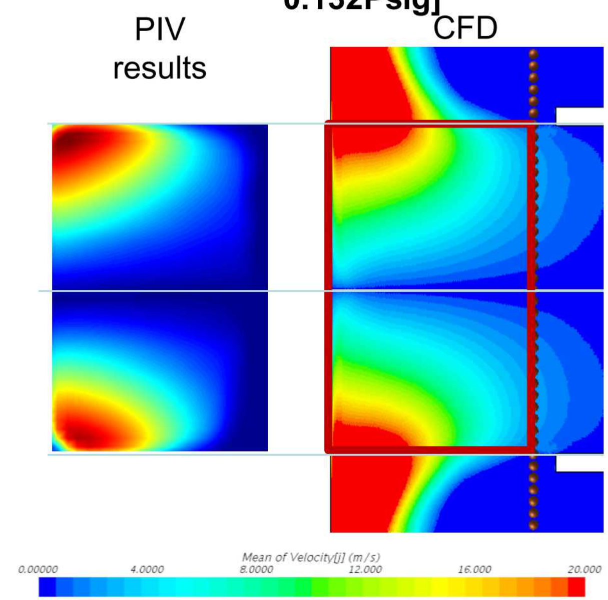
10mm – MFR = 0.0064 kg/s [P_{in_CFD} = 0.132Psig]





y-component velocity

10mm – MFR = 0.0064 kg/s [P_{in_CFD} = 0.132Psig]

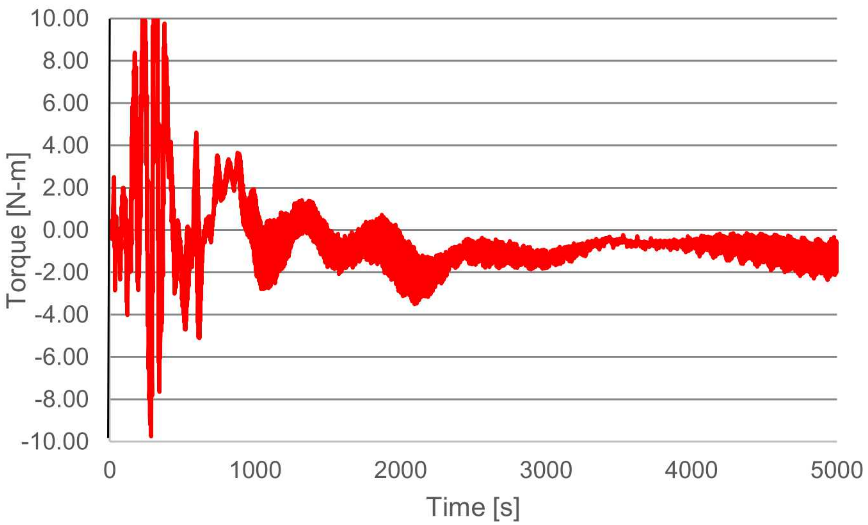


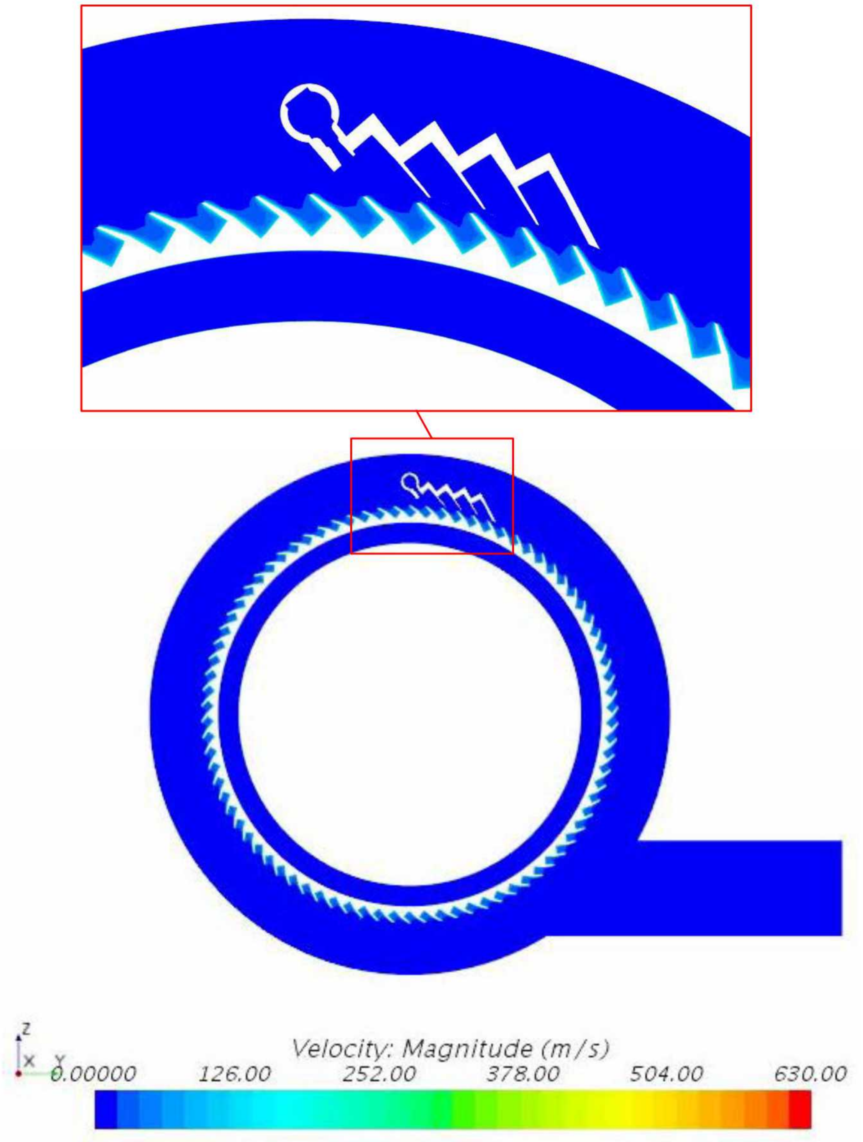
- ➢ There is an excellent agreement between CFD and TAMU experiment.
- ➢ 2019 activity will focus on application to two phase flow regimes

