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Presentation of Fukushima Analyses to U.S. Nuclear Power Plant Simulator Operators and Vendors

Donald A. Kalinich, Jeffrey N. Cardoni, and Douglas M. Osborn

Prepared by
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Presentation of Fukushima Analyses to U.S. Nuclear Power Plant Simulator Operators and Vendors

Donald A. Kalinich, Jeffrey N. Cardoni, and Douglas M. Osborn
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Abstract

This document provides Sandia National Laboratories' meeting notes and presentations at the Society for Modeling and Simulation Power Plant Simulator conference in Jacksonville, FL. The conference was held January 26-28, 2015, and SNL was invited by the U.S. nuclear industry to present Fukushima modeling insights and lessons learned.

ACKNOWLEDGMENTS

This work is funded through the U.S. Department of Energy's Light Water Reactor Sustainability Program.

CONTENTS

NOMENCLATURE

BOP	Balance of Plant
BWR	Boiler Water Reactor
BWROG	Boiler Water Reactor Owners’ Group
DOE	U.S. Department of Energy
INPO	Institute of Nuclear Power Operations
PVM	Parallel Virtual Machine
PWR	Pressurized Water Reactor
Q&A	Questions and Answers
RCS	Reactor Coolant System
R&D	Research and Development
SAMGs	Severe Accident Management Guidelines
SNL	Sandia National Laboratories
SOARCA	State-of-the-Art Reactor Consequence Analyses
T-H	Thermal Hydraulics
USNRC	U.S. Nuclear Regulatory Commission

1. INTRODUCTION

This section provides the motivation for Sandia National Laboratories' (SNL's) meeting and presentations at the Society for Modeling and Simulation Power Plant Simulator conference in Jacksonville, FL. This conference was held January 26-28, 2015, and SNL was invited by the U.S. nuclear industry to present Fukushima modeling insights and lessons learned.

1.1 Background

SNL attended a Boiler Water Reactor Owner's Group (BWROG) subcommittee meeting on emergency operating procedures and severe accident guidance the week of June 9, 2014. The BWROG meeting was hosted by Xcel Energy's Monticello Nuclear Generating Plant. During this meeting, the plant's training team showed various emergency and accident scenarios using their new plant simulator which included a first-of-a-kind interface with the SNL developed severe accident code, MELCOR (see Section 1.2 for further information on MELCOR). Also during this meeting, SNL presented current insights into the Fukushima Daiichi nuclear accident. As a result of this meeting, SNL has continued to provide additional MELCOR modeling insights to the Monticello simulator instructors. As a result of this continued contact, the Monticello Simulator Lead/Senior Operations Simulator Instructor, Joseph C. Yarbrough invited SNL to attend the Society for Modeling and Simulation Power Plant Simulator Conference.

SNL was asked to provide presentations and discussions related to Fukushima Daiichi impacts. Mr. Yarbrough felt SNL's modeling of Fukushima Daiichi, and comparing it with the known data and another severe accident code used by industry, MAAP, would be of interest and insightful for the simulator community. While MELCOR modeling is not directly applicable to all U.S. nuclear industry plant simulators, at the very least the phenomena observed at Fukushima Daiichi and the modeling would be of interest.

1.2 MELCOR

MELCOR is a fully integrated, engineering-level computer code that models the progression of severe accidents in light-water reactor nuclear power plants [1]. MELCOR is being developed at SNL for the U.S. Nuclear Regulatory Commission as a second-generation plant risk assessment tool, and the successor to the Source Term Code package. A broad spectrum of severe accident phenomena in both boiling water reactors (BWRs) and pressurized water reactors (PWRs) is treated in MELCOR in a unified framework. These include thermal-hydraulic response in the reactor coolant system, reactor cavity, containment, and confinement buildings; core heatup, degradation, and relocation; core-concrete attack; hydrogen production, transport, and combustion; fission product release and transport behavior. MELCOR applications include estimation of severe accident source terms, and their sensitivities and uncertainties in a variety of applications. MELCOR is also used to analyze design basis accidents for advanced plant applications (e.g., the Westinghouse AP-1000 design and the GE Hitachi Nuclear Energy ESBWR design).

Current applications of MELCOR include the USNRC sponsored State-of-the-Art Reactor Consequence Analyses (SOARCA) [2-5], and the U.S. Department of Energy (DOE) sponsored Fukushima Daiichi accident analyses [6-8].

2. MEETING AND PRESENTATION

The audience at the presentation was basically divided between nuclear power plant staffs (simulator operators, trainers, operations staff, etc.) and simulator vendors. As there is no mandate in either industry standard or regulations for the treatment of severe accidents in simulators, the interest of the nuclear power plant staff in the implementation of severe accident models in simulators is strictly dependent on the internal needs/desires of the individual plants (operators). The interest of the simulator vendors seems to be in terms of providing simulator severe accident modeling capability to both cater to the current non-regulatory interests and as a way to get ahead of the potential for future regulations either from the USNRC or industry self-imposed (i.e., through the Institute of Nuclear Power Operations – INPO).

Except for CORYS Thunder, all of the vendors that treated severe accident modeling did so using MAAP. The rational for using MAAP was explained as the operators already have severe accident MAAP models and MAAP code licenses. In the case of CORYS Thunder, they justified using MELCOR based on its parallel virtual machine (PVM) feature which allows the MELCOR reactor coolant system (RCS) model to be easily coupled with the Thunder T-H model for the containment and balance of plant (BOP).

Based on the Q&A after the SNL presentation, there was interest in the severe accident insights that have come from the SNL Fukushima analyses [6-8]. However, it was also apparent (based on the Q&A, other presentations, and one-on-one conversations) that at this time neither the nuclear power plant staffs nor the simulator vendors need or desire national laboratory-type severe accident models/analyses. The main reason for this is that the current state-of-the-art/best practices (as illustrated by recently completed SOARCA analyses [2-5]) result in plant models that run much slower than real-time, which is unacceptable in a simulator environment. Another reason is that without a driving need for complexity of the current models (which contributes to their slow execution) the cost of developing such models for simulator use cannot be justified.

Here are some points of interest that came up during the presentations and one-on-one discussions.

- In the CORYS Thunder presentation, it was stated that they use an explicit solver for their T-H code (Thunder), that they run with a 0.01 second time-step, and that their simulator models (on the order of 100 to 300 “nodes” (control volumes)) run faster-than-real time. What makes this interesting is that they use an explicit solver to avoid (what they claim is) the higher computational cost of inverting matrices as part of an implicit solver.

