



TMI presentation for Beyond Design Basis Workshop

May 2025

Changing the World's Energy Future

Shawn W St Germain



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Shawn St. Germain

Manager: Reliability,
Risk and Resilience
Sciences

DOE Modeling and Code Impact of TMI

Battelle Energy Alliance manages INL for the
U.S. Department of Energy's Office of Nuclear Energy



Idaho National Laboratory

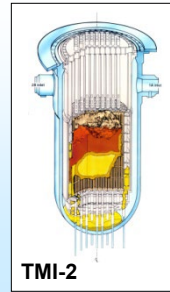
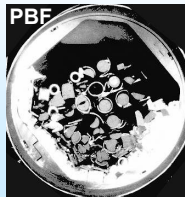
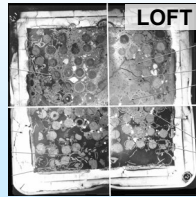
SCDAP/RELAP5

- 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



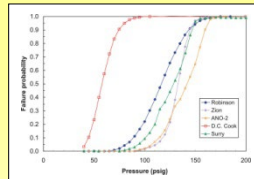
Experiments
and Analyses

Model
Development
and Assessment

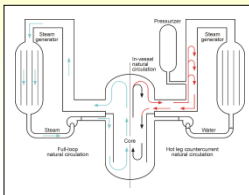
SCDAP/RELAP5-3D[®]

Applications

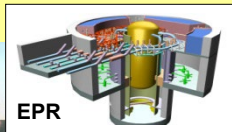
Severe Accident Resolution
(DCH, SGTR)



Severe Accident Mitigation Strategies
(Depressurization, Water Addition)

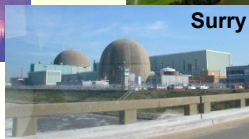
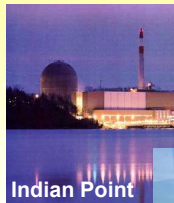


ALWR Evaluations
(AP600, APR 1400, EPR, SBWR)



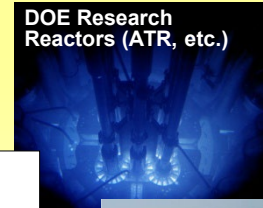
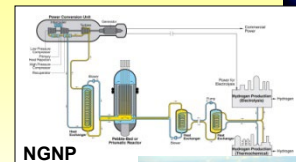
LWR

