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Protecting People and the Environment

NRC SEVERE ACCIDENT & MELCOR ACTIVITIES

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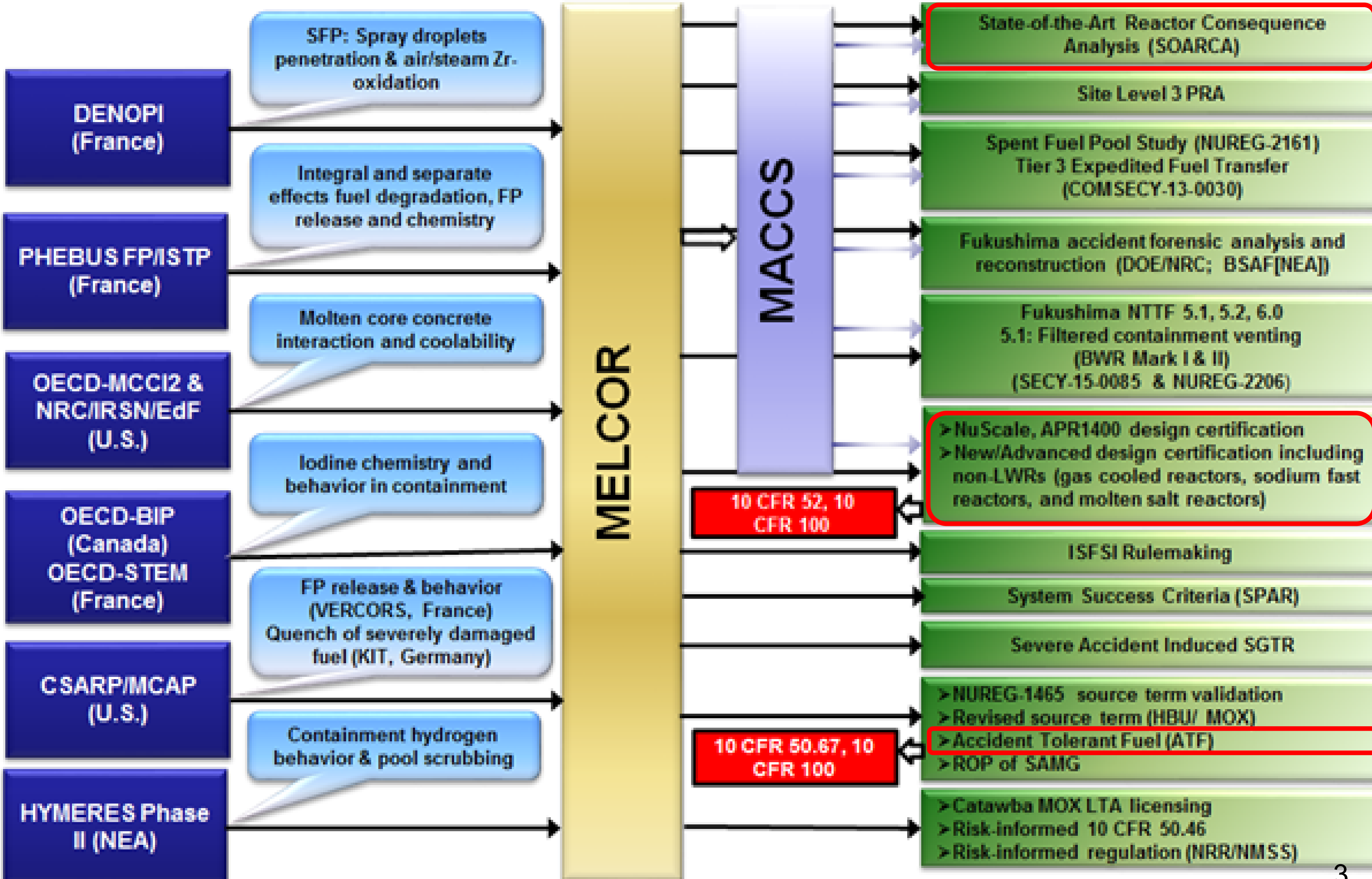
Zagreb, Croatia

Severe Accident Research Activities

- Support Risk-informing Regulations and Address Operating Reactor Issues and New Reactor Design Certification & Licensing (e.g., NuScale, ATF)
  - Maintenance of expertise of severe accident phenomenological knowledge and validated analytical tools
- International Collaboration
  - U.S. NRC Cooperative Severe Accident Research Program (CSARP)
  - Annual MELCOR Meetings
    - MELCOR Code Assessment Program (MCAP) - (Spring/USA)
    - European MELCOR User Group (EMUG) – (Spring/Europe)
    - Asian MELCOR User Group (AMUG) – (Fall/Asia)
  - NEA/CSNI and European Commission