- It was noted that there are cases where simulator model results do not match measured plant conditions and that the simulator operators will “tweak” their models to get a better match. This came up in the context of discussing how “tweaks” have been implemented in many of the Fukushima analyses [6-8] to address areas where the severe accident models cannot predict the plant data (e.g., wetwell cooling by torus room flooding, wetwell partial mixing, drywell head leakage, cooling water injection rates).

- As part of the Q&A after the SNL presentation, the question was asked, "Of the MAAP and MELCOR conceptual views of core damage progression (i.e., formation of molten pool vs. formation of solid debris bed), to which did the SNL presenter ascribe more validity." The answer provided[1] was, "Both. Neither. Or, with all facetiousness aside, that there is not sufficient data on full-scale core damage progression to declare one "better" than the other. Hence, the best way to treat this is to consider both. This is why ultimately severe accident analysis has to be done with consideration of uncertainty with regards to both inputs and models."

Appendix A provides the agenda for the conference. Appendix B and Appendix C provide the slides presented by SNL at the conference.

[1] The audience was informed that this answer was the personal professional opinion of the SNL presenter and was not an official response of SNL nor DOE.

3. SUMMARY

The simulator operators and vendors have an interest in severe accident despite there being no standards requirement or regulatory requirement. However, their need is for models/analyses that run in real time, which are much simpler than the current MELCOR state-of-the-art/best practices models. The conference audience was interested in the SNL presentations and their insights – at least from an intellectual perspective.

Given what could be characterized as a tepid interest in severe accident analysis, there were hints that there was a future potential to have to address severe accident issues -- for example, severe accident management guidelines (SAMGs) response and modifications -- in simulators. If this were to occur, then the need for handling severe accidents (specifically core damage progression) with a fidelity commensurate with that of the SAMG requirements could result in the simulator vendors (or their contractors) turning to SNL for guidance in this area. Specific severe accident R&D areas would be:

- Code improvements in MELCOR that would shorten run times to real time or faster. This includes updating the code's circa 1980 numerical solver as well as restructuring the code to allow its solver to be easily updated as numerical solver state-of-knowledge improves over time.

- Improvements to MELCOR nuclear power plant models that would shorten run times to real time or faster. This involves not only simplifying MELCOR nuclear power plant models, but also includes developing detailed phenomenon-specific models outside of MELCOR whose results are used to create simplified "abstraction" models (i.e., capturing the important physics) that are implemented into the MELCOR nuclear power plant models.

4. REFERENCES

1. Gauntt, R.O., et al., NUREG/CR-6119, “MELCOR Computer Code Manuals, Vol. 2: Reference Manuals, Version 1.8.6 (Vol. 2, Rev. 3),” USNRC, Washington, D.C., 2005.

2. USNRC, NUREG-1935, “State-of-the-Art Reactor Consequence Analyses (SOARCA) Report,” Washington, DC, 2012.

3. SNL, NUREG/CR-7110 Volume 1, “State-of-the-Art Reactor Consequence Analyses Project Volume 1: Peach Bottom Integrated Analysis,” USNRC, Washington, DC, 2012.

4. SNL, NUREG/CR-7110 Volume 2, “State-of-the-Art Reactor Consequnce Analyses Project Volume 2: Surry Integrated Analysis,” USNRC, Washington, DC, 2012.

5. SNL, NUREG/CR-7155, “State-of-the-Art Reactor Consequence Analyses Project: Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station – DRAFT Report,” USNRC, Washington, DC, 2013.

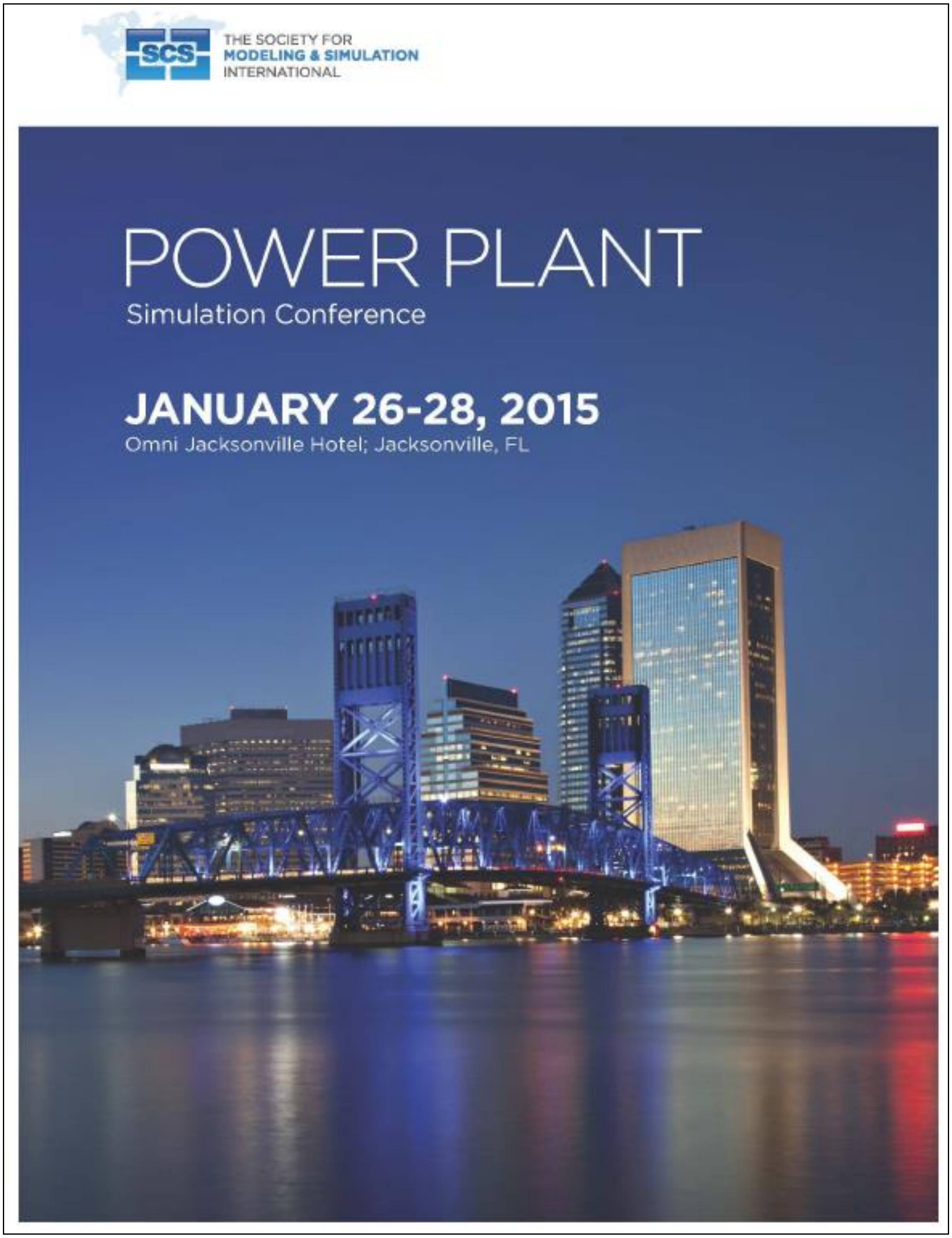
6. Gauntt, R.O., et al., SAND2012-6173, "Fukushima Daiichi Accident Study (Status as of April 2012)," SNL, Albuquerque, NM, 2012.

