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Overview of Sandia National Laboratories MELCOR Fukushima Analyses

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Contributors

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Outline

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- MELCOR – severe accident systems code
- In-country technical support; U.S. Embassy in Tokyo
- Initial Fukushima analyses (jointly sponsored by the USDOE and USNRC)
- OECD/NEA Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF) Project – Phase 1 (NRC sponsored)
- MELCOR/Modular Accident Analysis Program (MAAP) Crosswalk (jointly sponsored by the USDOE/EPRI)
- Upcoming work

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MELCOR Computer Code

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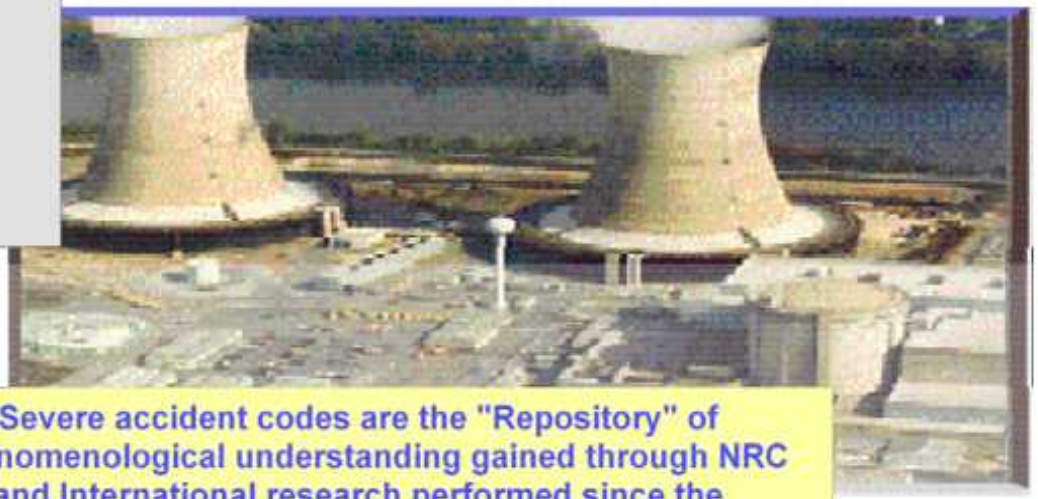
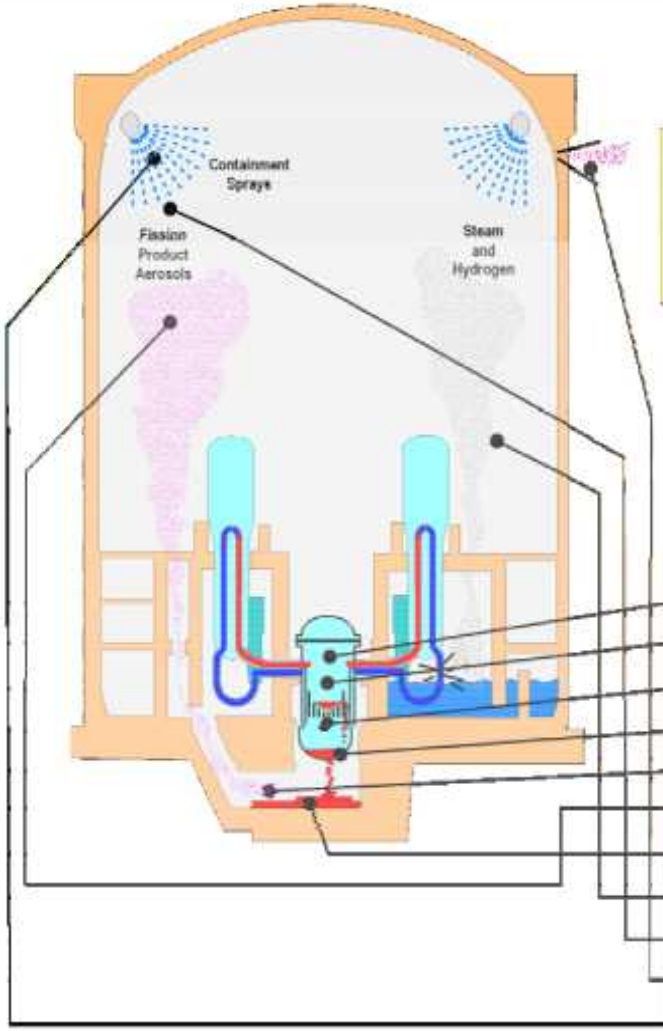
Modeling and Analysis of Severe Accidents in Nuclear Power Plants



