U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Advisory Committee on Reactor Safeguards Future Plant Designs Subcommittee Meeting on Integration of Source Term Activities in Support of Advanced Reactor Initiatives

February 17, 2022

AGENDA

- Opening Remarks
- Staff Introduction
- History and Evolution of LWR Source Term
- NRC analytical tools and past studies
- SCALE/MELCOR non-LWR reference plant analysis

Break

- Agenda Item IV Continued
- NuScale EPZ Sizing Methodology Topical Report, Rev. 2
- Light water SMR design certification source term approach
- Source term approach for early non-LWR movers

Lunch

- Accident-consequence-related regulation activities

Break

- Guidance and information for developing advanced reactor source term
- Guidance for developing advanced reactor source term (long-term)
- Opportunity for Public Comment
- Member Discussion

Adjourn

Integration of Source Term Activities in Support of Advance Reactor Initiatives

John Segala
NRR/DANU

February 17, 2022

Staff Introduction

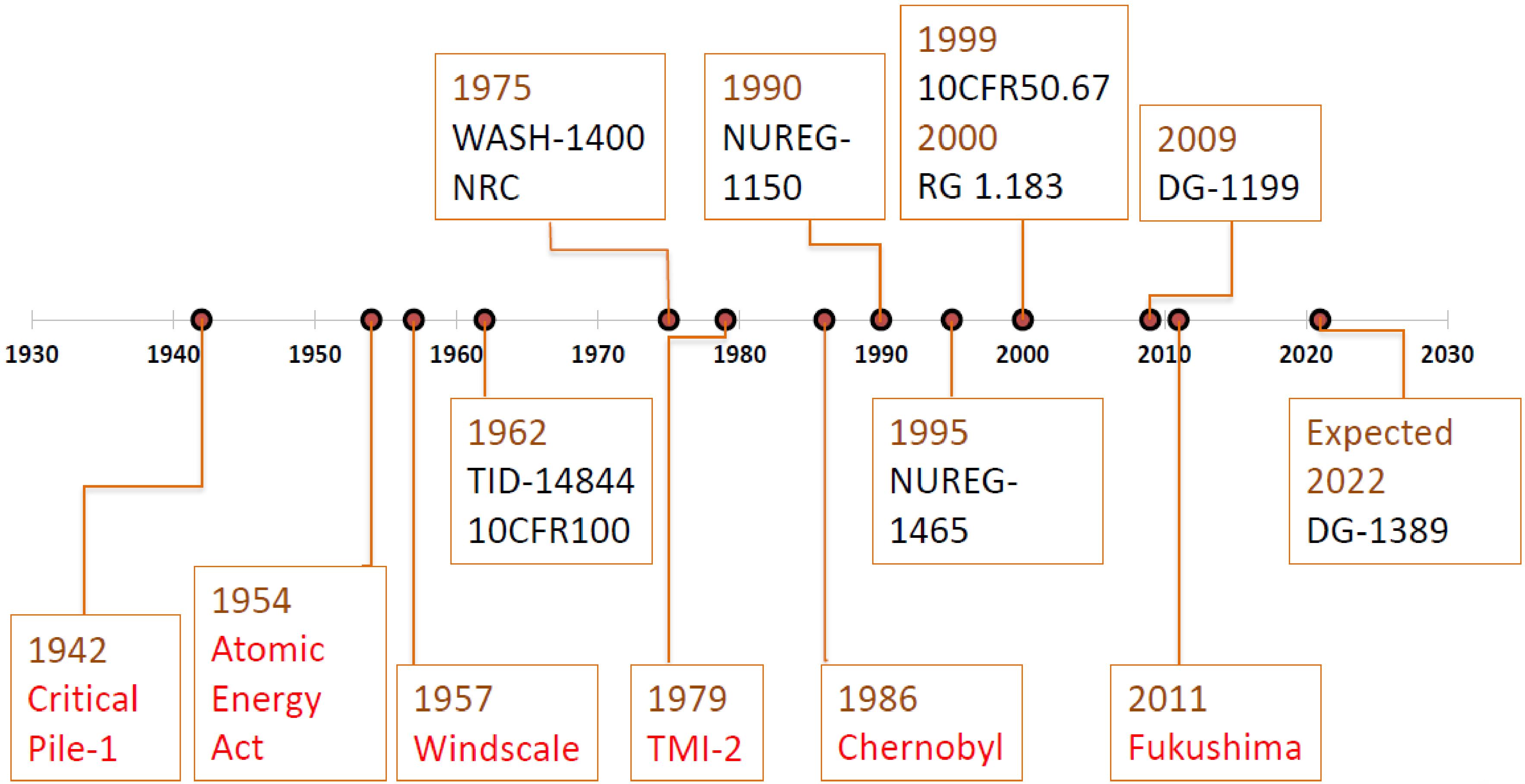
- Determining source terms is a critical component in the NRC's licensing process
- NRC team presenting today:
  - Mark Blumberg – NRR/DRA
  - Michelle Hart – NRR/DANU
  - Jason Schaperow – NRR/DANU
  - Bill Reckley – NRR/DANU
  - Tim Drzewiecki – NRR/DANU
  - Hossein Esmaili – RES/DSA