Existing LWRs

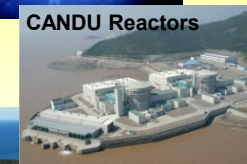


Non-LWR and MTRs

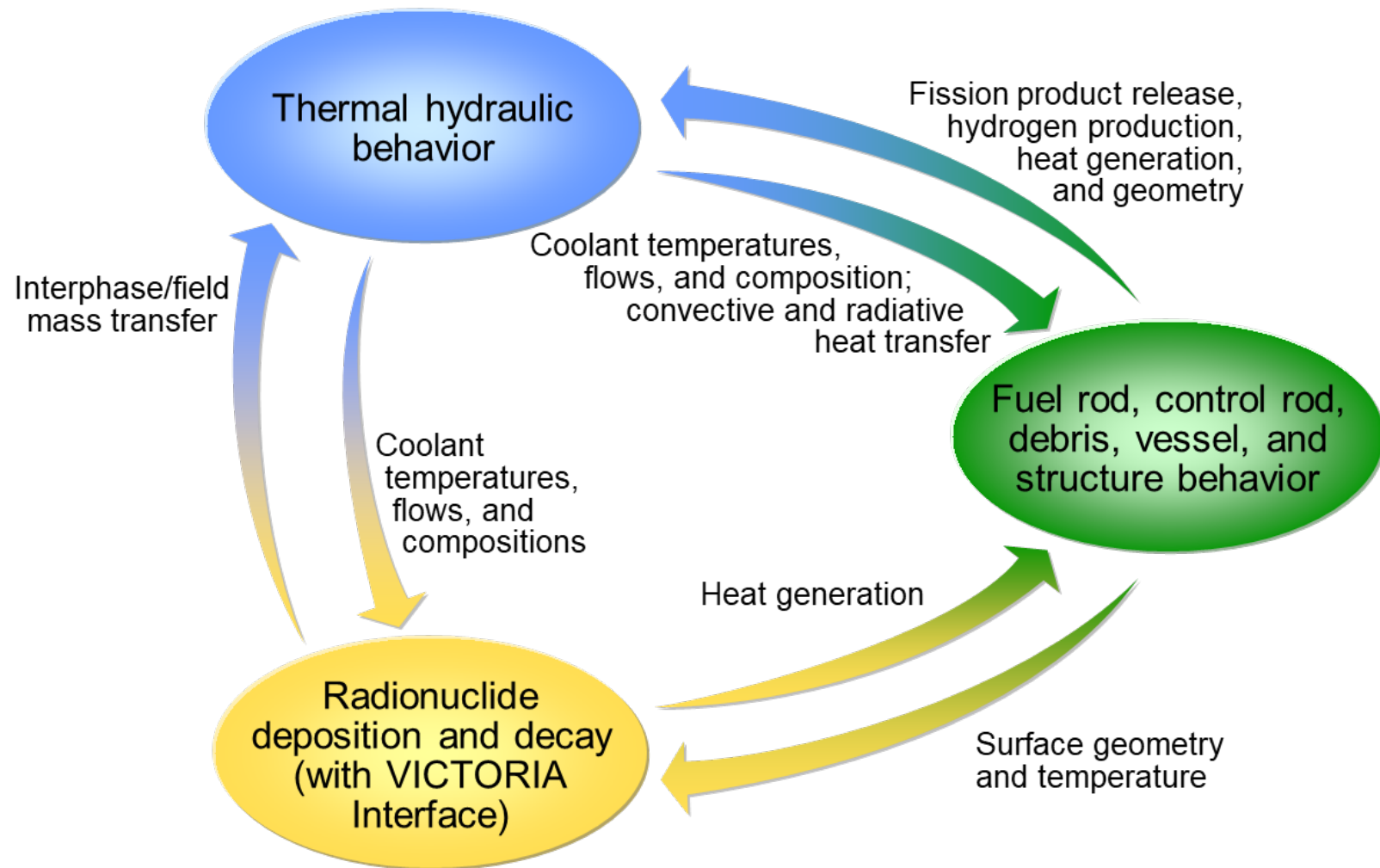
GENIV Reactors
(NGNP, etc.)



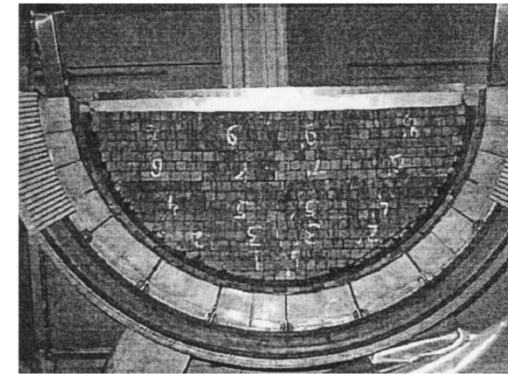
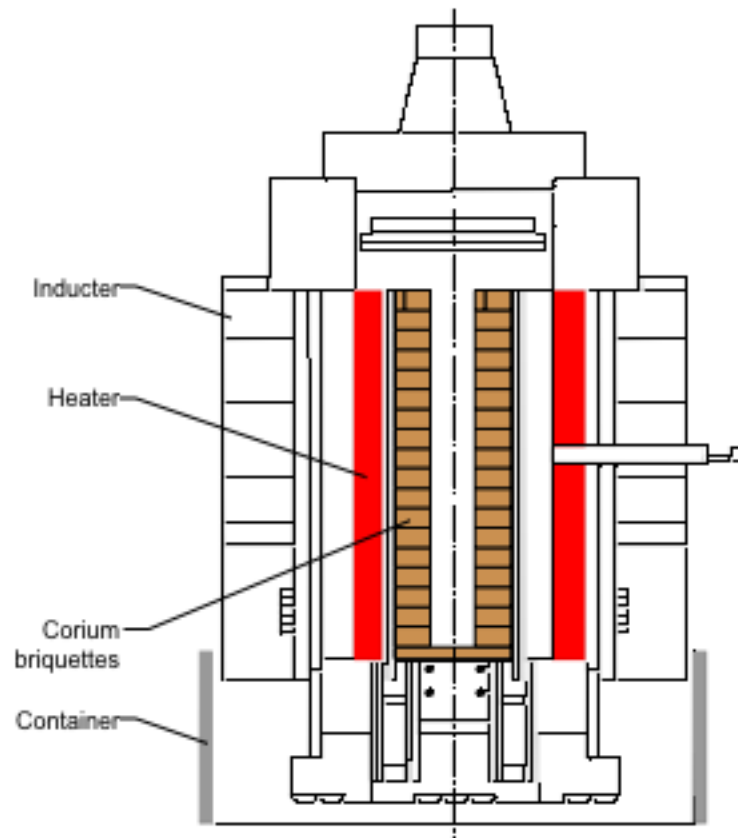
CANDU Reactors