Task 1-3: IAE CFD modeling (Full 3D RCIC Turbine)

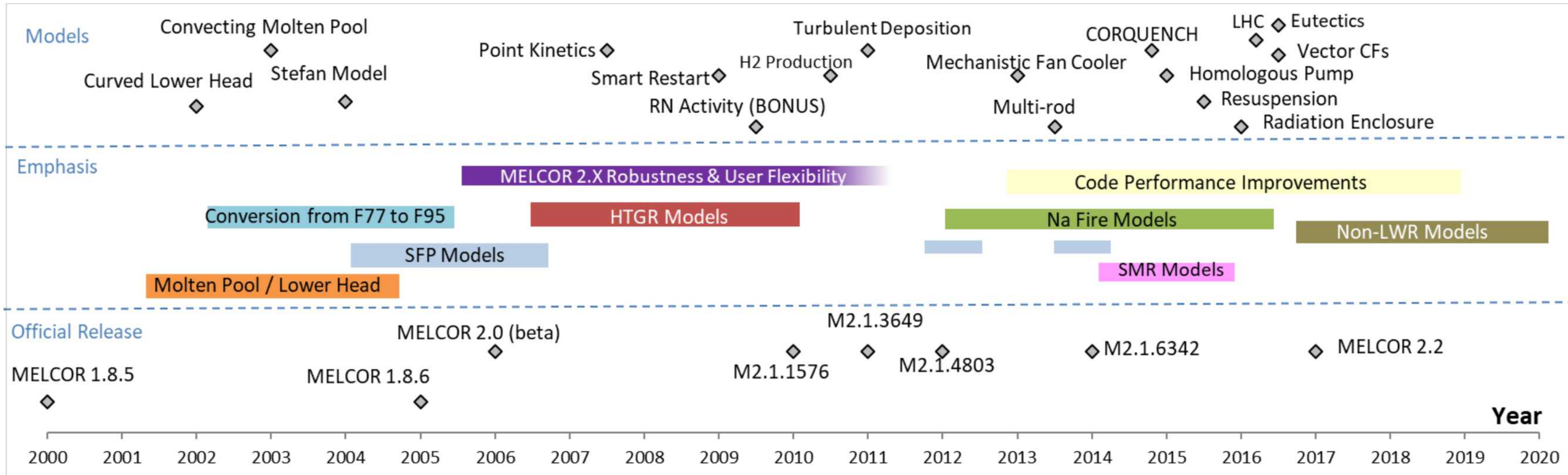
Based on the validation activity performed on super-sonic jets at IAE we have built a full 3D model of the GS-1 turbine for the evaluation of torque-rpm plots and isentropic efficiency in single and multi-phase flow

