INL/MIS-25-84388-Revision-0

TMI presentation for Beyond Design Basis Workshop

May 2025

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Changing the World's Energy Future

Idaho National Laboratory

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DOE Modeling and Code Impact of TMI

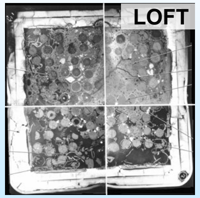
SCDAP/RELAP5

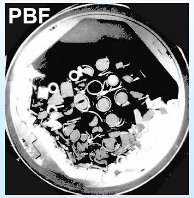
- SCDAP/RELAP5 computer code was designed to calculate the overall reactor coolant system (RCS) thermal-hydraulic response, core damage progression, and reactor vessel heat-up and damage
- This code was developed at Idaho National Laboratory (INL) under the sponsorship of the U.S. Nuclear Regulatory Commission (NRC)
- The TMI-2 accident has been used to support a peer review of the code, assess the quality of the results, and to benchmark other codes
- Several post TMI-2 experiments have supported code improvements
  - Loss of Fluid Tests (LOFT) – 38 tests of various breach sizes studied
  - Sandia Lower Head Failure Project
  - Melt Attack and Coolability Experiment (MACE)
  - CORA/QUENCH severe fuel damage experiments (NEA project)
  - RASPLAV Core melt experiments (NEA project)

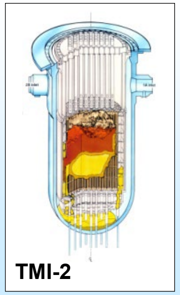
SCDAP/RELAP5-3D© Development

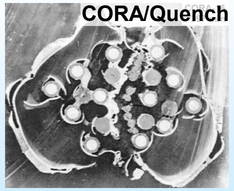
Model Development and Assessment Based on Data from:

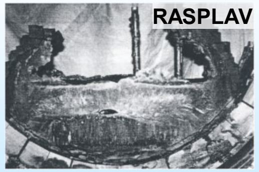
- DF/XR
- PHEBUS











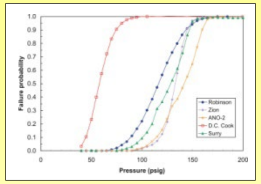
Experiments and Analyses

Model Development and Assessment

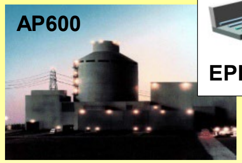
SCDAP/RELAP5-3D©

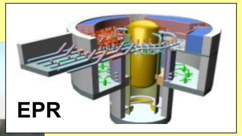
Applications

Severe Accident Resolution
(DCH, SGTR)

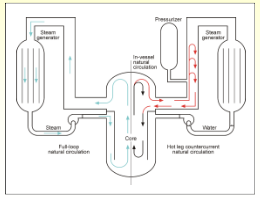


ALWR Evaluations
(AP600, APR 1400,EPR, SBWR)



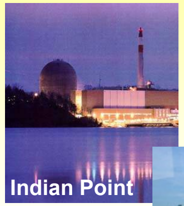


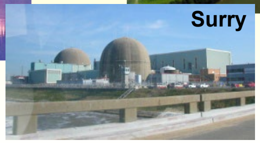
Severe Accident Mitigation Strategies
(Depressurization, Water Addition)