Code Development & Regulatory Applications



Non-LWR Licensing

- Strategy 1: Acquire/develop sufficient knowledge, technical skills, and capacity to perform non-LWR regulatory reviews
- Strategy 2: Acquire/develop sufficient computer codes and tools to perform non-LWR regulatory reviews
- Strategy 3: Establish a more flexible, risk-informed, performance-based, non-LWR regulatory review process within the bounds of existing regulations, including the use of conceptual design reviews and staged-review processes
- Strategy 4: Facilitate industry codes and standards needed to support the non-LWR life cycle (including fuels and materials)
- Strategy 5: Identify and resolve technology-inclusive policy issues that impact the regulatory reviews, siting, permitting, and/or licensing of non-LWR nuclear power plants (NPPs)
- Strategy 6: Develop and implement a structured, integrated strategy to communicate with internal and external stakeholders having interests in non-LWR technologies

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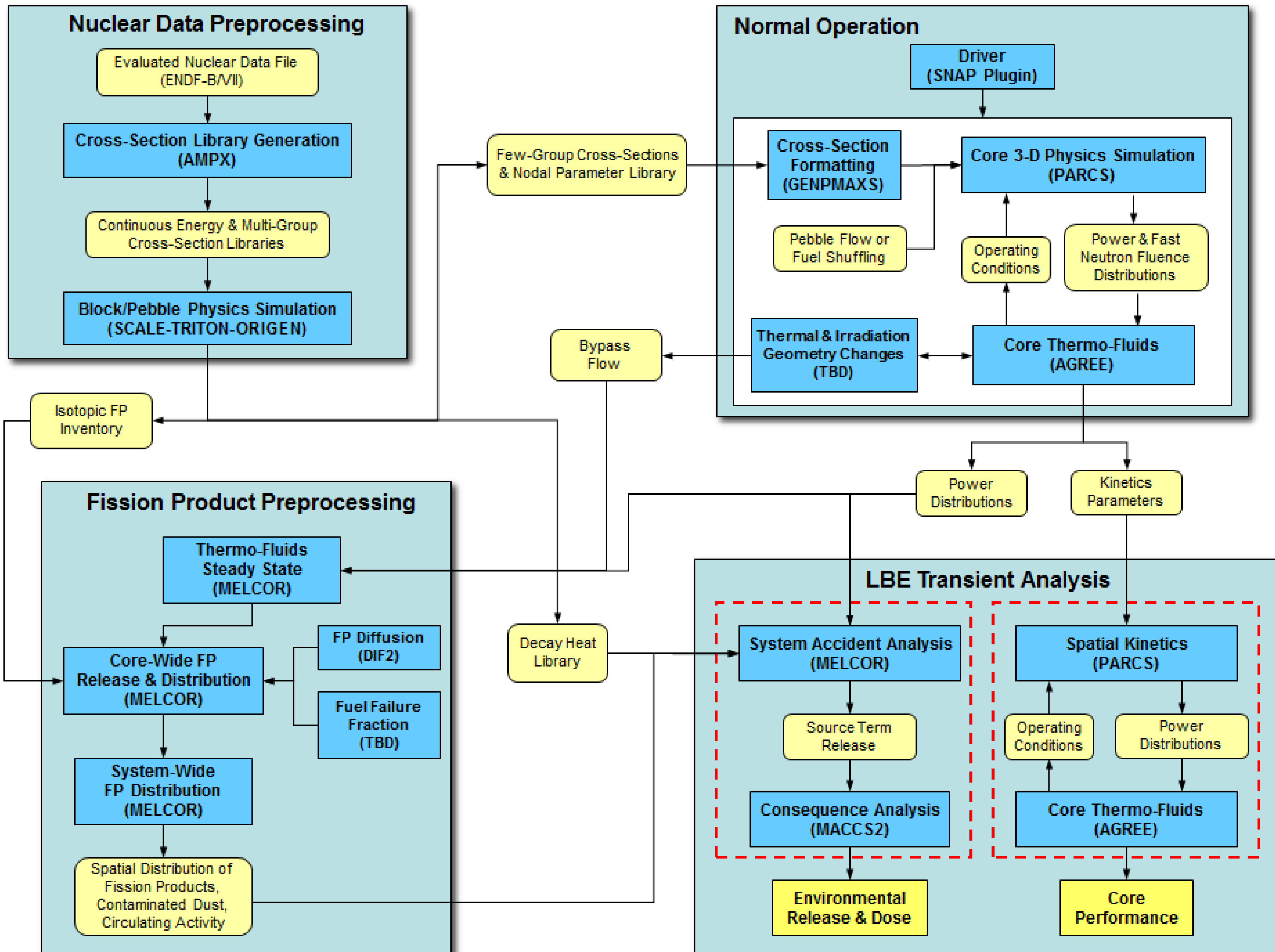
Non-LWR Technologies

Developer	Design	Power	Technology
Oklo Inc.	Oklo	~ 7 MWt	Compact fast reactor
Transatomic power	Transatomic	Small scale	Molten Salt Reactor
Terrestrial Energy	Integral molten salt reactor	400 MWt	Molten Salt Reactor
X-Energy	Xe-100	200 MWt	Modular High Temperature Gas Cooled
Terrapower	Molten chloride fast reactor (MCFR(	~2000 MWt	Molten Salt Reactor

Non-LWR Beyond Design Basis Events

- Development of evaluation models (example HTGR)
  - ACRS Future Plant Designs Subcommittee, April 5, 2011



ATF Design Concepts

- 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

Project plant available in NRC ADAMS (ML17325B771) at
https://www.nrc.gov/docs/ML1732/ML17325B771.html

ATF Regulatory Basis

Design-basis accident source term calculations are used to establish the adequacy of siting for commercial nuclear power plants and to ensure that adequate radiation protection exists for the control room and technical support center.