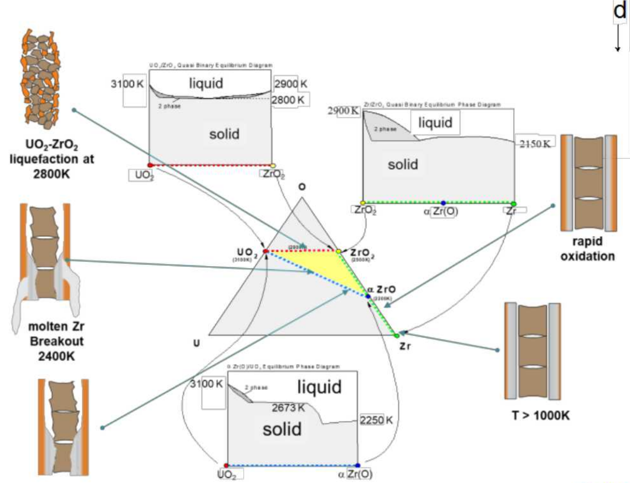
Computation of turbine torque

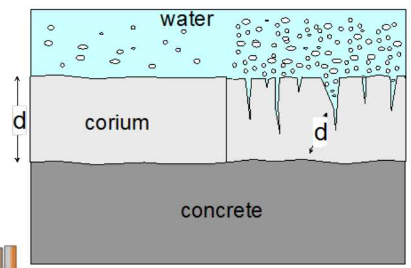


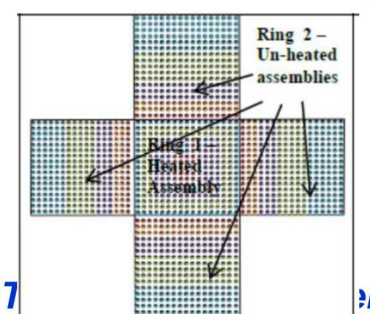


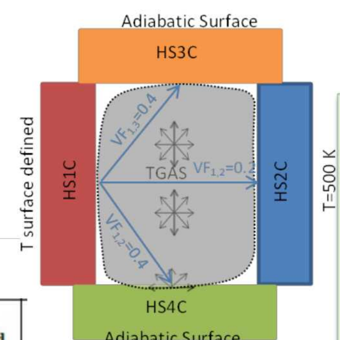
Task 2: MELCOR Development Activities

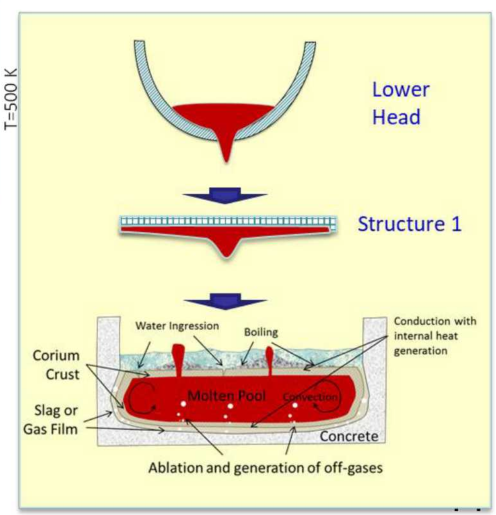




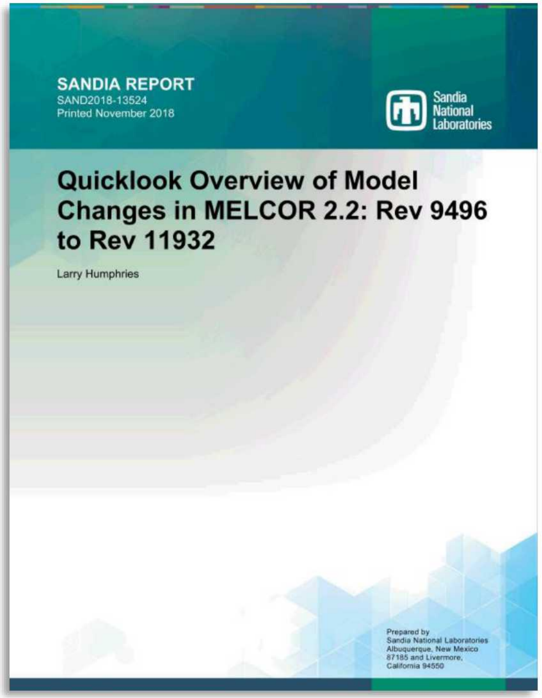








Task 2: MELCOR Development Activities



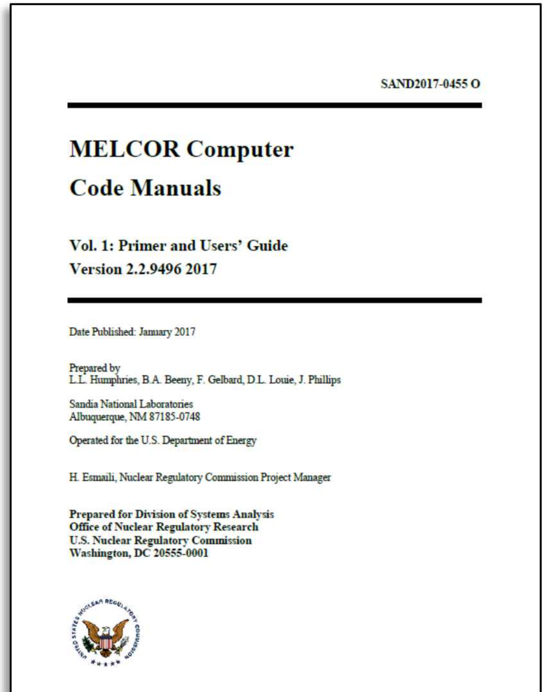
SANDIA REPORT
SAND2018-13524
Printed November 2018

Sandia National Laboratories

Quicklook Overview of Model Changes in MELCOR 2.2: Rev 9496 to Rev 11932

Larry Humphries

Prepared by
Sandia National Laboratories
Albuquerque, New Mexico
87185 and Livermore,
California 94550



SAND2017-0455 O

MELCOR Computer
Code Manuals

Vol. 1: Primer and Users' Guide
Version 2.2.9496 2017

Date Published: January 2017

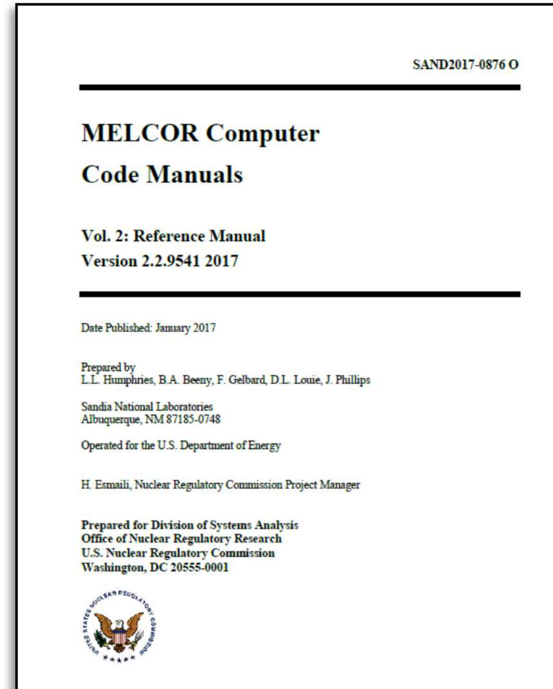
Prepared by
L.L. Humphries, B.A. Beeny, F. Gelbard, D.L. Louie, J. Phillips

Sandia National Laboratories
Albuquerque, NM 87185-0748

Operated for the U.S. Department of Energy

H. Esmaili, Nuclear Regulatory Commission Project Manager

Prepared for Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001



SAND2017-0876 O

MELCOR Computer
Code Manuals

Vol. 2: Reference Manual
Version 2.2.9541 2017

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L.L. Humphries, B.A. Beeny, F. Gelbard, D.L. Louie, J. Phillips

Sandia National Laboratories
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H. Esmaili, Nuclear Regulatory Commission Project Manager

Prepared for Division of Systems Analysis
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U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

MELCOR 2.2 Quicklook Overview of Model Changes in MELCOR 2.2

SAND2018-13524

Volume I: User Guide

SAND2018-13559 O

Volume II: Reference Manual

SAND2018-13560 O

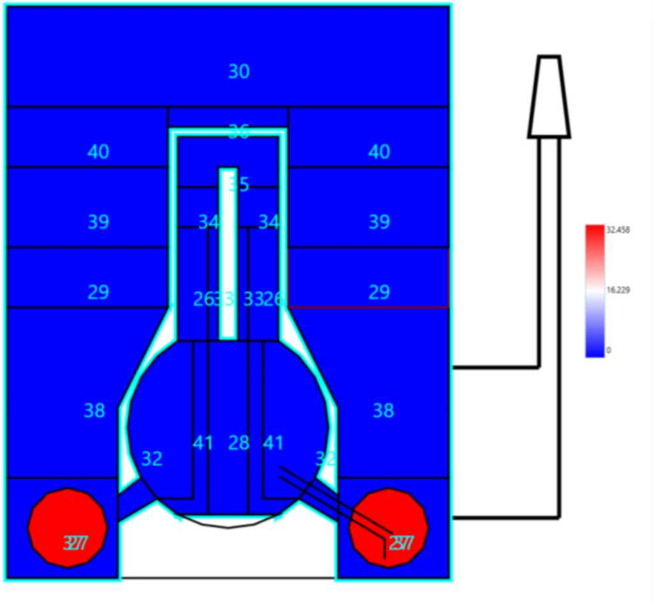
Task 2: Model enhancement of MELCOR, Accident Tolerant Fuel