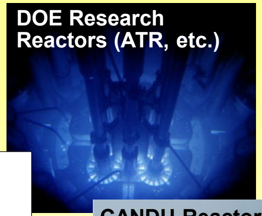
LWR

Existing LWRs

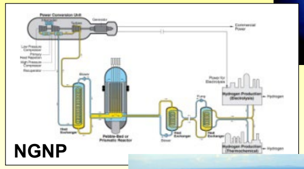




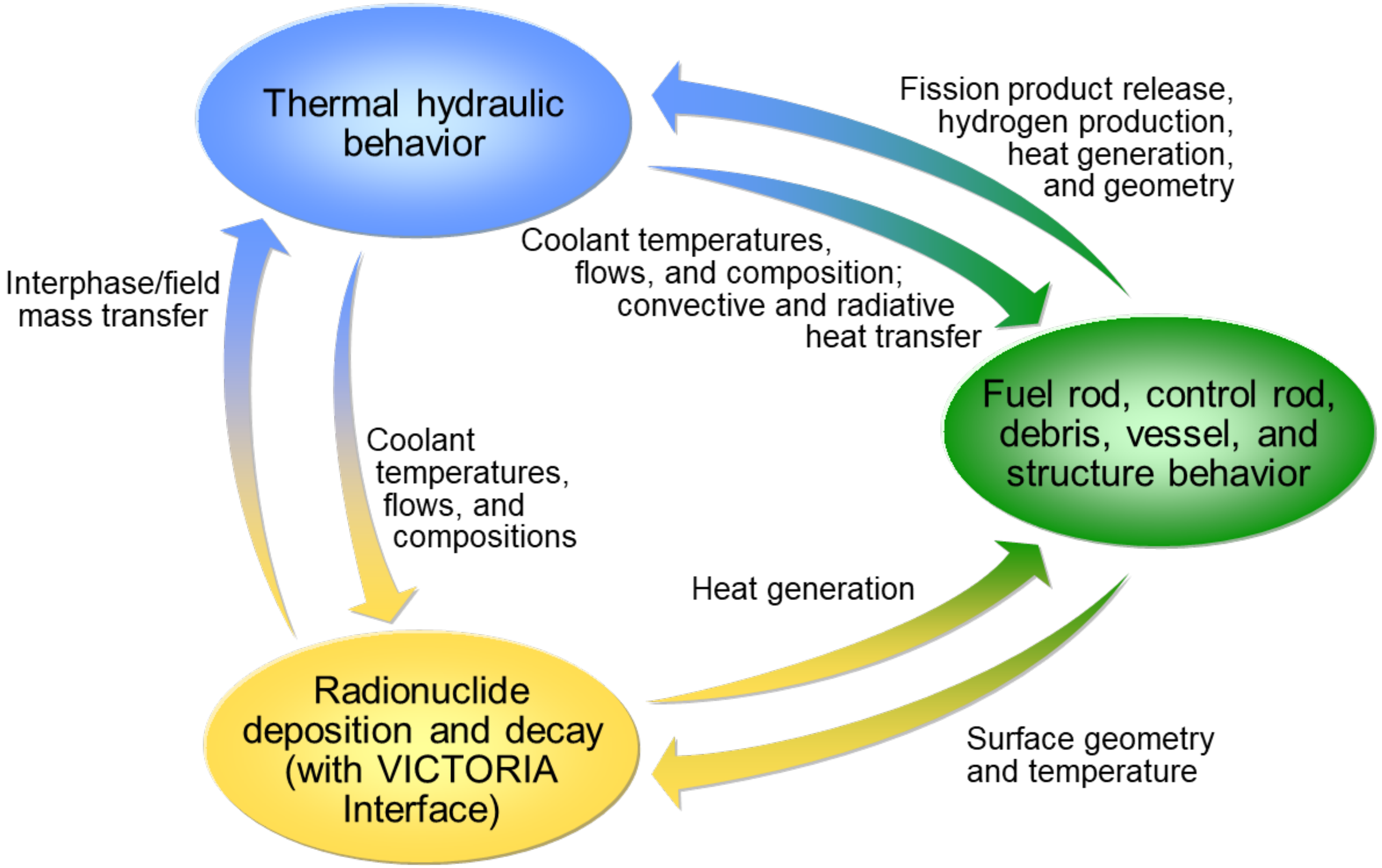
Non-LWR and MTRs



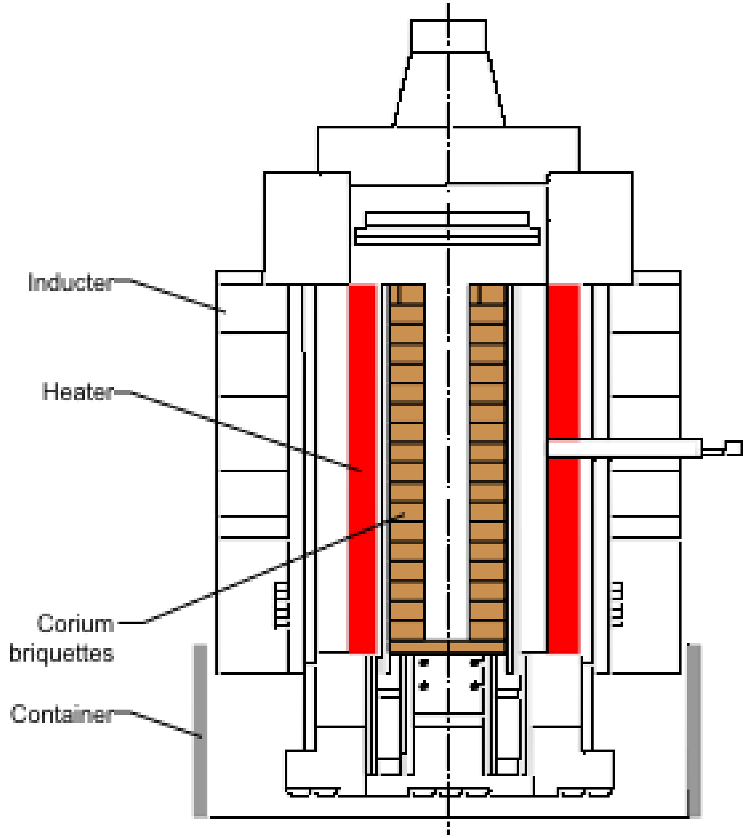