7. Kalinich, D.A., D.R. Denman, and D. Brooks, *DRAFT SAND report*[2], “Fukushima Daiichi Unit 1 Uncertainty Analysis – Exploration of Core Melt Progression Uncertain Parameters”, SNL, Albuquerque, NM, 2015.

8. Luxat, D., D. Kalinich, J. Hanophy, EPRI-3002004449, “Modular Accident Analysis Program (MAAP) – MELCOR Crosswalk, Phase 1 Study,” EPRI, Palo Alto, CA, 2014.

[2] The body of work for this report is complete and is currently in the final processes of completion.

APPENDIX A:
THE SOCIETY FOR MODELING AND SIMULATION POWER PLANT SIMULATOR CONFERENCE – AGENDA



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Brought to you by :

The Society for Modeling & Simulation International (SCS)

Nuclear Agenda

Monday, 26 January 2015

Welcome and Introductions by Jeff Mercer (Southern Nuclear—Vogtle)
8:30 – 9:00 Florida Salons AB

Session 1: Engineering & Human Factors Simulation
9:00 – 10:00 Florida Salons AB Chair: Jim Redwine (Columbia)

- 9:00: *Deploying a Part-Task Trainer for St. Lucie Unit 1 Shutdown Cooling Operations* by Vincent Gagnon (L-3 MAPPS)
- 10:00: *The THOR 3-G Graphical Development Tool* by Dave O'Farrell (CORYS) & Dave Young (CORYS)

10:00—10:30 COFFEE BREAK

Session 2: Next Generation Simulators
10:30 – 12:00 Florida Salons AB Chair: Terry Damashek (Wolf Creek)

- 10:30: *New Plants in Baraka (UAE)* by Majid Mirshah/Oussama Ashy (WSC)
- 11:00: *Preparing the Instructor Station for Windows Touch* by Raymond Dimitri-Hakim (L-3 MAPPS)
- 11:30: *Next Generation Modeling, and How You Get There* by Mike Fendley (GSE) **JOINT WITH FOSSIL TRACK**

12:00—13:30 LUNCH ON YOUR OWN

Session 3: Virtual Simulation
13:30 – 15:00 Florida Salons AB Chair: Mike Galle (Farley)

- 13:30: *Robinson—Major DCS Upgrade Training Using Glass Top Simulator* by Mladen Udbinac (WSC)
- 14:00: *3D Virtual Training Concepts by* Scott Zepplin (GSE)
- 14:30: *From Training Simulators to Learning Simulators* by Raymond Dimitri-Hakim (L-3 MAPPS)

15:00—15:30 COFFEE BREAK

Session 4: Fukushima Simulation Impacts
15:30 – 17:00 Florida Salons AB Chair: Joe Yarbrough (Monticello)

- 15:00: *Learnings from Fukushima on Severe Accident Phenomena* by Don Kalinich (Sandia National Labs)
- 15:30: *Fukushima Benchmarking Results/FLEX at Monticello* by Joe Yarbrough (Monticello) & Alex Broyles (Indian Point)
- 16:00: *Things to Know Before You Implement a Severe Accident Model* by Scott Zepplin (GSE)

Nuclear Agenda

Tuesday, 27 January 2015

Session 5: **Joint Session with Fossil Track**
8:30– 10:00 Florida Salons AB Chair: Jeff Mercer (Southern Nuclear—Vogtle)

- 8:30: *How Does ESKOM Apply Virtual Training at Their New 4,800MW Supercritical Power Plants?* by Kevin Brink (ESKOM) & Abrie Venter (Samahnzi)
- 9:00: *Research with a Room Full of Virtual Panels* by Kirk Fitzgerald (INL)

-One Size Fits All – Research Simulator for Multiple Plants

-The Simulator – a description of the simulator

-Analog to Digital – the "how" of what we are doing

-Modernization – "what" we are doing

- 9:30: *Enhancing Simulator Audio-Visual Capabilities* by Bernard Gagnon & Vincent Gagnon (L-3 MAPPS)

10:00—10:30 COFFEE BREAK

Session 6: Fukushima Simulation Impacts
10:30– 12:00 Florida Salons AB Chair: Joe Yarbrough (Monticello)

- 10:30: *How to Prepare for Your Electrical Models for Extended Blackout* by Scott Zepplin (GSE)
- 11:00: *First Principles Flooding Models for Internal and External Flooding* by Laurent Leo (CORYS) & John Shriver (CORYS)
- 11:30: *Severe Accident Solution for Training and Emergency Preparedness* by George McCullough (GSE)

12:00—13:30 LUNCH ON YOUR OWN

Session 7: Recent Simulator Upgrades
13:30– 15:00 Florida Salons AB Chair: Gary Degraw (River Bend)

- 13:30: *Vogtle Simulator Upgrades (RM2300 & Open 6)* by Nakeya Crawford (Vogtle)
- 14:00: *Upgrading the Heysham 1 Simulator to Support Dual-Unit Operations* by Vincent Gagnon (L-3 MAPPS)
- 14:30: *Turbine Control Upgrade at Farley* by Mike Galle (Farley)