- Near Term
  - Coated Cladding
    - Multiple vendors
    - Standard zirconium alloy material with thin coating applied to outside
    - Intent is to reduce corrosion and metal-water reaction
  - Doped fuel pellets
    - Reduce PCI by increasing pellet creep
  - Steel cladding (FeCrAl)
- Long Term
  - SiC (ceramic composite) Cladding
    - Pursued by multiple vendors
  - U_3Si_2 fuel pellets
    - Higher fuel density
    - Limited information on fuel performance
  - Lightbridge
    - Helical cruciform fuel rods
    - Metallic fuel co-extruded with clad

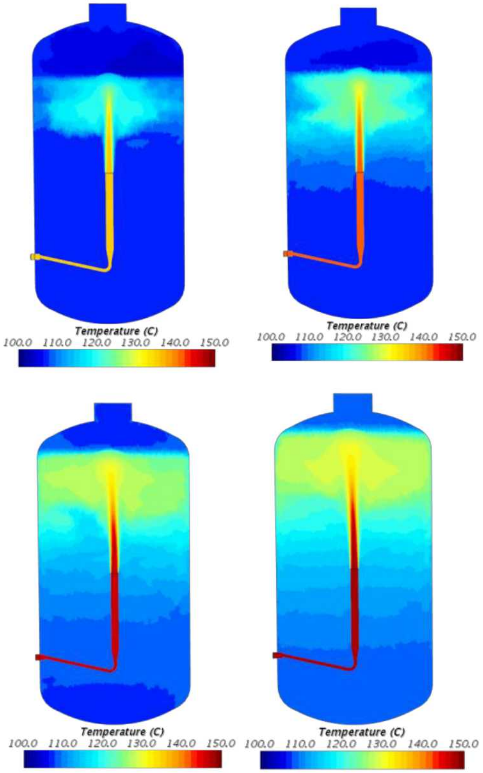
Task 2: Model enhancement of SAMPSON (Containment phenomena)

The BSAF project showed the importance of containment mechanism in the W/W and D/W regarding phenomena of temperature and gas stratification.

SAMPSON example of Mark-I containment



Example of CFD approach to containment stratification

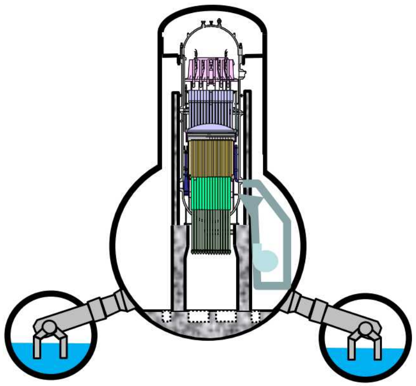


IAE joined an international benchmark in order to benchmark SAMPSON containment model in complex situations (hydrogen stratification, pool stratification)

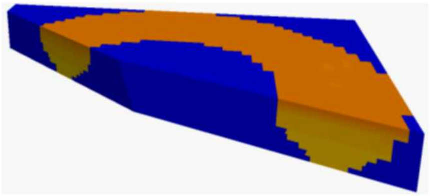
Task 2: Model enhancement of SAMPSON (pool scrubbing)

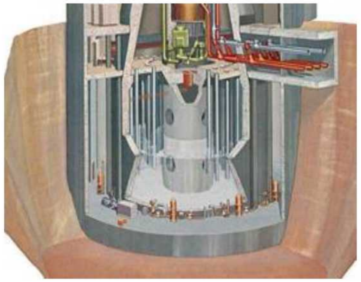
IAE has developed a 3D tool for evaluation of the temperature distribution in the pool and it is currently participating in the NEUGENIA IPRESCA project for extension of the pool model to pool scrubbing mechanisms.