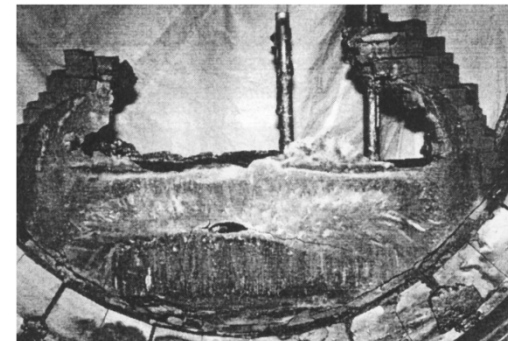
SCDAP/RELAP5



RASPLAV Provided Insights about Stratification in Relocated Molten Corium Materials



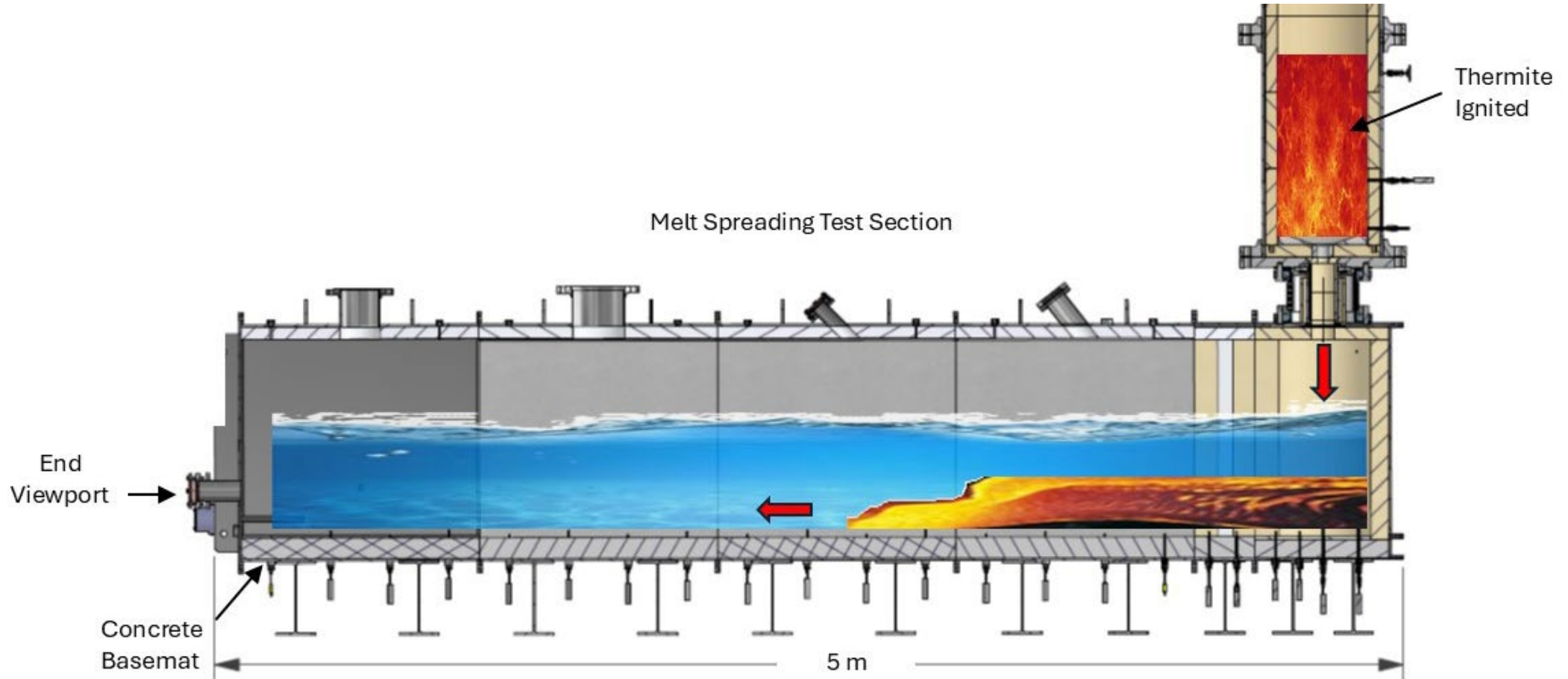
Before



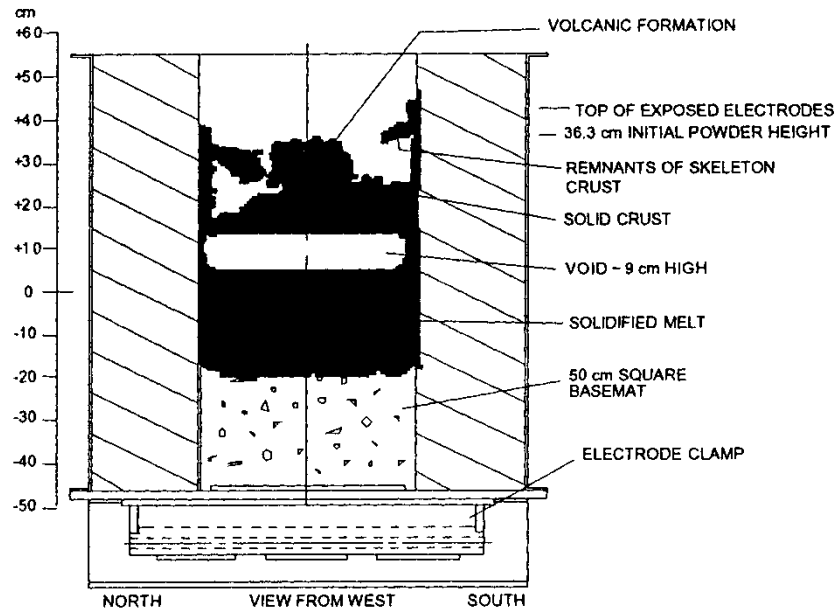
After

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

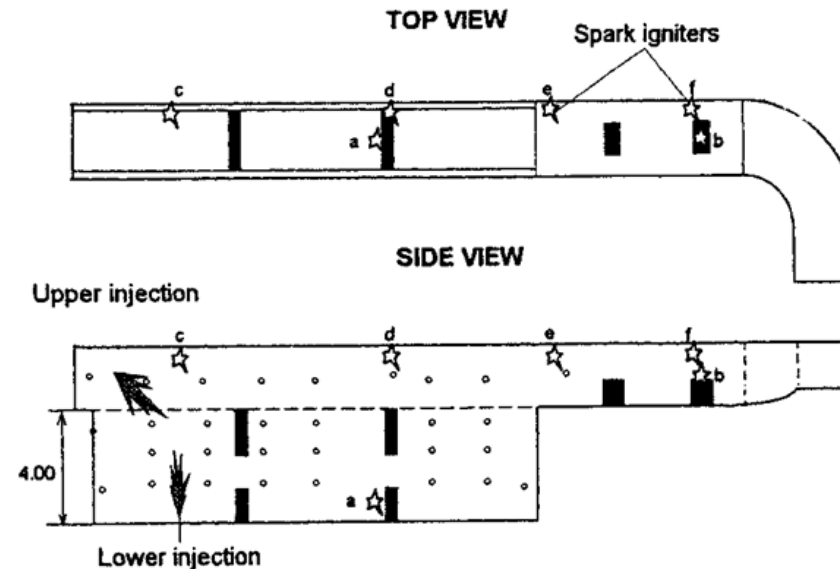
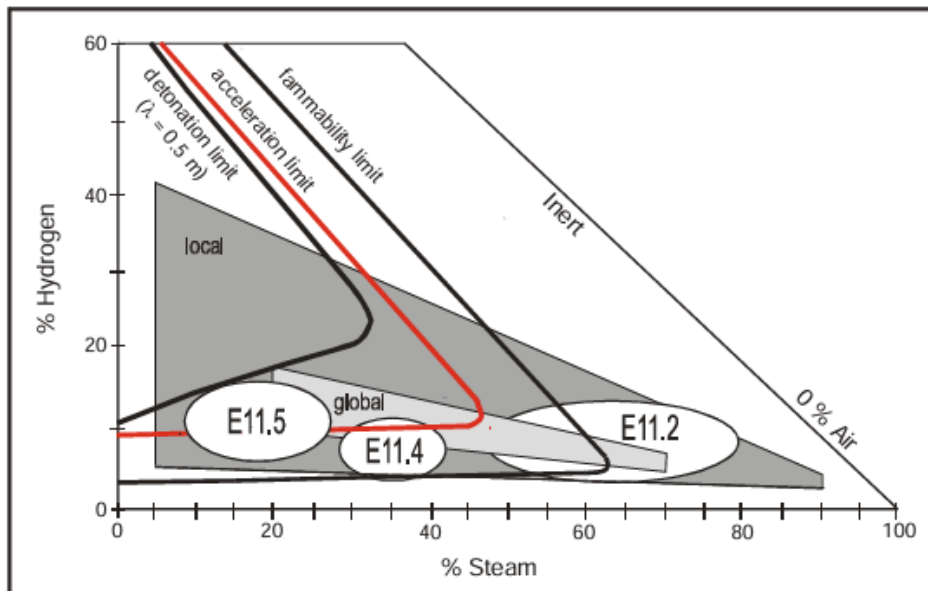