10 CFR 50.67 "*Accident source term*" requires the evaluation of the consequences of applicable design basis accidents & 100.11 "*Determination of exclusion area, low population zone, and population center distance*" requires fission product release values when evaluating the site. Both regulations state that:

The fission product release assumed for these calculations should be based upon a **major accident** hypothesized for purposes of site analysis or **postulated from consideration of possible accidental events**, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have **generally been assumed to result in substantial meltdown of the core** with subsequent release of appreciable quantities of fission products.

The "in-containment" source term is used in the analysis of a defense-in-depth measure to assess the adequacy of reactor containments and engineered safety systems.

Existing Licensing Basis

- Most Operating Reactor use Source Term in TID-14844
  - Same for both PWR & BWR
  - Based on heating irradiated pellets in a furnace
  - Instantly available to containment
- Alternate Source Term Available with RG 1.183 (NUREG-1465)
  - Series of mechanistic codes linked together (STCP)
  - Based on NUREG-1150 (*Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*) and research done following Three Mile Island accident
  - Distinct releases for PWRs and BWRs
  - Chemical categorization of radionuclides
  - Table of Release Fraction (RF) and timing for each phase and chemical group
    - Four phases of release and release timing
    - First two phases used for AST and the regulatory process

Applicants can use these "pre approved" source terms for siting calculations instead of developing and justifying their own

Design Basis Source Term Development Process

(example: MOX & High Burnup Fuel)

Experimental Basis ← PIRT process

- Oxidation/Gas Generation
- Melt Progression
- Fission Product Release
- Fission Product Transport

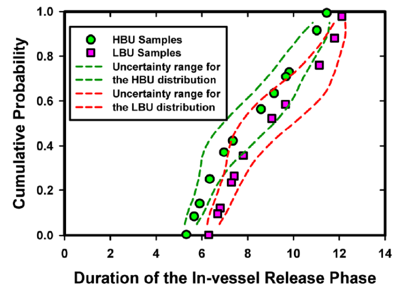
MELCOR

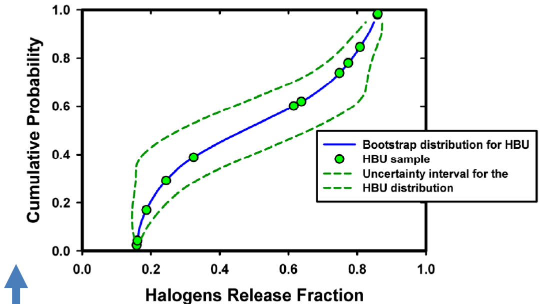
Accident Analysis

Scenario # 1	Scenario # 2
....	....
Scenario # n-1	Scenario # n

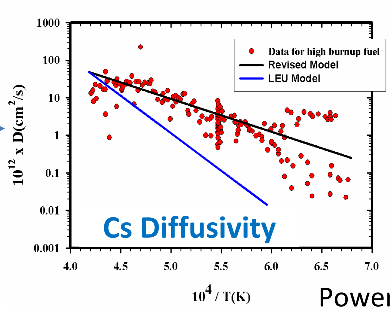
Synthesize timings and release fractions

Design Basis Source Term





Cesium release from high burnup fuel
Comparison to results of VERCORS Test



- Similar RFs to NUREG-1465 but prolonged release
- Differences not from change of fuel but from code advances

Fission Product class	Gap 0.5 hr	In-vessel 1.3 hr	Ex-vessel 2 hr	Late 10 hr
Noble Gases	5%	95%	~0	~0
Iodine, bromine	5%	35%	25%	1%
Cesium	5%	25%	35%	1%
Tellurium	~0	5%	25%	0.5%
Ba, Sr	~0	2%	10%	~0
Ru, Mo, Pd, etc.	~0	0.25%	0.25%	~0
Lanthanides	~0	0.02%	0.5%	~0
Cerium group	~0	0.05%	0.5%	~0

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Powers, et al. “Accident Source Terms for Light Water Nuclear Power Plants Using High-Burnup or MOX Fuel”, SAND2011-0128 January 2011

ATF Severe Accident Summary

- Experimental data (clad oxidation, RN release, core degradation) needed to modify MELCOR for different fuels and to synthesize a revised design basis source term
- Preliminary assessment of expected changes for ATF clad/fuel combination
- Revised design basis source term for ATF needs to be used (e.g., RADTRAD) to ascertain that approved TS changes comply with regulation (e.g., 10 CFR 50.67) for NPPs which plan to use ATF
- Other uses (e.g., emergency planning, incident response center) of ATF needs to be assessed