SAMPSON/POOL3D application to two types of Mark containment

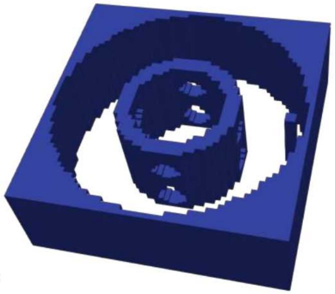


Mark I containment

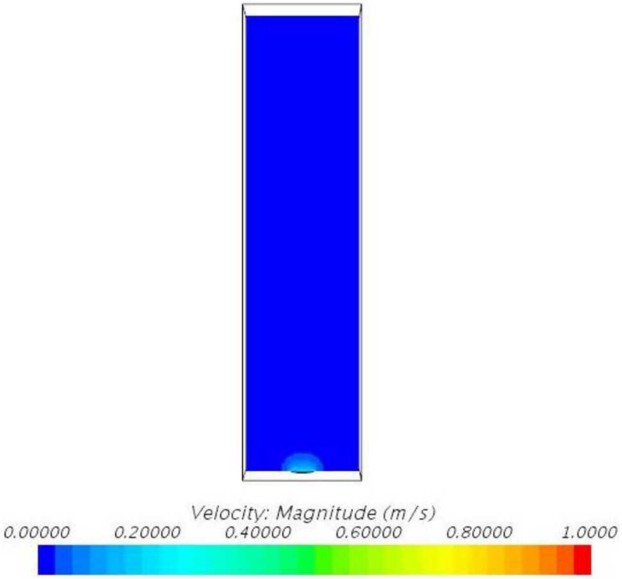




Mark II containment

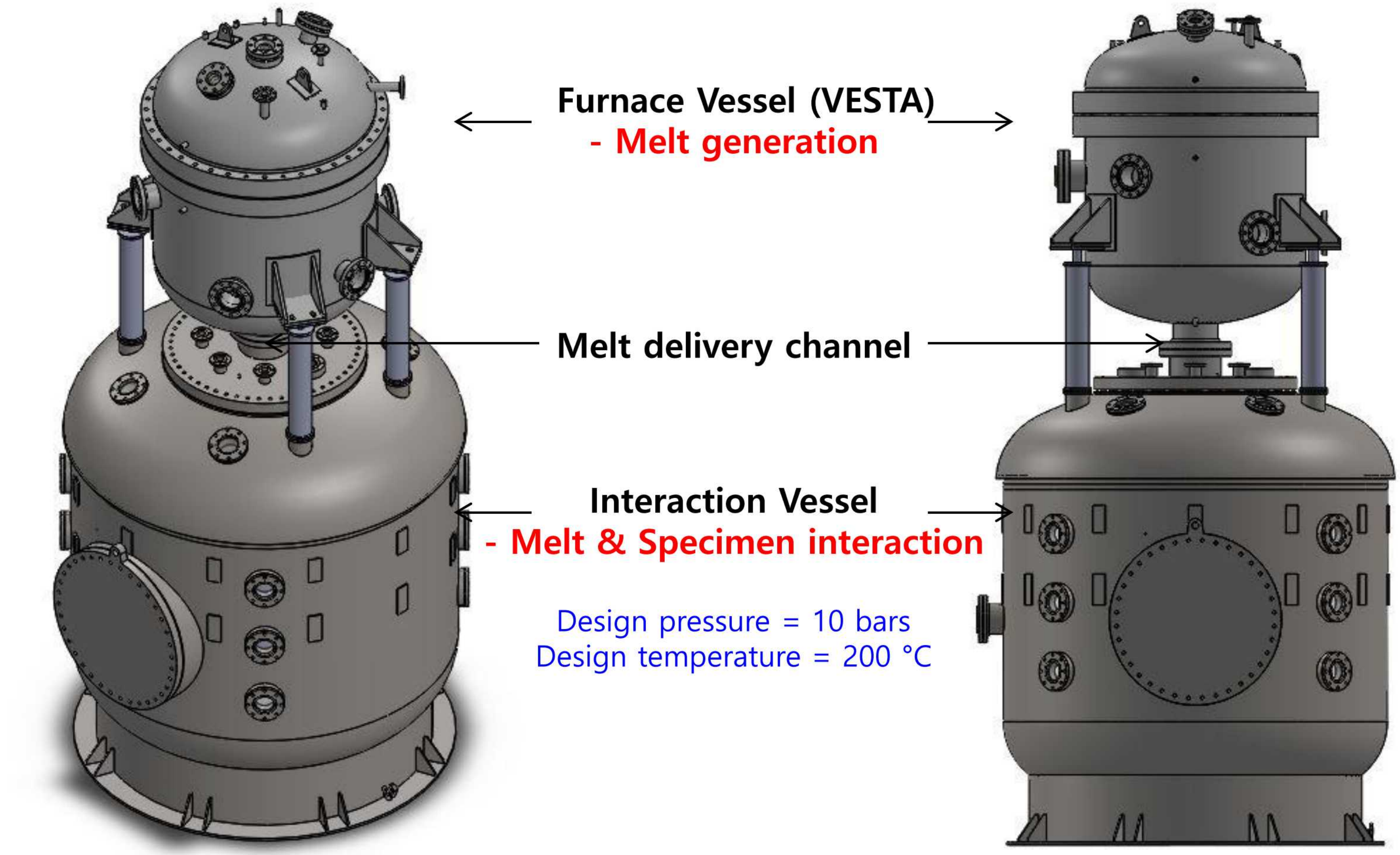


Example of CFD effort to support SAMPSON scrubbing modeling

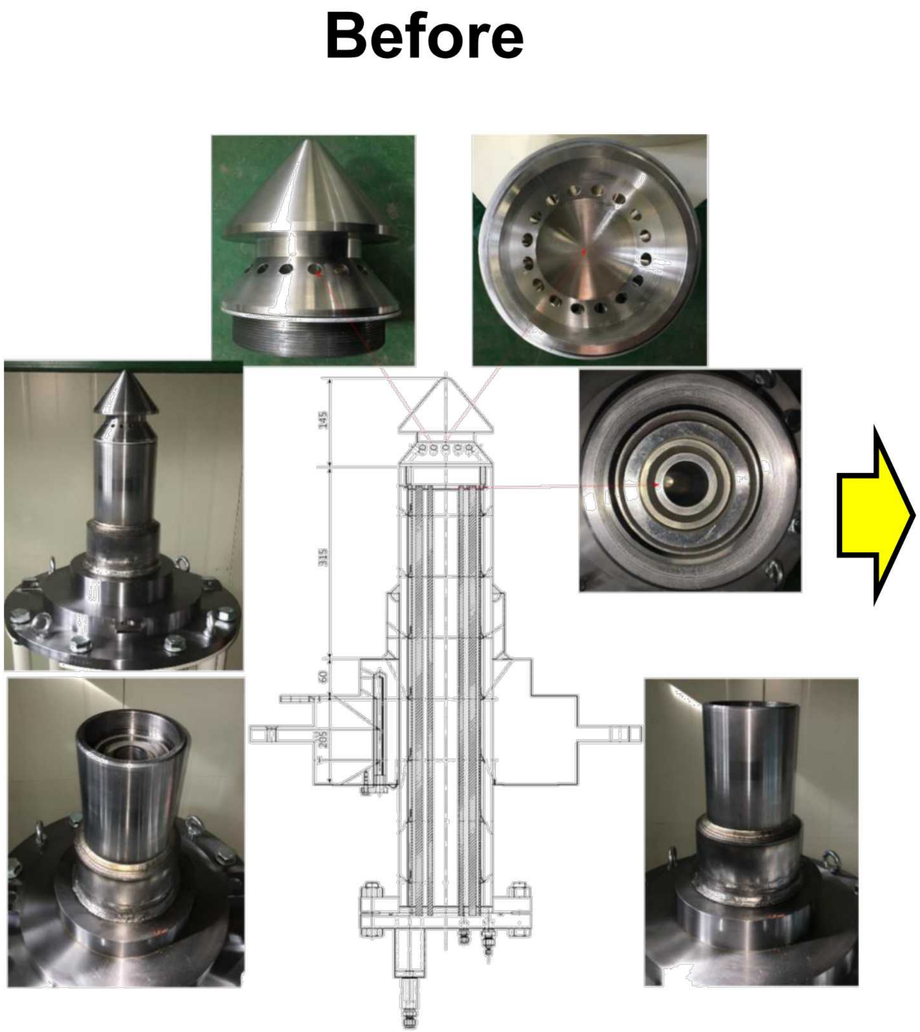