- Large scale, prototypic tests:
 - 100 to 2,000 kg (220 to 4,400 lbs) prototypic corium
 - 30 cm x 30 cm to 120 cm x 120 cm (1 ft x 1 ft to 4 ft x 4 ft) concrete basemat area
 - UO_2 , ZrO_2 , and Zr corium materials heated up to 2350 K (3770 °F)
 - Electrodes to simulate decay heat
 - Water added after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



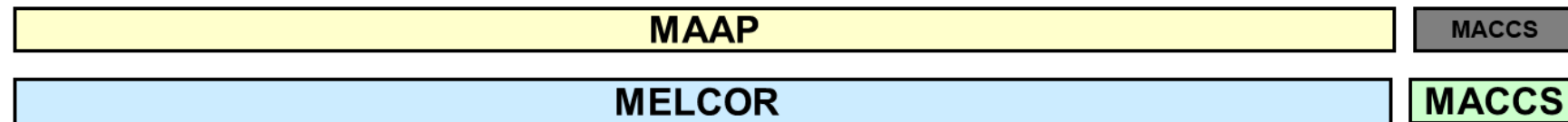


MELCOR

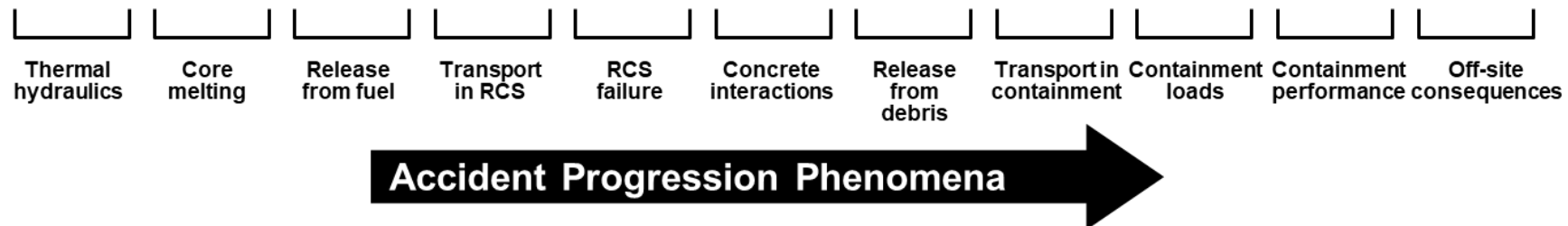
- MELCOR is a fully integrated, engineering-level computer code designed to analyze severe accidents in nuclear power plants
- MELCOR was created at Sandia National Laboratories (SNL) for the U.S. Nuclear Regulatory Commission (NRC)
- Development of MELCOR was motivated by WASH-1400 following the TMI 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