15:00—15:30 COFFEE BREAK

Session 8: International Simulator Upgrades
15:30– 17:00 Florida Salons AB Chair: Jeff Mercer (Southern Nuclear-Vogtle)

- 15:30: *Overcoming Challenges on the Daya Bay Simulator I/O System Replacement Project* by Gregory Zakaib (L-3 MAPPS)
- 16:00: *TPC Chin Shan GE BWR Simulator Upgrade* by Joel Dixon (WSC)
- 16:30: *Return of Experience of FSS Modernization in Slovakia* by Pascal Gain (CORYS)

Wednesday, 28 January 2015

Session 9: Configuration Management
8:30 – 10:00 **Florida Salons AB** **Chair: Gerry Wyatt (Palo Verde)**

- 8:30: *Mochovce Unit 3&4 Full Scope Simulator Project Key Points and Successes* by Mike Fendley (GSE)
- 9:00: *Experiences with Configuration Management System & Functionality with ANS 3.5* by Dr. Burkhard Holl (KSG)
- 9:30: *Configuration Management at Plant Vogtle 1&2* by Nakeya Crawford (Vogtle)

10:00—10:30 COFFEE BREAK

Session 10: Simulator Testing
10:30 – 12:00 **Florida Salons AB** **Chair: Pablo Rey (Tecnatom)**

- 10:30: *Post Event Simulator Test Experiences at Tecnatom* by Pablo Rey (Tecnatom)
- 11:00: *Pre Validation Testing of Plant Modifications on the Simulator* by Don Dea (Cooper)
- 11:30: *Operators Training in Major Plant Modifications Implemented in Advance in FSS. Testing Program and Limits Identification* by Pedro Diaz (Tecnatom)

12:00—13:30 LUNCH ON YOUR OWN

Session 11: Regulations
13:30 – 15:00 **Florida Salons AB** **Chair: Jeff Mercer (Southern Nuclear-Vogtle)**

- 13:30: *NRC Regulatory Perspective* by Scott Sloan & Larry Vick (NRC/NRR)
- 14:30: *ANS 3.5 Working Group* by Jim Florence (Nebraska Public Power District)

15:00—15:30 COFFEE BREAK

Session 12: Severe Accident
15:30 – 17:00 **Florida Salons AB** **Chair: John Signorelli (Waterford 3)**

- 15:30: *Full Integration of the MELCOR Severe Accident Models with Training Simulators* by Barney Panfil (CORYS)
- 16:00: *Applying MAAP5 for Real-Time Severe Accident Simulation* by Vincent Gagnon (L-3 MAPPS)

Nuclear Agenda

Thursday, 29 January 2015

Session 13: USUG Annual Business Meeting (USUG Members Only)
8:30 – 10:00 Florida Salons AB Chair: Jeff Mercer (Southern Nuclear—Vogtle)

10:00—10:30 COFFEE BREAK

Session 14: USUG Annual Business Meeting (USUG Members Only)
10:30 – 12:00 Florida Salons AB Chair: Jeff Mercer (Southern Nuclear—Vogtle)

12:00—13:30 WORKING LUNCH

Session 15: USUG Regional Workshops
13:30 – 15:00 Tampa, Florida Salons AB, Omni Ballroom

L-3 MAPPS Owners Circle™ Conference Day 1 (invitation only)
15:00 – 17:00 Pensacola Chair: Michael Chatlani (L-3 MAPPS) and Bernhard Weiss (L-3 MAPPS)

Nuclear Agenda

Friday, 30 January 2015

L-3 MAPPS Owners Circle™ Conference Day 2 (invitation only)
8:30 – 15:30 Pensacola Chair: Michael Chatlani (L-3 MAPPS) and Bernhard Weiss (L-3 MAPPS)

Notes

APPENDIX B:
MAAP-MELCOR CROSSWALK PHASE 1 REVIEW AND PROGRESS UPDATE – PRESENTATION

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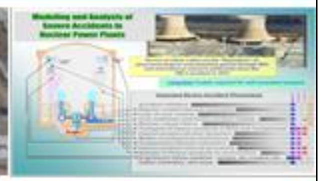

MAAP-MELCOR "Crosswalk" Phase 1 Review and Progress Update

Don Kalinich, Sandia National Laboratories
David Luxat, ERIN Engineering

Topics for discussion

- Review of…
  - work scope
  - FY13 work
- Update on FY14 work
- Sequence Summary Results
- Summary of key results
  - preliminary conclusions
  - limited set of comparison plots
- Future Work

Topics for discussion

- Review of…
  - work scope
  - FY13 work
- Update on FY14 work
- Sequence Summary Results
- Summary of key results
  - preliminary conclusions
  - limited set of comparison plots
- Future Work

FY13 Work

- MAAP and MELCOR 1F1 results were presented in October 2013
- It was identified that the models and accident sequences were not consistent
  - IC operation, FW coastdown, water and component inventories
  - SRV failure vice MSL failure
- Differences in the models and codes result in differences outputs between the codes, making comparisons difficult
- Regardless, preliminary differences were identified
  - In MELCOR, debris cannot completely block a core flowpath; MAAP can completely block a core flowpath
  - MAAP calculates the formation of an in-core molten pool over top a crust, with the molten pool eventually failing into the downcomer/jet pumps; MELCOR calculates solid debris relocating to the lower core plate; eventually failing the plate and allowing debris then relocate into the lower plenum

FY13 Work

- MAAP and MELCOR 1F1 results were presented in October 2013
- It was identified that the models and accident sequences were not consistent
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FY14 Work

- Updated models to reflect latest NEA plant data and BSAF boundary conditions.
- Modified models to minimize water and component inventory differences
- Ran models with a common accident sequence
  - IC operation, FW coastdown, decay heat
  - Turned off MSL failure in MELCOR model; forced SRV failure (stuck-open) at 7 hr in both models
- Developed a set of common results figures
- Documented latest results, comparison, conclusions, and recommendations for Phase 2 work in an EPRI report

Sequence Summary Results

- MAAP calculates core degrades to form a crust with an overlying molten pool within the active core region. The crust/molten pool completely block axial flow through the core. Eventually the molten pool melts through the core shroud, allowing molten material to relocate to the lower plenum via the downcomer/jet pumps. Relocated material forms crust with an overlying molten pool within the lower plenum.
- MELCOR calculates the core degrades mainly in the form of solid particulate debris that relocates to the lower core plate. Some small fraction of molten material relocates into the lower plenum before lower core plate failure. Axial flow through the core is never completely blocked by debris. Once the lower core plate fails, debris relocates into the lower plenum. Degradation and failure of the control rod guide tubes results in further fuel failures. The majority of the relocated material remains solid particulate debris.