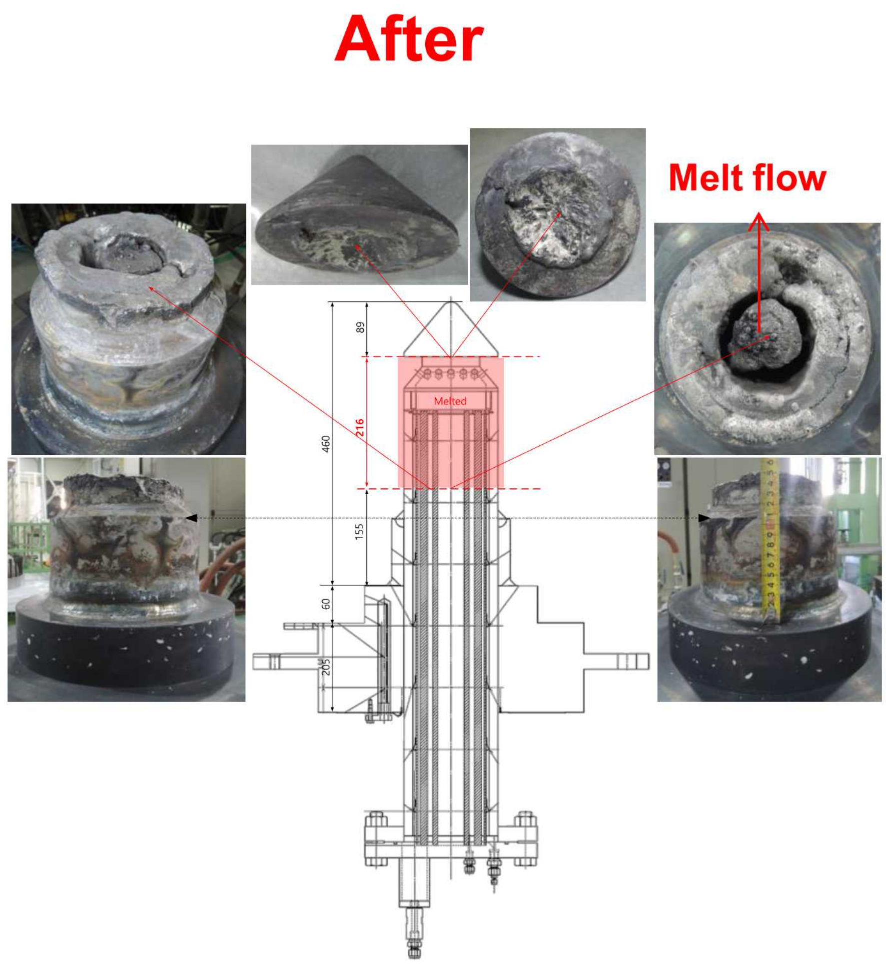
Task 4-3: Experimental activity on lower head penetration failure – KAERI experiment

- Large-scale experiment
- Prototypic materials and environment

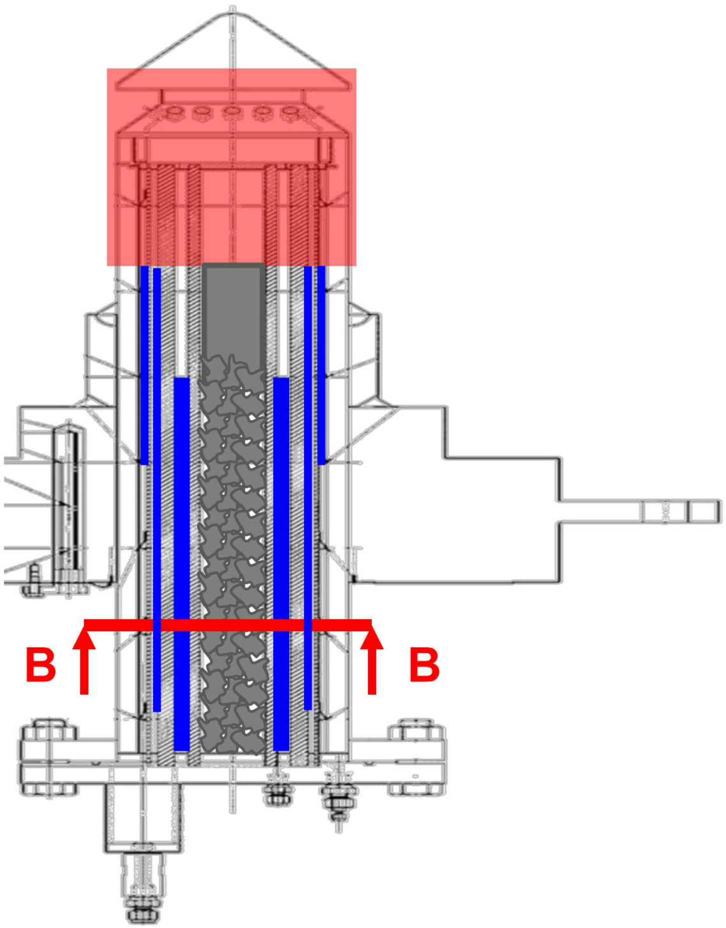


Task 4-3: Experimental activity on lower head penetration failure – CRGT experiment