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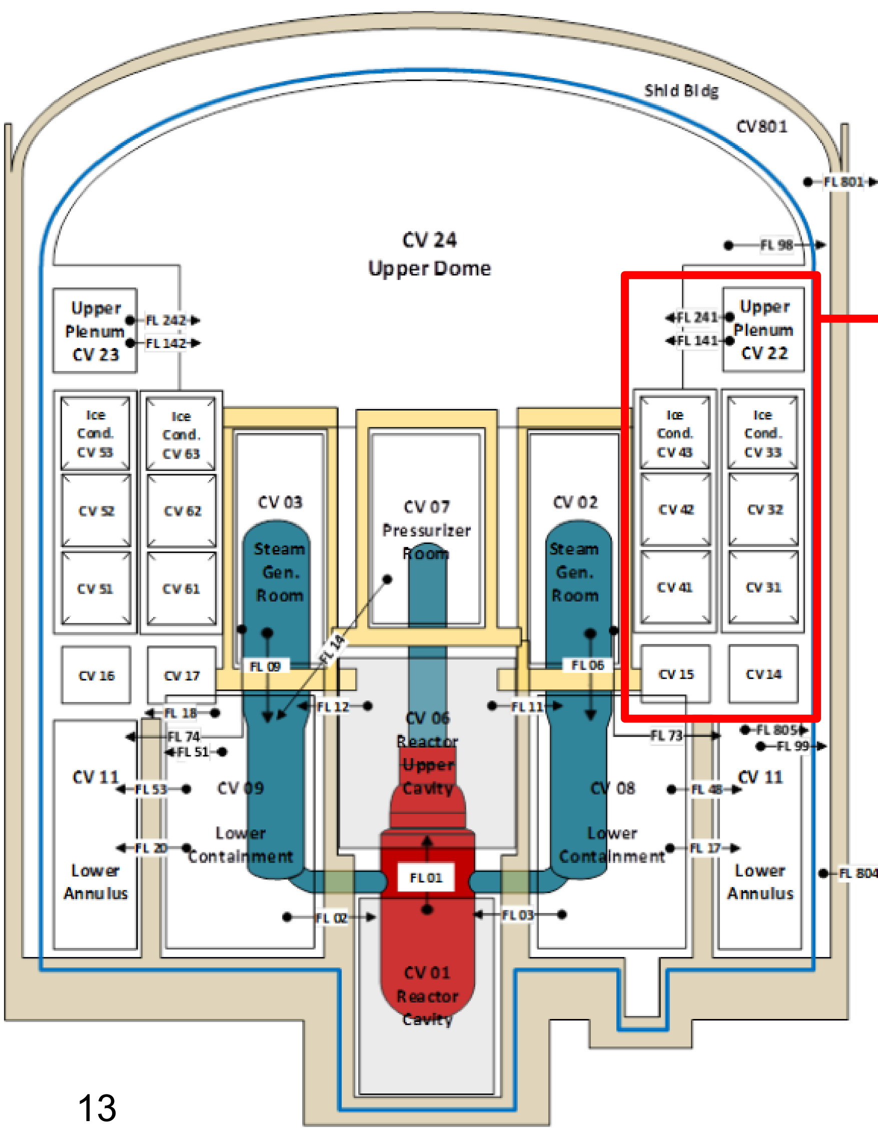
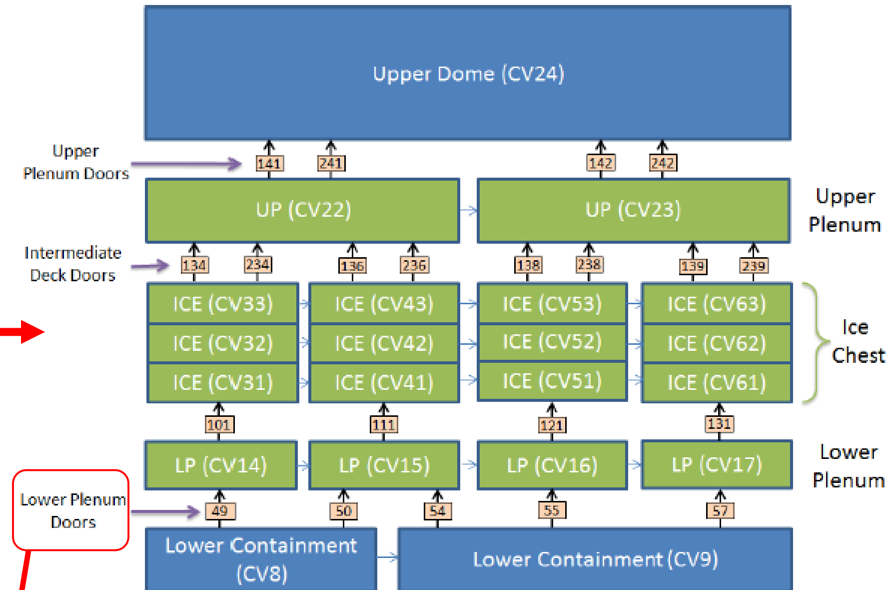
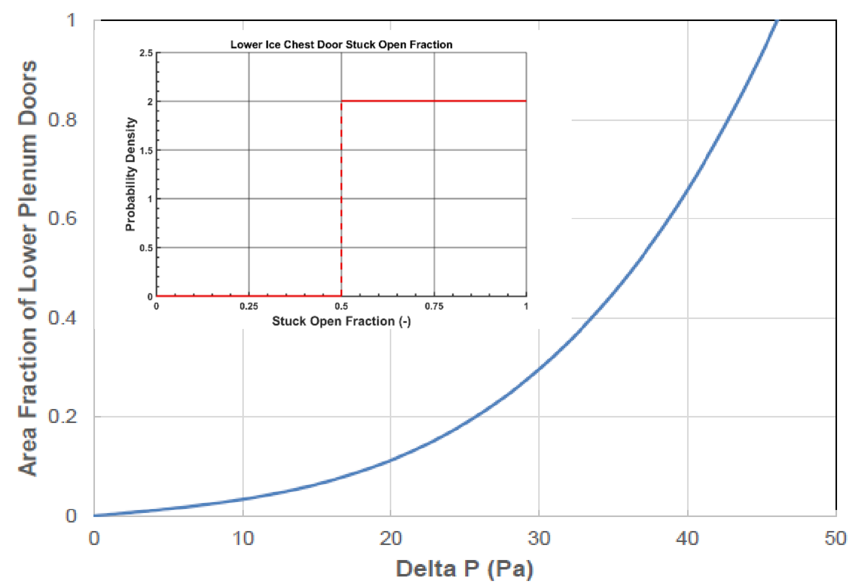
Sequoyah SOARCA Approach