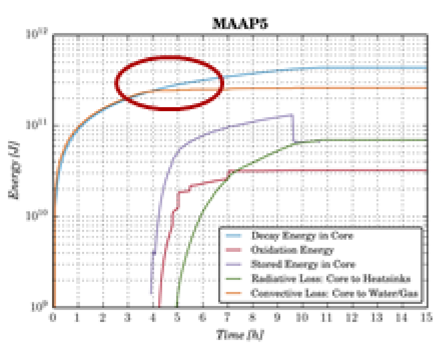
Summary of Key Results

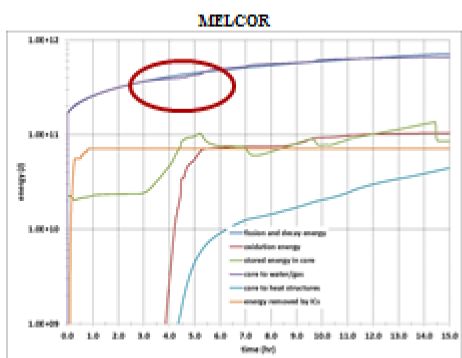
- Differences in code physics models and inputs, along with a paucity of plant data, makes creating "apples-to-apples" plant models and outputs for comparison difficult
- Up to the point of core degradation the code results match relatively well. Difference seen in boil down is due the partitioning of water between RPV volumes in the two codes
- MAAP predicts complete blockage of axial flowpaths; MELCOR has a minimum porosity (code default = 5%)
- MELCOR calculates a much larger amount of energy transferred from core materials to RPV water/gases than MAAP
- MAAP models the heat transfer (area and hx-fer coef.) from particulate debris as decreasing with decreasing debris bed porosity; MELCOR models heat transfer surface area as increasing with the volume of particulate as its effective hydraulic diameter does not vary with porosity

Summary of Key Results

- MAAP does not model radial relocation of particulate debris; MELCOR models radial relocation of particulate debris as a "leveling" effect with a user-defined time constant
- MAAP has an a priori assumption for the lower plenum debris bed; MELCOR performs a series of calculations to determine how debris moves (over the lower plenum spatial nodalization) to form the lower plenum debris bed

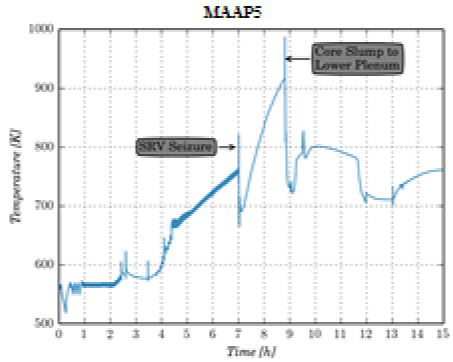
System Energy Balance

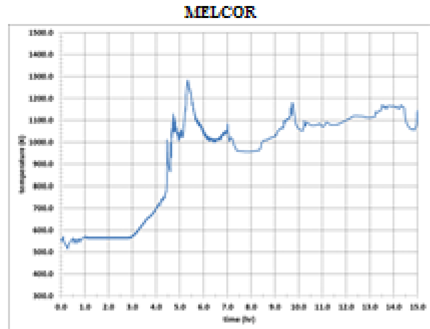




- MELCOR rejects more energy from the core materials than does MAAP5

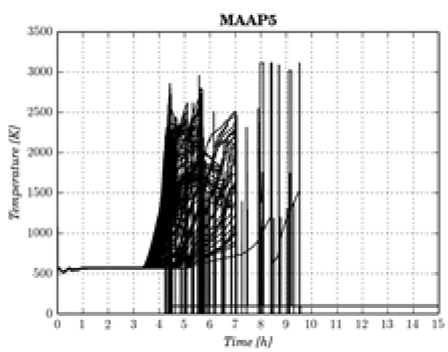
Steam Dome Temperature

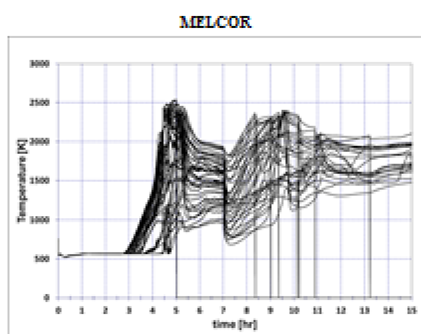




- MAAP5 predicts lower steam dome temperatures compared to MELCOR

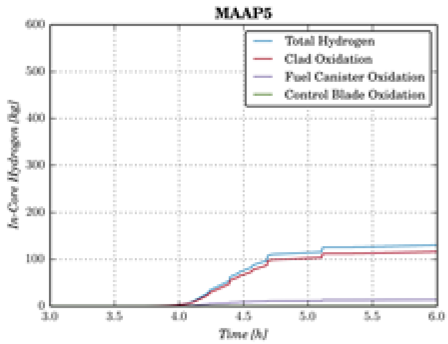
Fuel Temperature

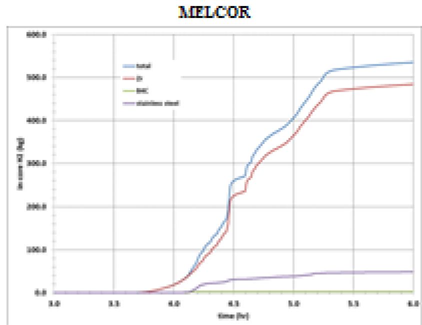




- MAAP5 predicts higher fuel temperatures compared to MELCOR

Hydrogen Generation





- MELCOR predicts higher hydrogen generation compared to MAAP5

Future Work (Phase 2?)