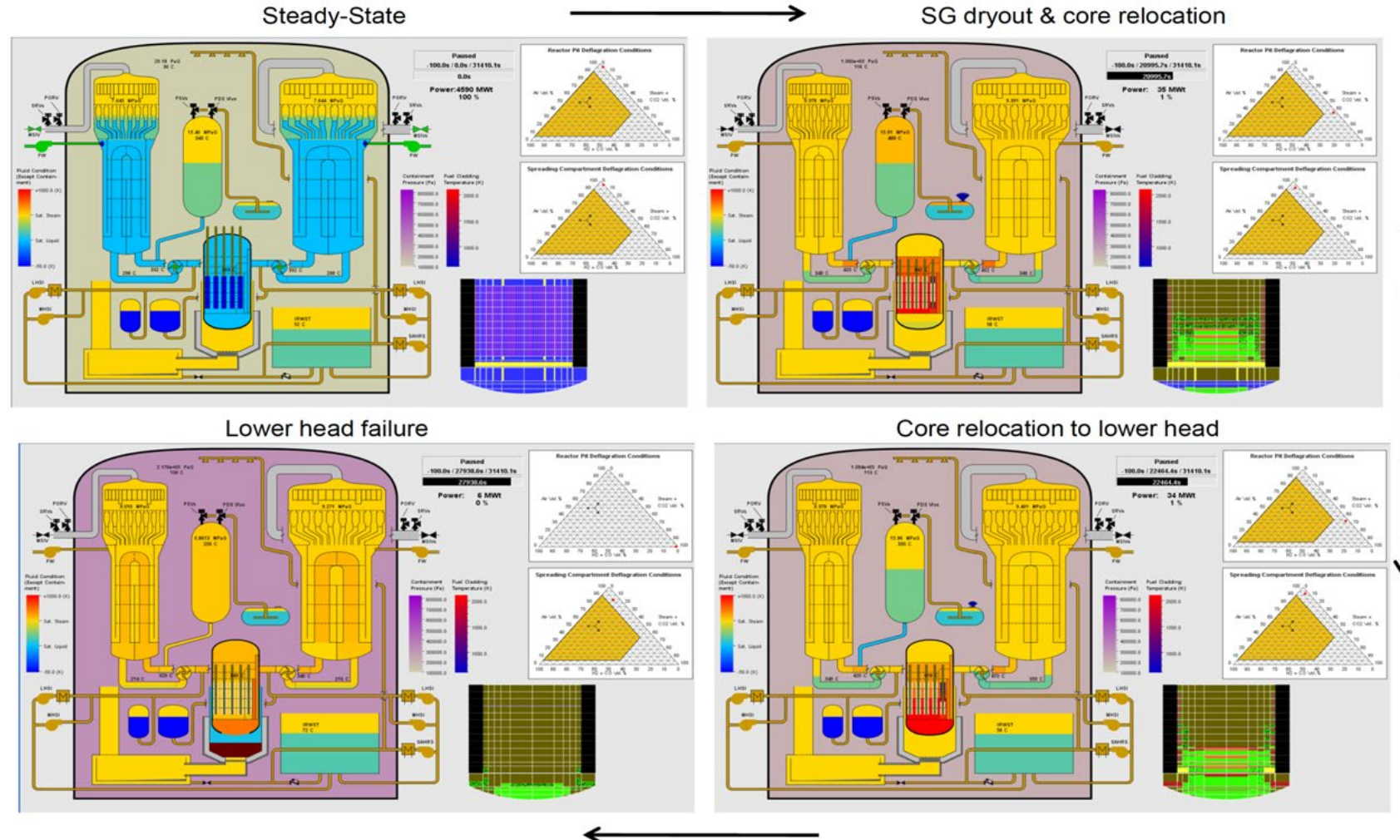
Integrated Codes



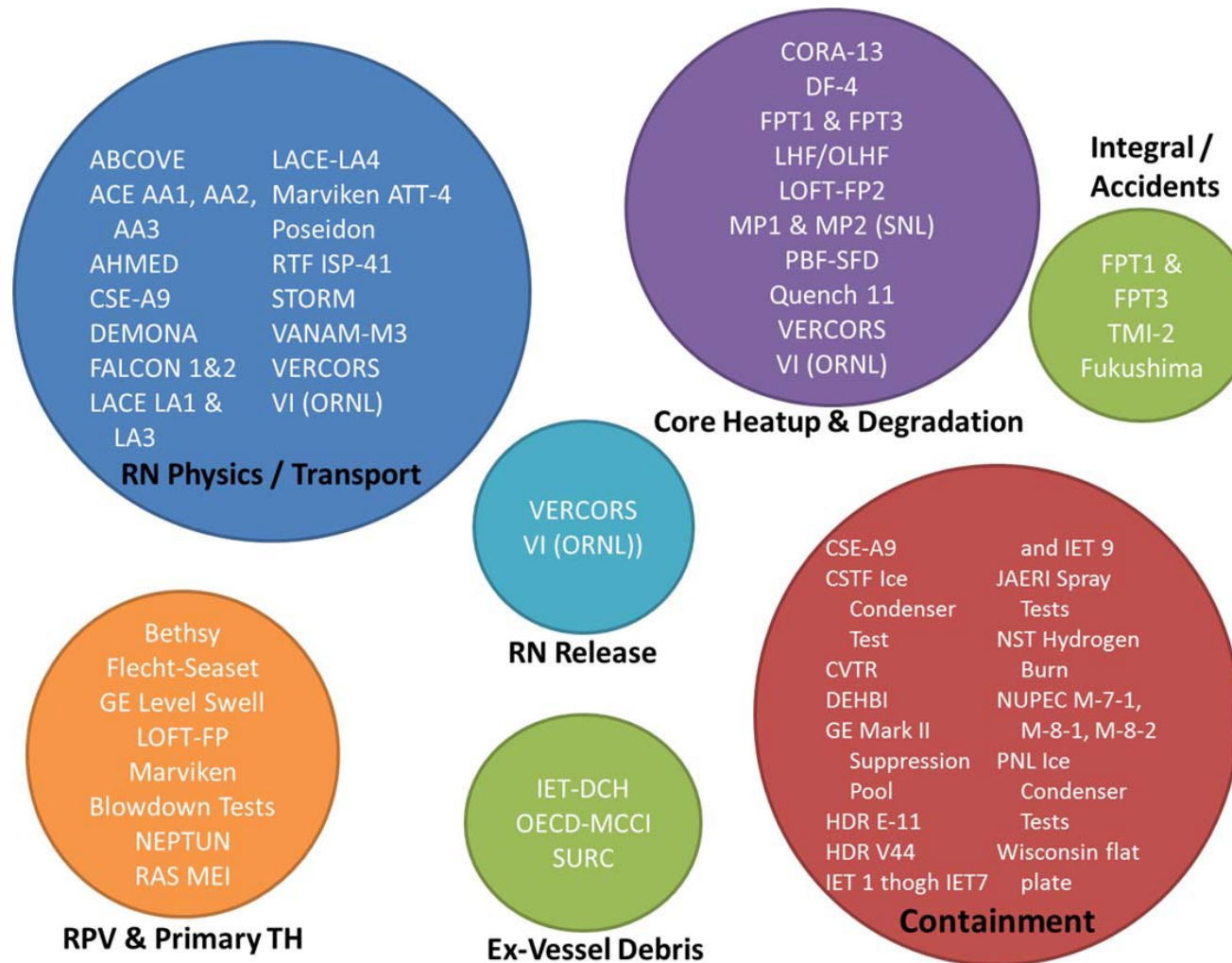
Detailed Mechanistic Codes



MELCOR User Interface



Experiments/Accidents used for MELCOR validation





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