- SOARCA goals/objectives:
  - Develop body of knowledge on the realistic outcomes of severe reactor accidents
  - Incorporate state of the art modeling using latest versions of the codes (MELCOR version 2.2 & MACCS version 3.10)
- Focus on issues unique to ice condenser containment
- Consider latest plant- and site-specific information available including:
  - Core inventory, Population, Emergency response
- Integrate consideration of uncertainty into accident progression and consequence analysis
- Two primary variations of seismically initiated unmitigated SBO
  - Short-term SBO is the focus of uncertainty analysis: loss of all AC power and turbine-driven auxiliary feedwater pump (TDAFW) not available
  - Long-term SBO: sensitivity analysis involving loss of all AC power and TDAFW initially available but fails after batteries deplete
- SOARCA Sequoyah NUREG/CR report is in the NRC publication process

MELCOR Containment Model







MELCOR Model Parameters (STSBO)

Figures of merit studied include cesium/iodine release magnitude, in-vessel hydrogen generation, containment failure time, and time of initial release

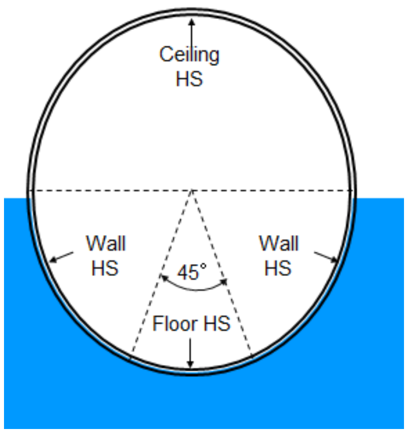
Table ES-1 Uncertain MELCOR parameters used in unmitigated STSBO UA

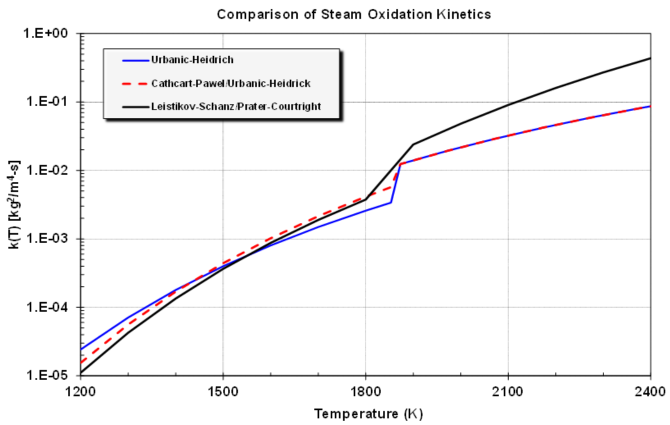
Sequence Related Parameters
Primary safety valve stochastic number of cycles until failure-to-close
Primary safety valve open area fraction
Secondary safety valve stochastic number of cycles until failure-to-close
Secondary safety valve open area fraction
In-Vessel Accident Progression
Melting temperature of the eutectic formed of fuel and zirconium oxides
Oxidation kinetics model
Ex-Vessel Accident Progression
Lower flammability limit hydrogen ignition criterion for an ignition source in lower containment
Containment rupture pressure
Barrier seal open area
Barrier seal failure pressure
Ice chest door open fraction
Particle dynamic shape factor
Time within the Fuel Cycle
Time-in-cycle

Orange indicates additional parameters considered in current UA
Blue indicated updated parameters considered in the current UA

Improved Modeling (draft → final)

- Pressurizer relief tank (PRT)
  - Heat transfer to the water pool on the outside of PRT
  - Modeling of fission product distribution in the PRT atmosphere and pool, and deposition on the walls
- Modeling of hydrogen ignition in the lower containment as a result of flow of hot gases from PRT
- Oxidation kinetic modeling
- Revised modeling of safety valves
- Modeling of TD-AFW performance using the new homologous pump model