Severe accident codes are the "Repository" of phenomenological understanding gained through NRC and International research performed since the TMI-2 accident in 1979

Integrated models required for self consistent analysis



Important Severe Accident Phenomena

	MELCOR	CONTAIN	VICTORIA	ICOMP RELAP 5
Accident initiation	■			■
Reactor coolant thermal hydraulics	■			■
Loss of core coolant	■			■
Core meltdown and fission product release	■		◢	■
Reactor vessel failure	■			■
Transport of fission products in RCS and Containment	■	■	■	
Fission product aerosol dynamics	■	■	■	
Molten core/basemat interactions	■	■		
Containment thermal hydraulics	■	■		
Fission product removal processes	■	■	■	
Release of fission products to environment	■	■	■	
Engineered safety systems - sprays, fan coolers, etc	■	■		
Iodine chemistry, and more	■			

NRC code developed by SNL

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In-country Technical Support

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Activities in U.S. Embassy in Tokyo

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- Initial emphasis was on estimating risk potential to returning U.S. citizens that were previously evacuated
- State of reactors and spent fuel pools was vague
  - Communication was slow and cautious (to a fault)
    - Information exchange not flowing freely
  - Confusion over damage condition was frustrating
    - Significant loss of public confidence in both industry and government
  - Accidents were not clearly terminated even weeks after initial events
    - Water flow, nuclear shutdown, secondary seismic, hydrogen concerns
  - Concerns over future secondary seismic events was large
    - Reactors damaged and vulnerable to subsequent earthquakes
- Daily consults with U.S. military (based at Yakota)
- Daily consultation and assistance to NRC and DOE
  - Severe accident interpretation
  - Source term estimation
  - Explanation of numbers and results
    - Bq/cc, Curies, Sieverts (micro, milli), REMs, numbers, powers of ten 10^{3}, 10^{9}, 10^{-6}
  - Exploring mitigation options and success
- Ongoing forensics investigation of events
  - Computer code reconstruction of accidents