GENIV Reactors
(NGNP, etc.)

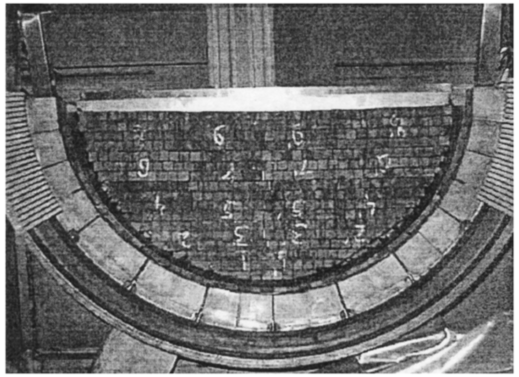


SCDAP/RELAP5

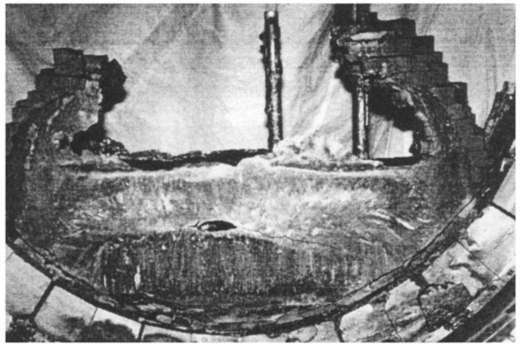


RASPLAV Provided Insights about Stratification in Relocated Molten Corium Materials