- Urbanic-Heidrick (25%)
  - Used in *DRAFT* UA
- Catchart-Pawel/ Urbanic-Heidrick (25%)
- Leistikov-Schanz/ Prater-Courtright (50%)

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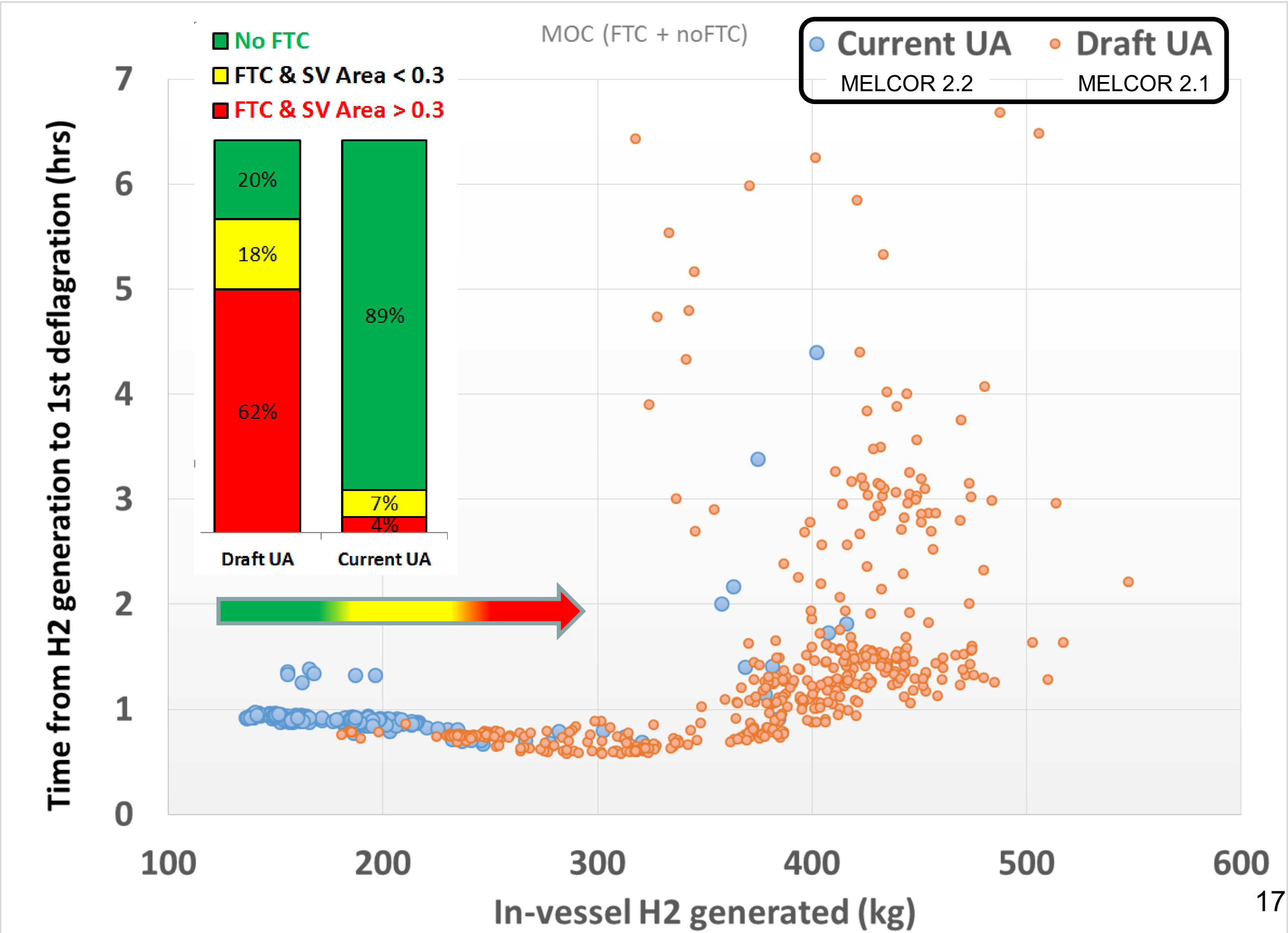
Code Update (draft → final)