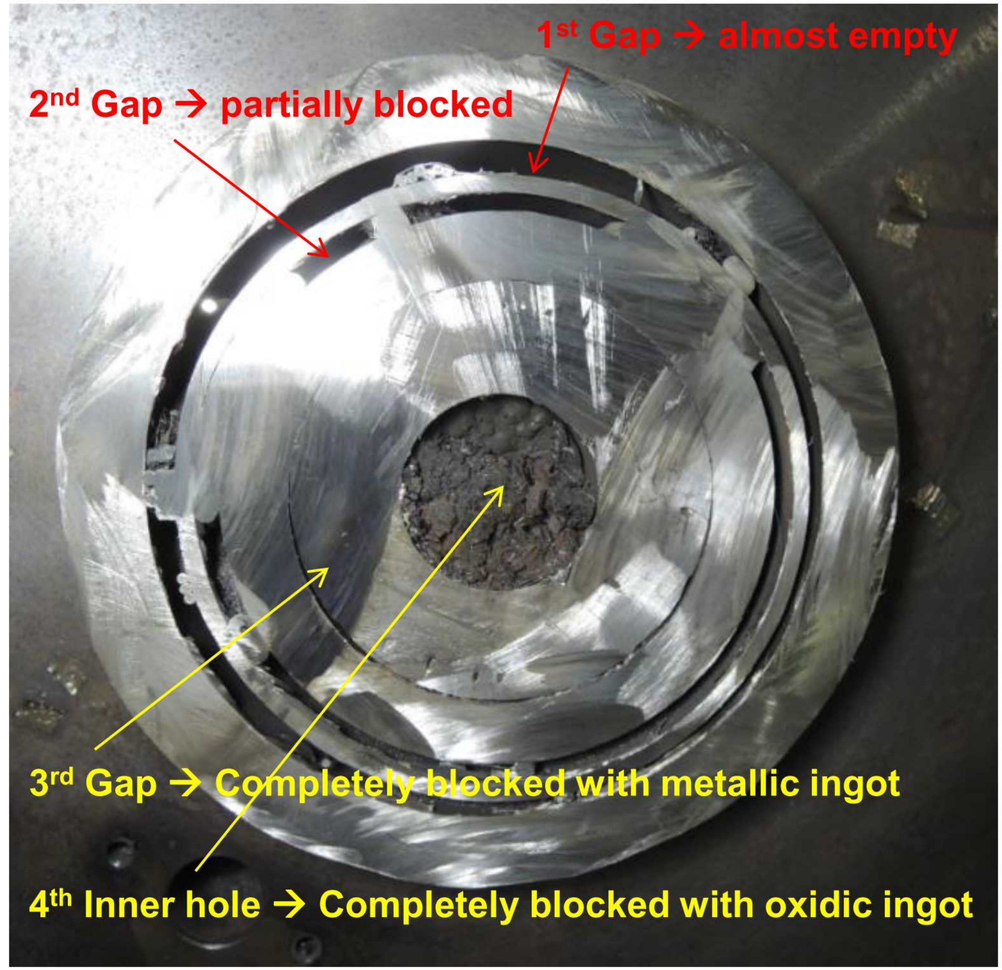
Task 4-3: Experimental activity on lower head penetration failure – CRGT experiment



Oxide brittle material

SS metal ingot

B-B cross section



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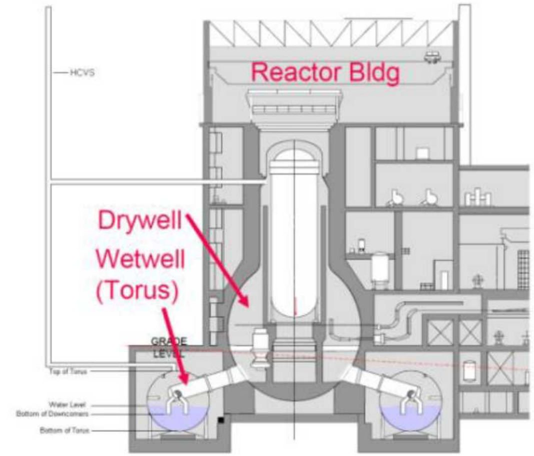
Severe Accident Code Advancements: Standalone Melt Spreading and Debris Coolability Model Development

Driver:

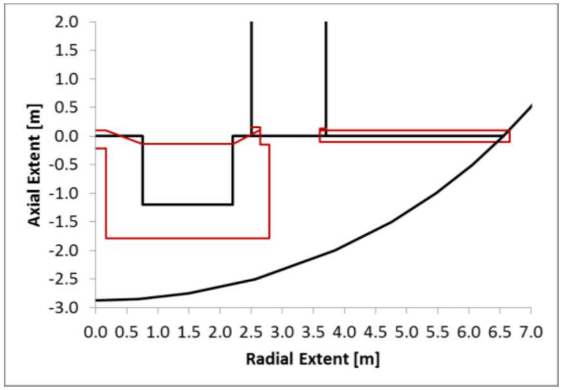
- NRC severe accident capable vent Order EA-13-109.
- Industry implementing Severe accident Water Management (SAWM) approach as an alternative to installing external filters on BWR containment vents.
- Analysis tools needed to support plant-specific implementation of SAWM in SAMGs; specifically:
  - Melt spreading model (MELTSPREAD) to predict core debris distribution in containment following vessel breach
  - Core debris coolability model (CORQUENCH) to assess long-term stabilization and coolability of post-spread debris.