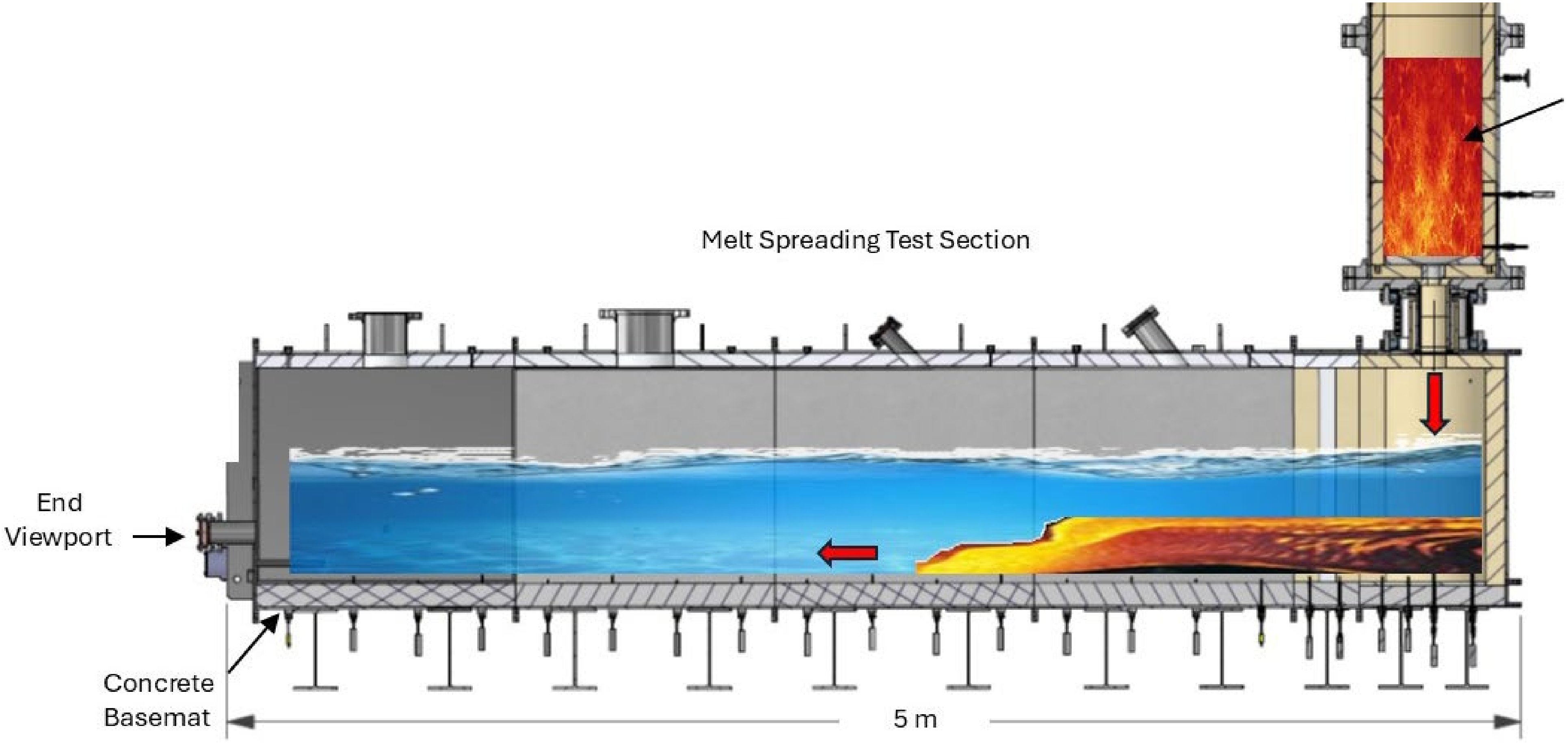
Before



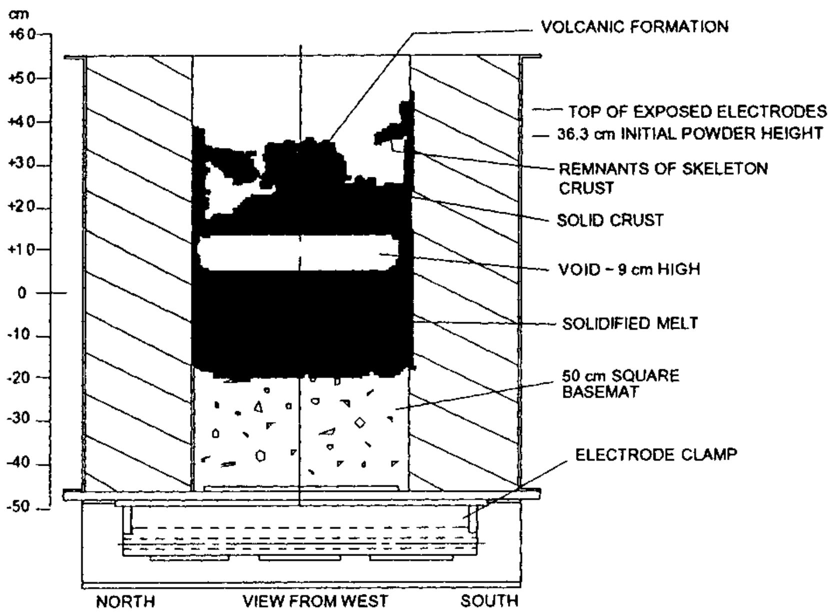
After

Stratification dependent on presence of carbon and fraction of unoxidized zirconium
(AW-200-2 used C-22 with 81.8 wt% UO_2, 5.0 wt% ZrO_2, 13.2 wt% Zr, and 0.3 wt% C)