- Detailed examination/comparison of heat transfer from particulate debris
  - Involvement of code developers likely needed
- SRV venting interaction with the wetwell
  - Stand-alone models with "identical" boundary conditions
  - Heat transfer to wetwell pool
  - Pool scrubbing of source term
- Simulation of Recovery Actions
  - Water injection recovery prior to significant loss of the rod-like core geometry
  - Water injection recovery following significant loss of rod-like geometry
  - Water injection following core slumping into the lower plenum
- Ex-Vessel Core Melt Progression
  - Stand-alone models with "identical" boundary conditions
  - Axial and radial concrete ablation
  - Containment heating and pressurizations
  - Source term generation
- Investigation of Simulation Uncertainties

APPENDIX C:
SNL BSAF UPDATE – PRESENTATION

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SNL BSAF Update

Don Kalinich, Jeff Cardoni, Jesse Phillips, Kyle Ross, Randall Gauntt
Sandia National Laboratories

U.S. DEPARTMENT OF ENERGY NNSA

Topics for discussion

Sandia National Laboratories

- Executive Summary
- Brief Model Background
- 1F1 best estimate case results
- 1F3 best estimate case results
- Impact of uncertainty on results
  - why is this important
  - 1F1 and 1F3 example results
- Summary

2

Executive Summary

- 1F1 and 1F3 BSAF best estimate cases completed
  - Accident signatures look similar to previous results; and to most of the TEPCO data
  - Event timings and values are different, but not markedly so
  - ready to move forward to Phase II source term analyses
- Accounting for uncertainty is important in forensic analyses (locus of inputs) and predictive analyses (locus of solutions)

3

Brief Model Background

- SNL MELCOR Fukushima models are based the Peach Bottom SOARCA model; reflects current MELCOR BWR Mk-I best practices
- Models have been updated with the best-available Fukushima inputs (e.g., TEPCO December 2011 data set, IEA November 2013 data set, BSAF BCs); developed surrogate inputs where necessary

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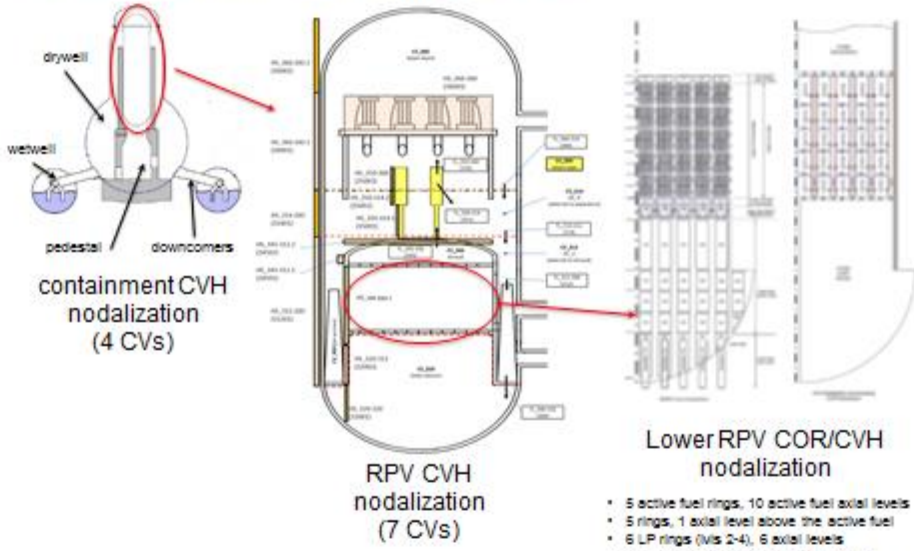
Brief Model Background



- 5 active fuel rings, 10 active fuel axial levels
- 5 rings, 1 axial level above the active fuel
- 6 LP rings (lvls 2-4), 6 axial levels
- 5 ch x 5 byp CVs or 5 ch x 1 byp CVs

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1F1 Best Estimate (BE) Case

- Revised decay heat/RN inventory input with results from SNL SCALE6 analyses
- Implemented BSAF feedwater coastdown injection rate
- IC implementation includes efficiency as a function of RPV pressure; carry-over from previous 1F1 analyses
- SRV gasket failure not implemented; MSL failure model activated
- Did not implement wetwell stratification; not amenable MELCOR lumped-parameter conceptual model nor with the SPARC90 scrubbing model
- BSAF Water injection rates (2% of total) increased by 20x; needed for drywell head lifting/leakage to occur

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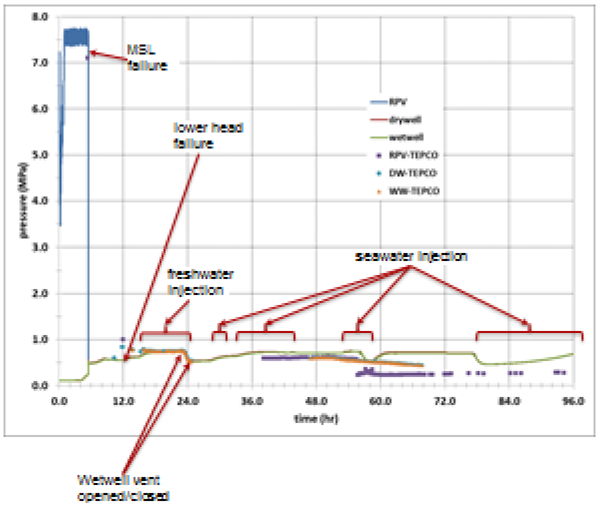
1F1 BE– RPV/DW/WW pressure



- MSL failure at ~6 hr
- LH failure at ~12 hr
- Containment pressure increase at ~12 hr not captured; likely due to relatively "cold" particulate debris (rather than "hot" molten pool) ejection
- late-time pressure changes are related to changes in water injection
- ad hoc leakage model will need to be implemented to capture late-time leakage

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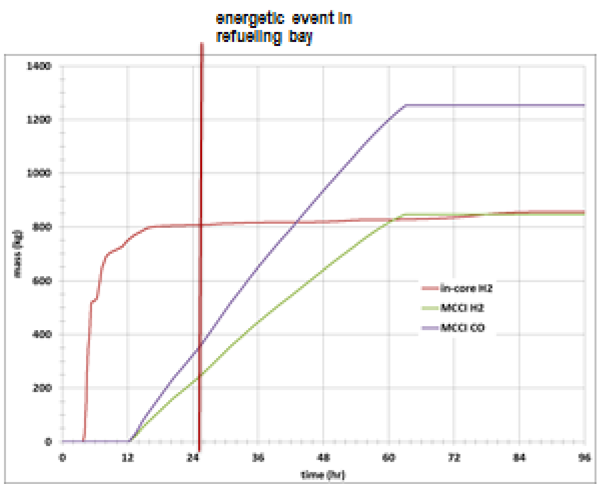
1F1 BE – combustible gases