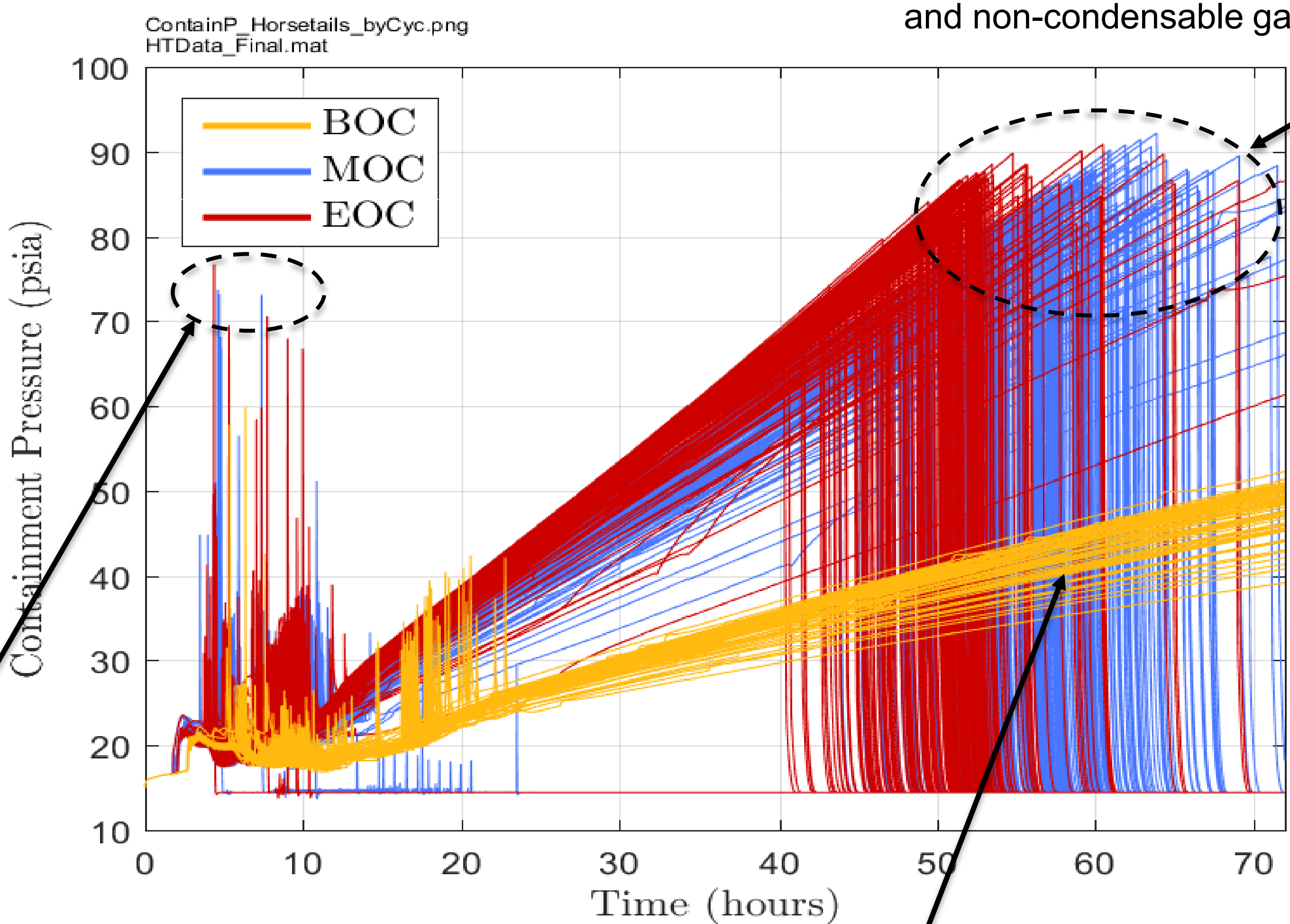
- Various MELCOR 2.2 code updates including
  - Corrections to the reflood quench model
  - Lipinski dryout model not used above the core support plate
  - Decay heat transfer to small fluid volumes
  - Correction to fuel rod collapse modeling (temperature failure criteria)
  - Ex-vessel debris cooling and spreading models
- Presentation to ACRS on April 18, 2017
  - Changes in early failures in new UA (MELCOR 2.2) calculations are due to modifications in the safety valve failing to close
  - Reduction in hydrogen generated in-vessel due to code changes not as important as model changes

STSBO Overall System Response



Overall Containment Failure Outcomes

Long-tem containment over-pressurization failure due to prolonged steam production and non-condensable gas generation