Mk I BWR Containment

Accomplishments:

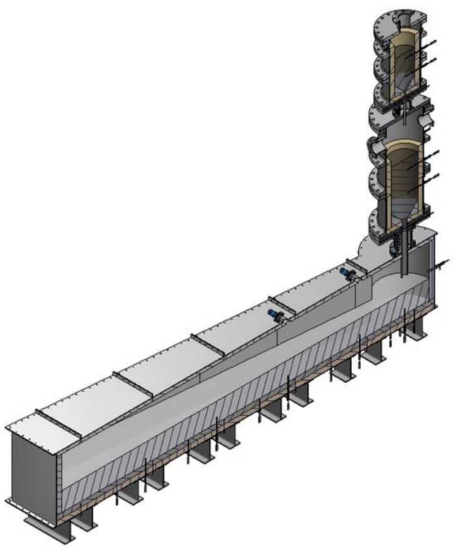
1. **Model improvements finalized, and both MELTSPREAD and CORQUENCH open-sourced to support industry in implementing SAWM.**
2. **SAWM scoping analysis with these tools for a US BWR (Peach Bottom) completed to support industry implementation efforts.**

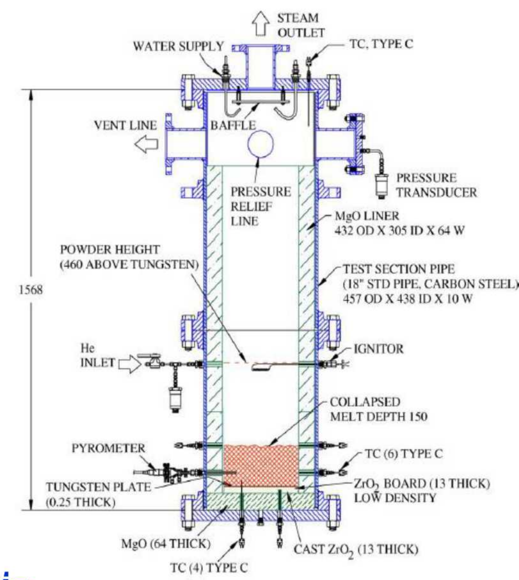


Fukushima Unit 1 Calculated Cavity Erosion Profile

Task 2: Planned Participation in ROSAU Program

- **5 year OECD-sponsored program launching 1 May focused on Reduction of Severe Accident Uncertainties.**
- **Objectives: provide reactor material test data and validated models addressing the following knowledge gap areas:**
  - Coolability of high metal content, in-vessel core debris: support core recovery actions and better understand ex-vessel debris coolability.
  - Underwater spreading of core debris: no reactor material data in this area; extent of spreading strongly affects debris coolability.
- **Planning 6 large scale underwater spreading tests and 5 additional core debris cooling experiments during the 5 year program.**
- **SNL and IAE will both cooperatively perform severe accident code assessments as part of this program (MELCOR and SAMPSON)**





Task 2: Planned Code Assessment Activities as Part of the ROSAU Project

- MELTSPREAD and CORQUENCH will be further refined and validated against experiment results obtained the results of the ROSAU project as program moves forward.
- In addition, Argonne will collaborate directly with IAE on code development activities in this area. Planned first-year activities are as follows:

1. Documentation of Core Debris Coolability Models:

- Document existing water ingression and melt eruption core debris cooling models in CORQUENCH in sufficient detail to support implementation of these equivalent models in SAMPSON

2. MELTSPREAD-CORQUENCH/SAMPSON-DSA Crosswalk:

- Perform ex-vessel debris spreading and coolability analysis with MELTPSREAD and CORQUENCH for an existing Japanese plant, given melt pour conditions predicted by SAMPSON as input.
- Key results will be benchmarked against similar results obtained with SAMPSON.

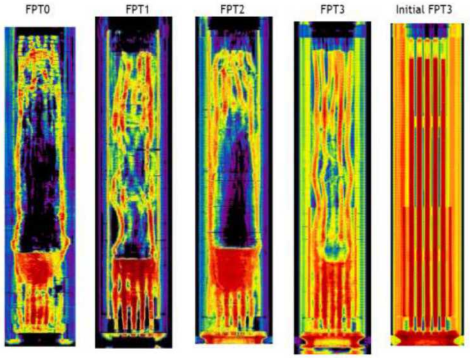
Summary of Activities – 1

- **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**~~
- **Task 1-3 RCIC Modeling and Experiments**
  - Cooperative evaluation of extended RCIC performance outside of original operating range
  - Currently performing experimentation and testing
  - 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 1-5 Fission Product Release Assessments**
  - Completed analyses
- **Task 2 MELCOR Model Enhancements**
  - Begun to implement necessary changes to capture ATF behavior during severe accidents
  - Have improved code robustness and numerics

Summary of Activities – 2

- **Task 2 SAMPSON Model Enhancements**
  - Containment phenomena
- **Task 2-1 ROSAU**
  - Kick-off meeting in May (Hosted by SNL)
- **Task 3 Fundamental effects of sea water and lower plenum debris behavior**
  - Variation of heat transfer modeling with salt concentration
  - Analysis of deposition on the surface
- ~~**Task 4-1 Hydrogen CFD analyses**~~
- **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
  - Currently performing CFD modeling
- **Task 4-3 Experimental studies of lower head penetration failure**
  - Performing large prototypic experiments

Source: Tokyo Electric Power Company



QUESTIONS & COMMENTS?

References

- **Code manuals, sample problem, and source codes can be accessed at the following links:**
  - CORQUENCH4.1-beta release:

    https://anl.box.com/s/ja6veg9rlsqrqv4icil93ir1vozxghls

  - MELTSPREAD3 release:

    https://anl.box.com/s/5zin9jsq0k9mlik55jba6w9v558ttv1o

- **M. T. Farmer, "A Case Study on Severe Accident Water Addition and Water Management (SAWA/SAWM) for a Mark I Containment," ANL-18/21, September 2018. https://publications.anl.gov/anlpubs/2018/09/145503.pdf**