NRC Team Led by Chuck Casto Supporting Ambassador John V. Roos

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Initial Fukushima Analyses

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Initial Fukushima Analyses

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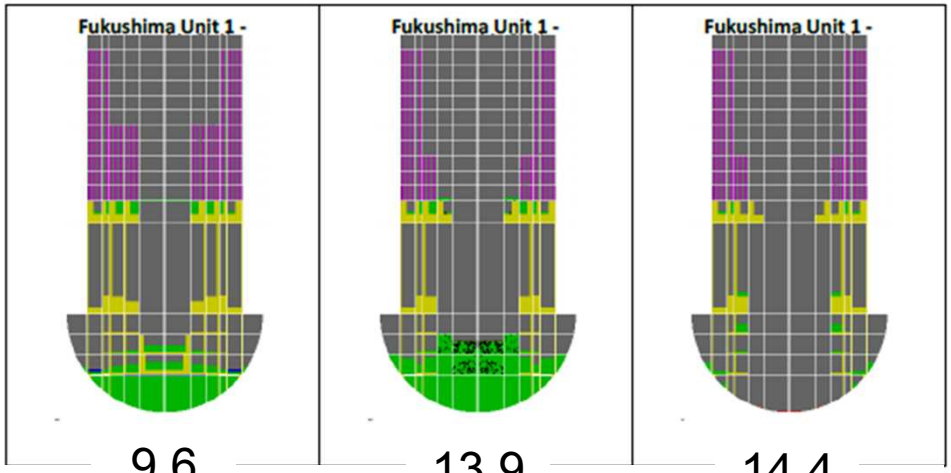
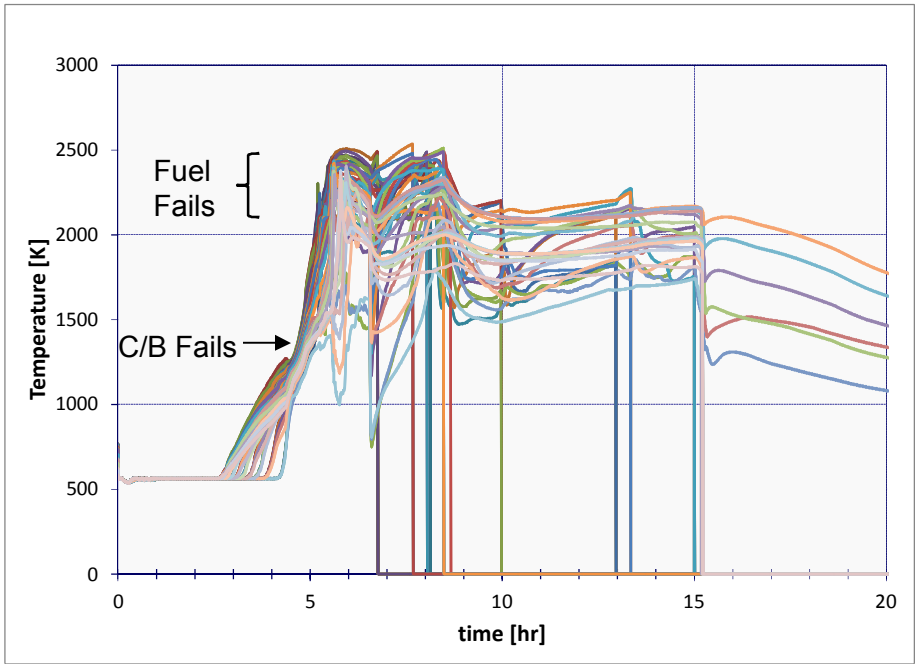
- SNL MELCOR Fukushima reactor models are based the Peach Bottom State-of-the-Art Reactor Consequence Analyses (SOARCA) model; reflects current MELCOR BWR Mk-I best practices
- Models were updated with the best-available Fukushima inputs (e.g., TEPCO December 2011 data set); developed surrogate inputs where necessary
- Created new models for 1F1, 1F2, and 1F3 reactors, and 1F4 spent fuel pool
- Results matched (sparse) T-H data in general
- "Tuning" would be needed to account for
  - potential RPV failure mechanism (SRV failure vs. MSL failure)
  - 1F2 torus room flooding
  - 1F2 and 1F3 wetwell mixing/stratification
  - 1F1 and 1F3 venting and injection operations

1F1 Core Temperatures and Damage State





- Core damage starts at ~ 4 hours – Control Blades fail first
- Core exit gas temperatures very high
- Lower head eventually fails (~14 hr)

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OECD/NEA Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF) Project – Phase 1

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OECD/NEA BSAF Project – Phase 1

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- Used models developed for initial Fukushima analyses
- Models were updated with the best-available Fukushima inputs (e.g., IEA November 2013 data set); developed surrogate inputs where necessary
- Revised decay heat/RN inventory input with results from SNL ORIGEN-S/ARP analyses
- Implemented BSAF-defined accident sequence boundary conditions (i.e., common case)
- Implemented SNL-defined best-estimate accident sequence boundary conditions (i.e., best estimate case)
- 1F1 and 1F3 BSAF cases completed
  - accident signatures look similar to previous results; those of other BSAF participants (with exceptions); and to most of the TEPCO data
  - event timings and values are different, but not markedly so