- sufficient mass of combustible gases (H_2, CO) produced to support an energetic event in the refueling bay at ~25 hr
- lumped-parameter codes operate at too high a granularity to really predict gas composition time evolution; requires detailed analysis (i.e., CFD) to quantify
  - concentrations
  - buoyancy effects
  - steam condensation
  - leakage to/from environment
  - building heat transfer

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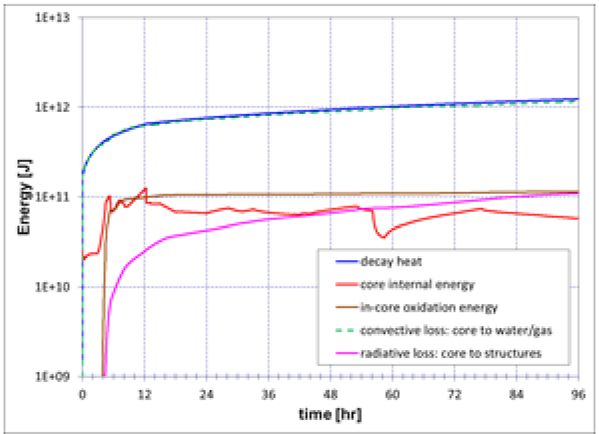
1F1 BE – energy balance



- Majority of core energy input rejected by convection to gas/water (green dashed line = blue line)
- Leads to "cold" particulate debris (instead of "hot" molten pool)
- Likely cause of lack of pressure spike at time of LH failure and need for 20x BSAF water injection to lift drywell head
- This was identified in the MAAP/MELCOR Crosswalk study; path forward yet to be determined

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1F3 BE Case

- BSAF B.C.s included
  - Wetwell and drywell sprays; timing and flow rate
  - Containment vents (via wetwell) and timing
  - After-scram trip and coast-down curves: MSIVs, turbine stop valve, feedwater, fission power, etc.
- BSAF B.C.s not included:
  - RCIC and HPCI
  - Freshwater and seawater injection rates
  - In-core tube failure (SRM, IRM, TIP)
  - Wetwell thermal stratification
- Non-BSAF B.C.s included
  - RCIC B.C. with a level controller – otherwise very comparable flow rates to the BSAF RCIC
  - HPCI B.C. based on preliminary BSAF information – assumes degraded injection after ~30 hours due to low RPV pressure; HPCI tuned to facilitate in-core oxidation to get MSL failure at the "correct" time
  - MSL failure model
  - Seawater injection rates adjusted to get lower head failure
  - Recirculation pump leak added to obtain reasonable containment pressure since the wetwell is only 1 node – 'primes' the containment pressure for the severe accident

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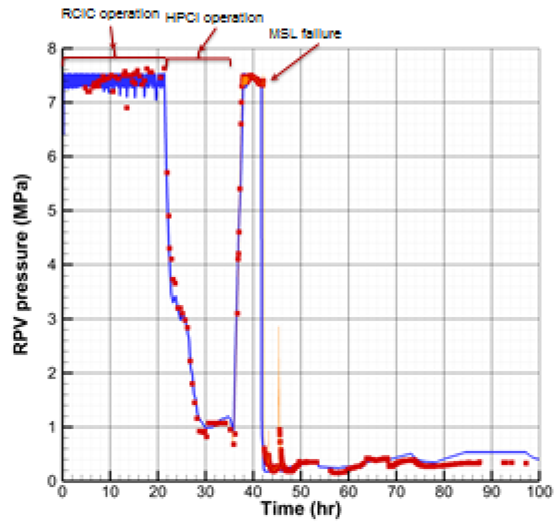
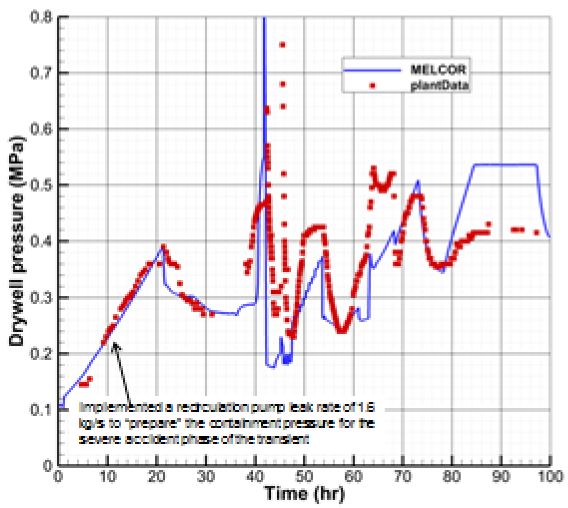
1F3 BE– RPV pressure



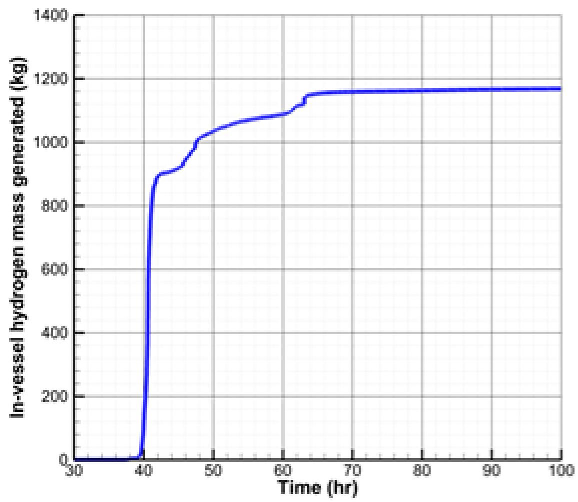
- RCIC and HPCI B.C.s based on initial BSAF information; allowed for general agreement with plant data
- Sets up the severe accident portion of the sequence
- MSL failure calculated to occur around 42 hr

1F3 BE– DW pressure



- general agreement with plant data
- The largest containment pressure peak (near 45 hours after the initial RPV depressurization and first major containment peak) may be caused by core slumping into the lower plenum
- This peak and subsequent peaks are strongly dependent on the assumed WW venting behavior,
  - seawater injection magnitudes
  - core/RPV degradation progression
- too much injection (subcooling) AND too little injection (no water to boil) can suppress containment pressure during certain time periods
- the flatline after 80 hours is an assumed WW gas leak that levels out around 0.53 MPa (based on the plateau around 65-68 hours in the plant data)
- some sort of leak assumption is necessary to transport combustible gas to the Rx building

1F3 BE– H_2 generation



- Rapid oxidation begins about 5 hr after water level drops below TAF

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But what about uncertainty?