Argonne National Laboratory (ANL) Reactor Severe Accident Test Facility



MACE Tests Provide Key MCCI Insights

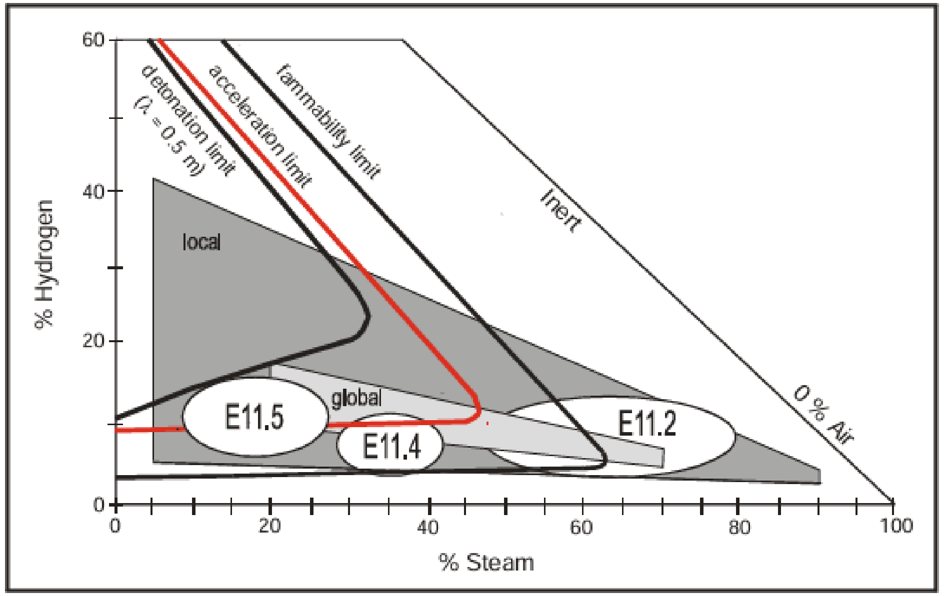


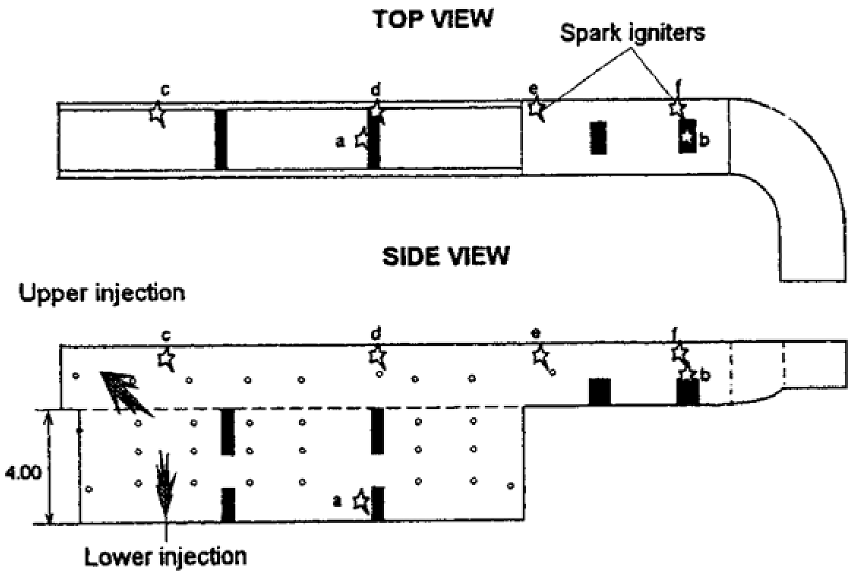
- Large scale, prototypic tests:
  - 100 to 2,000 kg (220 to 4,400 lbs) prototypic corium
  - 30 cm x 30 cm to 120 cm x 120 cm (1 ft x 1 ft to 4 ft x 4 ft) concrete basemat area
  - UO_2, ZrO_2, and Zr corium materials heated up to 2350 K (3770 °F)
  - Electrodes to simulate decay heat
  - Water added after corium melts