OECD/NEA BSAF Project – Phase 1

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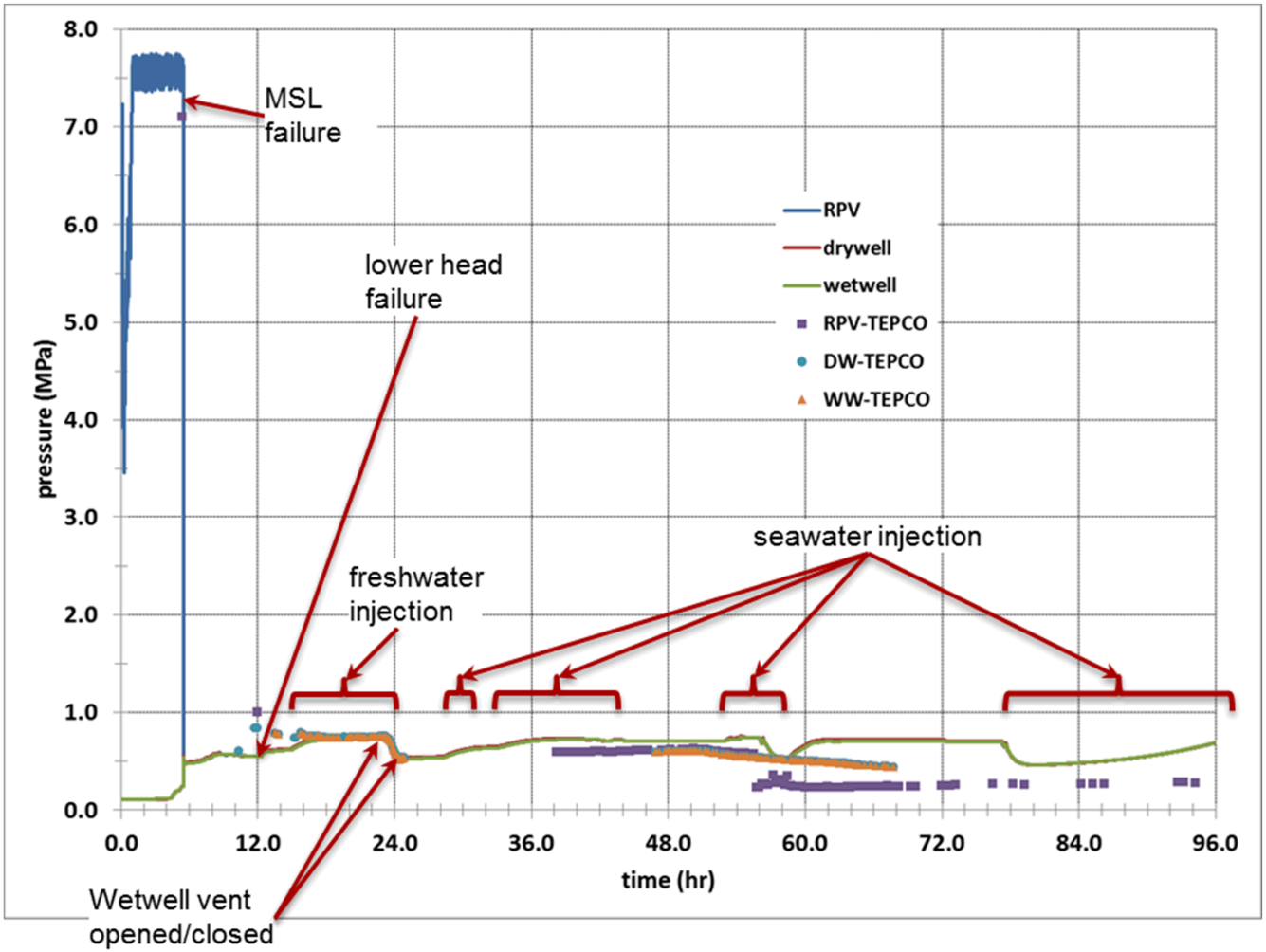
1F1 Best Estimate Case – RPV/DW/WW Pressures



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MELCOR/Modular Accident Analysis Program (MAAP) Crosswalk