- All of our best-estimate/best-practices cases are but one of a locus of potential inputs and their results are but one of a locus of potential solutions
- Uncertainty (in input parameters and models) will produce significant variations the accident sequences
- The impact of this is that…
  - "tweaks" made to fit the forensic data may not be valid over the entire range of input parameter and model uncertainty
  - The next accident may not be within the range of validity of the "tweaks" and current "best-practices"

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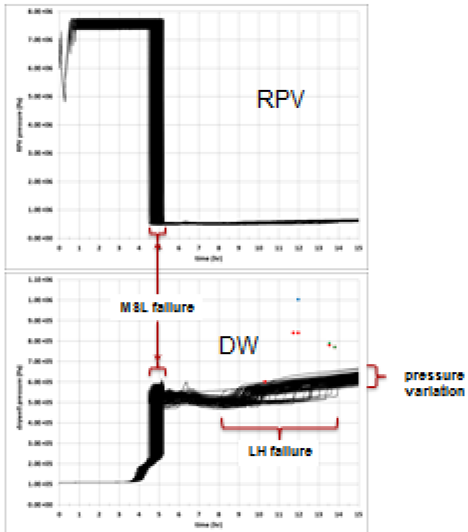
1F1 Example



- 100 realizations with random sampling from the distribution of decay heat curves
- decay heat characterized by combining the ANS-5.1 decay heat uncertainties on primary fissile nuclides with SCALE best-estimate calculations
- Yields variation in
  - MSL failure time
  - LH failure time
  - RPV/containment pressure

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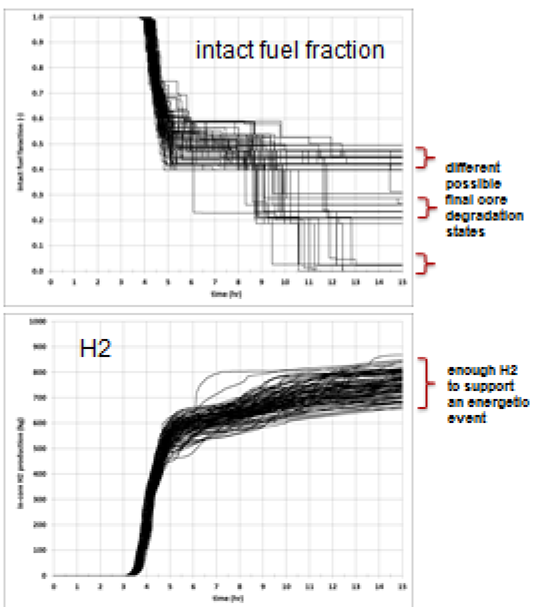
1F1 Example



- H_2 in-core production results have variation in initiation time and late-time value
- These results and those for RPV and containment pressure (previous slide) are due to variation in core melt progress

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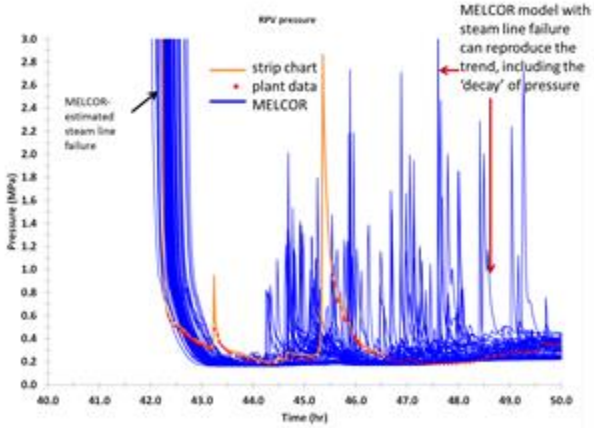
1F3 Example



- 100 realizations that vary
  - wetwell vent opening fraction
  - water injection rate
  - quench parameters
- Some realizations capture the timing, some capture the peak
- There is not a single solution; several different combinations of uncertain variables can reproduce the data trend

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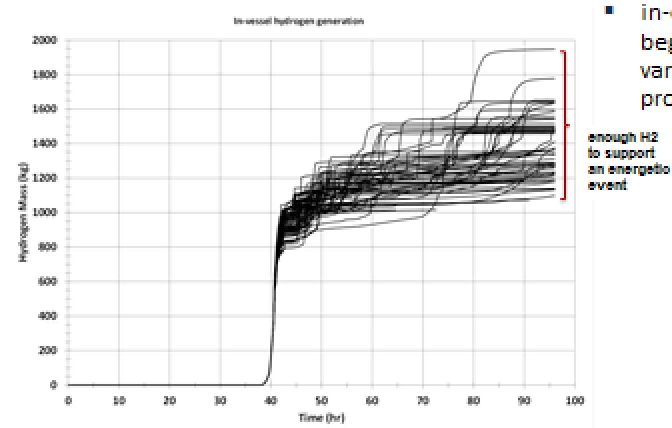
1F3 Example



- in-core H2 generation begins to deviate due to variation in core melt progression

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…and what does this all mean?

- "Tweaked" deterministic analyses are useful for identifying/handling ill-defined phenomena that are postulated to influence forensic results (e.g., 1F2 torus cooling, venting, water injection)
- However, input and model uncertainty have the potential to invalidate "tweaks" tied to forensic results, which can render them invalid for predictive analyses
- Experience has shown that source term results have significant variation; this will be important to handle for BSAF Phase II analyses

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Summary

- 1F1 and 1F3 best estimate accident signatures are similar to those from older models/analyses; they match well enough with the limited data
- Still looking at 1F1 initial ex-vessel behavior
- Accident signatures are very dependent on boundary conditions (e.g., water injection rate, RPV depressurizations mechanism, RCIC & HPCI operation)
- Signatures can be sensitive to uncertainty in BCs and other inputs (explicitly seen in these results and those in the results of a separate 1F1 core-damage progression uncertainty analysis)

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