- Observed:
  - High initial heat transfer from corium
  - Significantly lower heat removal after crust forms on upper surface
  - Voiding in corium region beneath crust
  - Pool swelling followed by eruptions enhances heat removal

Post TMI-2 Research into Hydrogen Detonation and Control

- Several organizations conducted hydrogen detonation and hydrogen control experiments
  - Russian “Kurchatov Institute” RUT facility
  - Brookhaven National Laboratory High-Temperature Combustion Facility
  - German FZK facility
    - Research data was incorporated into the GASFLOW code which can be coupled with MELCOR

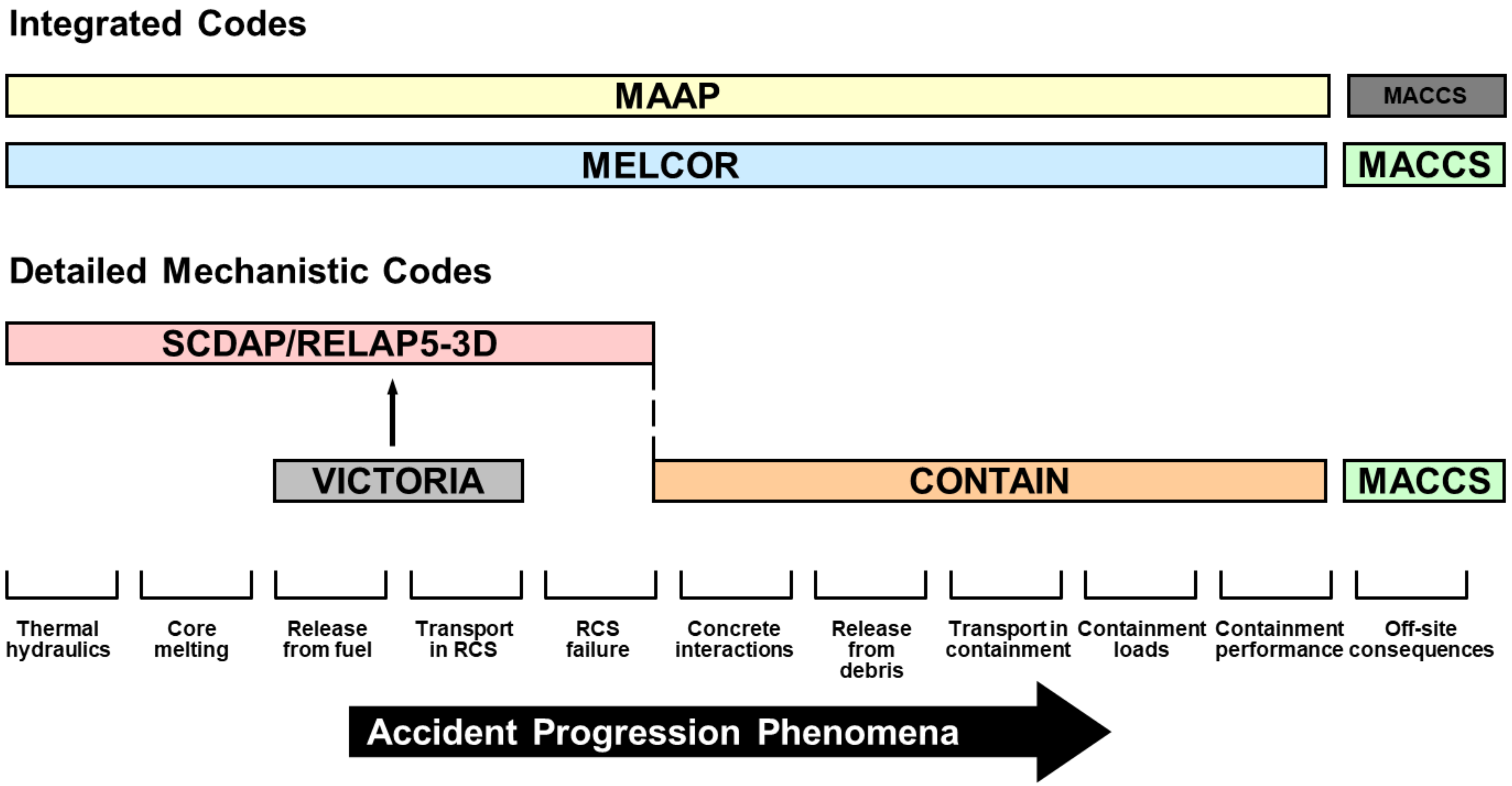




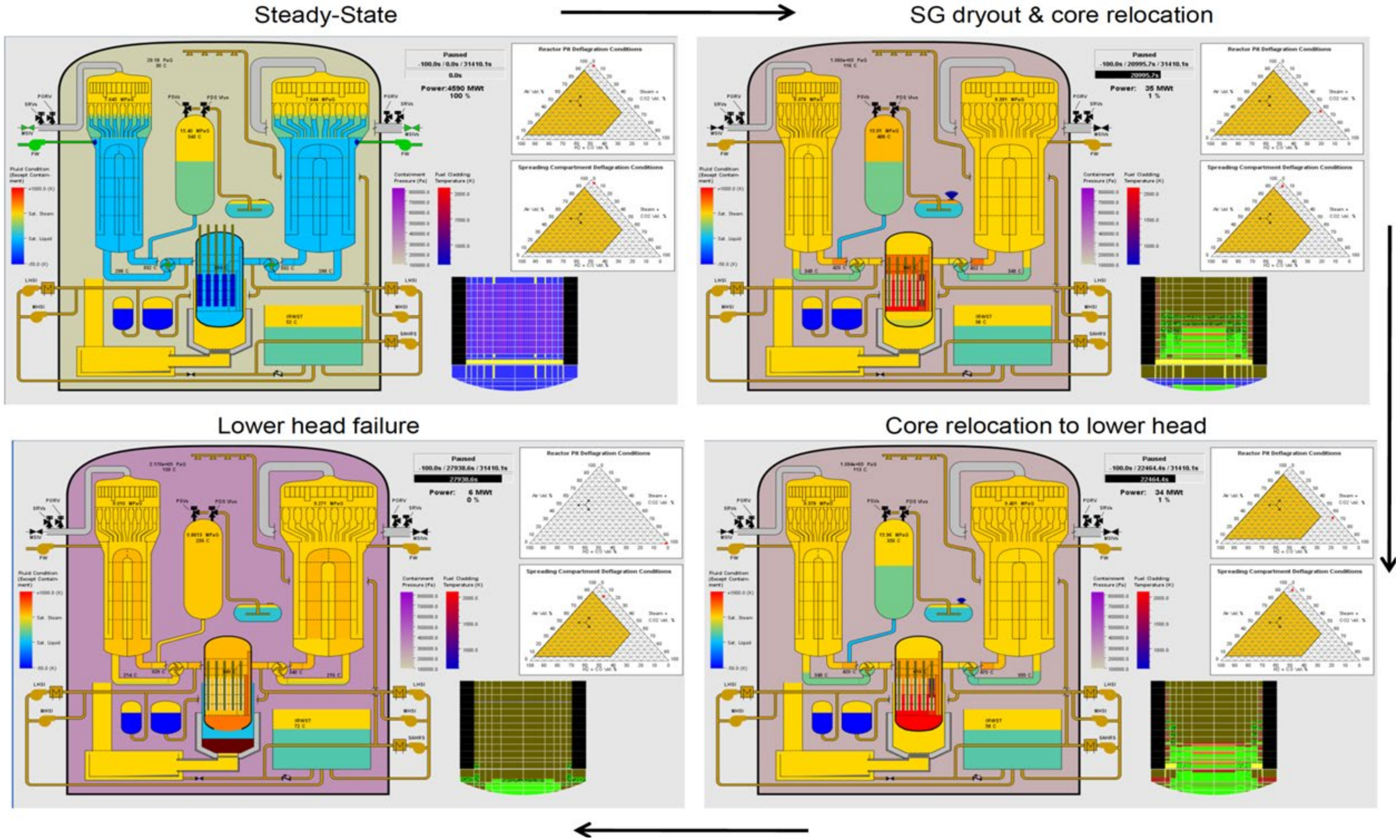
MELCOR

- MELCOR is a fully integrated, engineering-level computer code designed to analyze severe accidents in nuclear power plants
- MELCOR was created at Sandia National Laboratories (SNL) for the U.S. Nuclear Regulatory Commission (NRC)
- Development of MELCOR was motivated by WASH-1400 following the TMI accident
- Development of MELCOR began in 1982 and continues today to address emerging issues, incorporate new experimental information and create a repository of knowledge on severe accident phenomena