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MELCOR/MAAP Crosswalk

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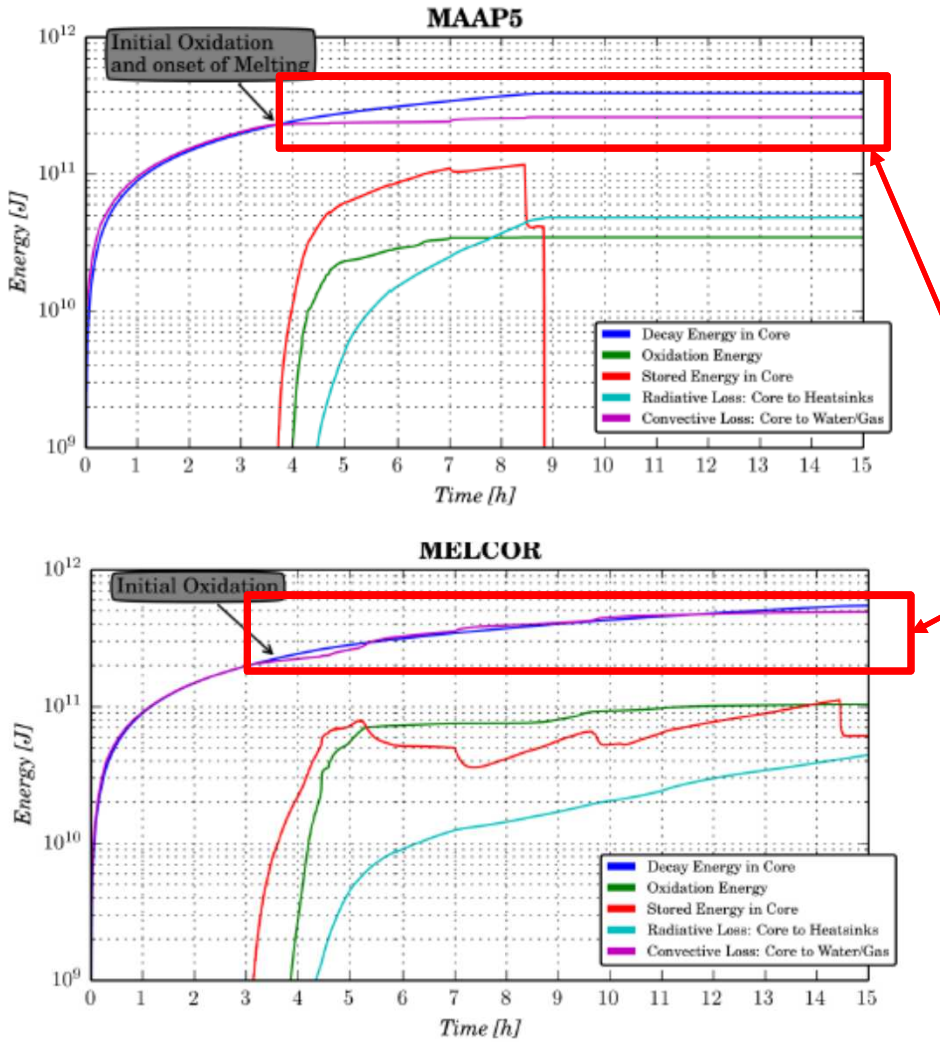
- Compare MAAP (severe accident code developed by FAI/EPRI) and MELCOR results to identify areas where the codes differ in their treatment of accident phenomena
  - Impetus for the work was differences found between MAAP and MELCOR ex-vessel materials results from the initial SNL and EPRI analyses
- Use Fukushima Daiichi Unit 1 (1F1) models to create results for comparison
- Focus on core melt progression
- Use comparison of results to identify where phenomena are treated/modeled differently
- Updated models to reflect latest available plant data and BSAF boundary conditions
- Modified models to minimize water and component inventory differences
- Ran models with a common accident sequence
- Developed a set of common results figures

MELCOR/MAAP Crosswalk



Key Differences

- MAAP core damage progression results in a molten pool
- MELCOR core damage progression results in particulate debris bed.
- MELCOR calculates a much larger amount of energy transferred from core materials to RPV water/gases than MAAP

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Upcoming Work

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Upcoming Work

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- BSAF Phase 2
  - Calculate and compare source term results with dose/isotopic data within the reactor containments, reactor buildings, and the environment
  - Will have to account for source term transport in plant and the environment
- Phase 2 Crosswalk
  - Perform a similar MAAP/MELCOR study evaluating a stylized TMI-2 accident
  - Preliminary discussions have been held with the Institute of Applied Energy (Japan) to perform a 1F1 crosswalk, as well as the proposed TMI-2 crosswalk
  - IRSN may also participate with their Accident Source Term Evaluation Code (ASTEC)

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Thank you for your attention.

Questions?

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