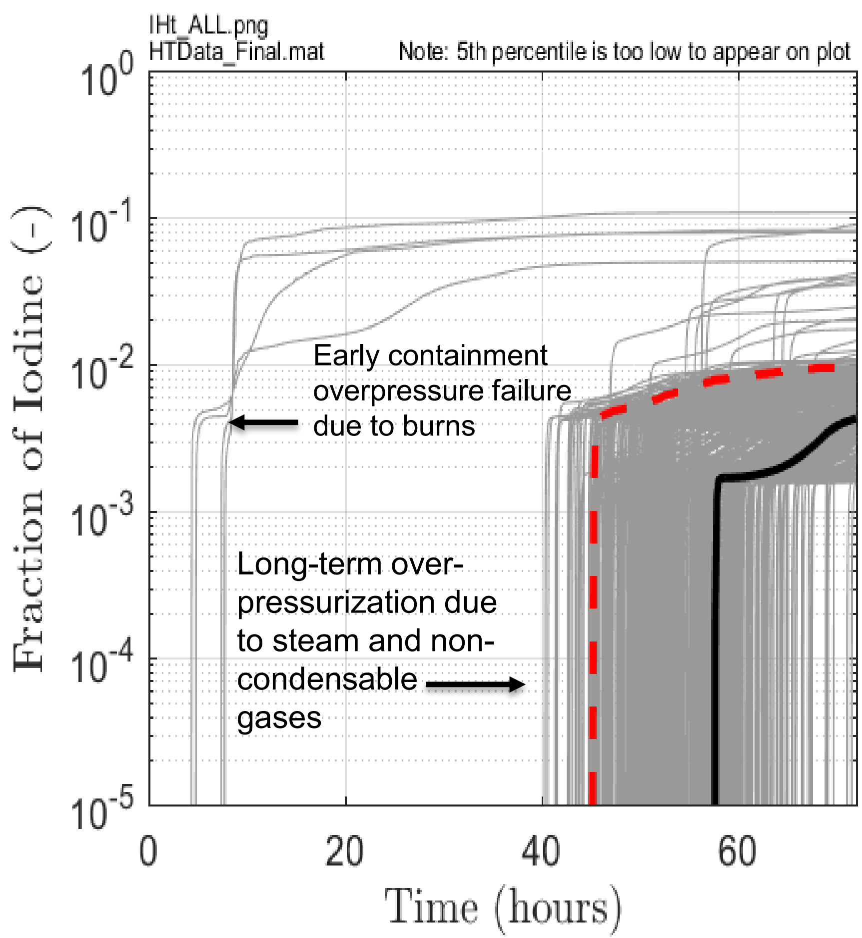


Early containment overpressure failures due to sufficiently large burns in containment

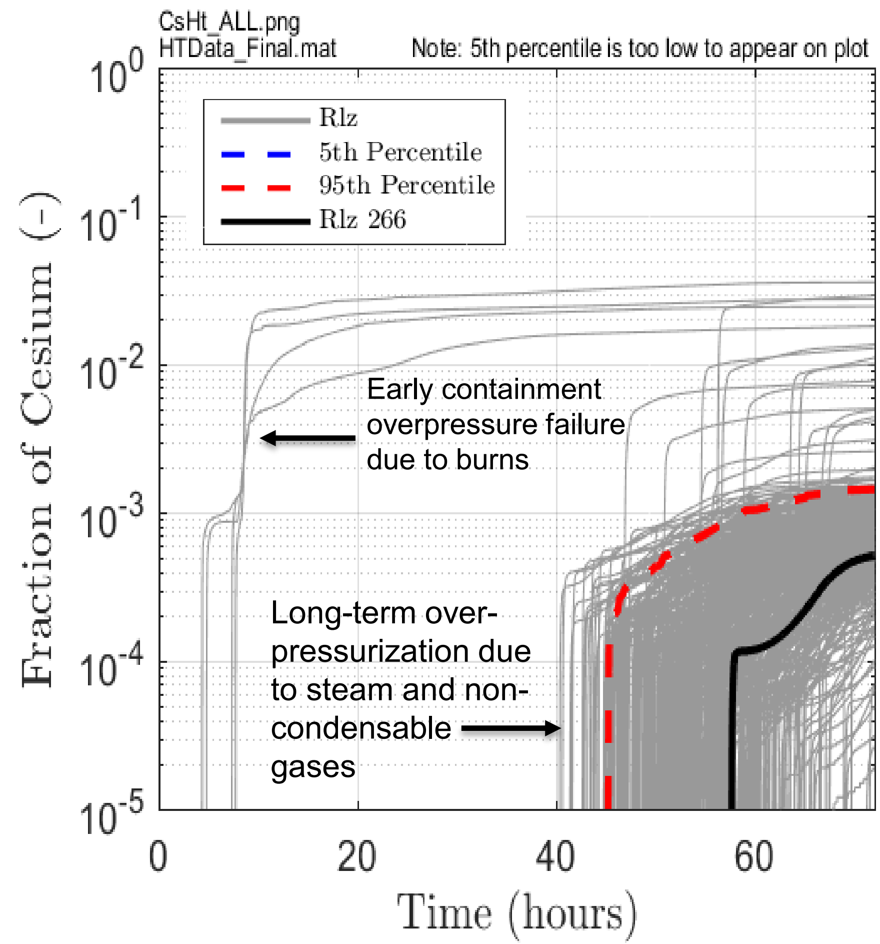
No BOC cases exhibit long-term overpressure failure before 72 hours

Cesium & Iodine release fractions

All realizations - Iodine



All realizations - Cesium



Regression analysis reveals main contributors are the primary SV cycling, time-in-cycle, containment rupture pressure, and eutectic melt temperature

Severe Accident Progression
STSBO High Level General observations

- Consequences strongly (and intuitively) affected by *early vs. late* containment failure. Early containment failure dominated by hydrogen combustion, and late containment failure results mainly from ex-vessel phenomena (e.g., core-concrete interaction)
- Early containment failures occur *only on the first hydrogen burn* (subsequent burns do not challenge containment integrity)
- Protracted safety valve (SV) cycling produces *lower in-vessel hydrogen* by the time of first burn
- Pressurizer SV failure to close (with large open area) results in greater hydrogen production and transport to the containment prior to the first burn, which increases the potential for early containment failure
- Late containment failures generally have reduced source term benefiting from fission product settling