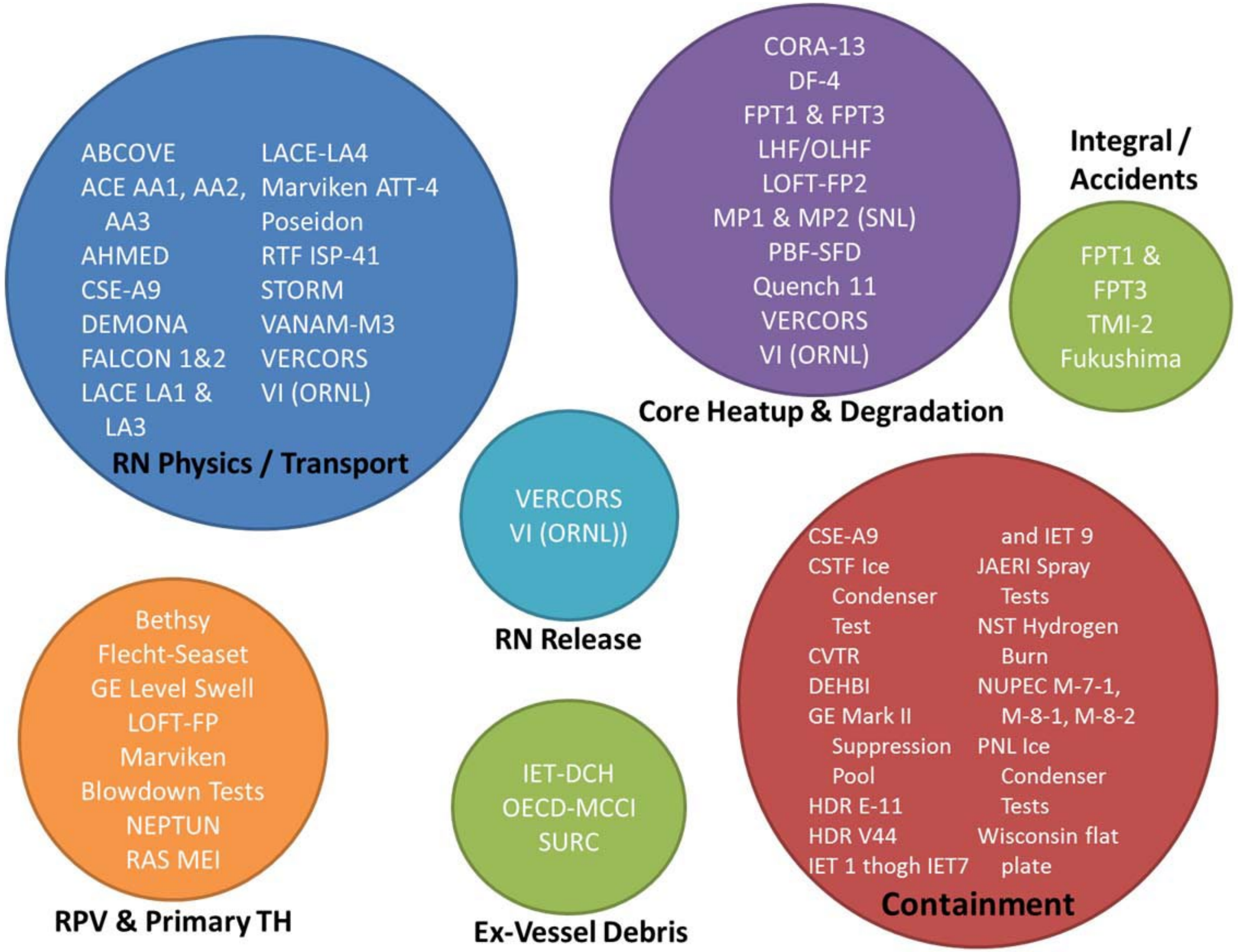
Severe Accident Phenomena modeled by codes



MELCOR User Interface



Experiments/Accidents used for MELCOR validation



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