History and Evolution of LWR Source Terms

Mark Blumberg

Radiation Protection and Consequence Branch
Division of Risk Assessment
Office of Nuclear Reactor Regulation

LWR Source Term Timeline



History – Regulatory Use of Source Terms

- Siting critical issue
  - Safety & Cost
- Principle hazard – Public Exposure
  - Siting key element in protecting public health
- Earliest reactors used containments
- Atomic Energy Commission proposed siting on population densities
- Ultimately decided siting would be based upon dose calculations

10 CFR 100.11

- Footnote to 10 CFR 100.11(a) is a performance-based rule to evaluate the defense-in-depth provided by the containment
- Nearly all current reactors were licensed originally to the Technical Information Document (TID) -14844 which provides guidance on the containment source term for the Loss of Coolant Accidents (LOCAs) involving fuel melt
  - Based on heating fuel ‘chips’ in a furnace
  - 100% noble gases (Xe, Kr)
  - 50% iodine (half deposits instantaneously)
  - 1% of other radionuclides as particles
- Iodine Chemical Form
  - 91% as I_2(g) (elemental); 5% particles; 4% CH_3I (organic)
- All instantly available from start of accident in the containment
- Source terms for Non-LOCA events are provided in RG 1.195, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light Water Reactors”

NUREG-1465 Source Term

- Radionuclide behavior observed during the Three Mile Island Unit 2 accident in 1979 did not appear at all to be like the Technical Information Document (TID)-14844 source term
- NRC initiated research effects in the area of severe accidents which culminate in publication of NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants.” (1990)
- Source term depends on the nature of the accident
- The NUREG-1465, “Accident Sources Terms for Light-Water Nuclear Power Plants” (1995) source term was derived from the risk significant sequences in NUREG-1150

10 CFR 50.67, RG 1.183

- NRC staff developed RG 1.183 Rev. 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." (July 2000) to support implementation of 10 CFR 50.67, "Accident Source Term"
  - Applicable to nuclear power reactor applicants and licensees who voluntarily adopt 10 CFR 50.67
  - Provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an AST
  - Used the NUREG-1465 early in-vessel fuel melt source term for LOCAs
  - RG 1.183 also provides Non-LOCA release fractions
  - Identified the significant attributes of an acceptable AST

TID-14844 vs. NUREG-1465

BWR Source Term

	NUREG 1465		TID 14844
	Gap	Early In-vessel	
Duration (hours)	.5	1.3	Instant.
Noble Gases (%)	5	95	100
Halogens (%)	5	25	50
	Elemental I2– 4.85 Aerosol (CsI) – 95 Organic 0.15		Elem. – 91 Aerosol –5 Organic 4

Source Term Updates Proposed in DG-1199

- In October 2009, the NRC issued for public comment DG-1199 as a proposed Rev. 1 of RG 1.183
- Addressed fuel utilization at the time for Non-LOCA accidents
- The NRC staff has elected not to finalize DG-1199 and is issuing DG-1389 as a replacement

Source Term Updates Proposed in DG-1389

- Staff plans to include changes proposed in DG-1199 as modified by public comments
- Provides guidance to address the review of near-term accident tolerant fuel (ATF) designs with burnups up to 68 GWd/MTU peak rod-average) and U-235 enrichments up to 8.0 weight percent.
- Considered impact of fuel fragmentation, relocation and dispersal[1]
- On going research efforts is underway to update the SAND2011-0128 accident source term to accommodate higher burnup and increased enrichments for LOCA releases.

1 NRC Memorandum, "Letter Report on Evaluation of the Impact of Fuel Fragmentation, Relocation, and Dispersal for the Radiological Design Basis Accidents in Regulatory Guide 1.183 (ADAMS Accession No. ML21197A067)"

Source Term Updates Proposed in DG-1389 (cont.)

- A future RG 1.183 update is expected to accommodate higher burnups and enrichments
- An acceptable analytical procedure for predicting plant-specific non-LOCA radionuclide release fractions has been included and provides flexibility and margin recovery
- Separate BWR and PWR non-LOCA steady state release fractions

Key Messages

- One of the ways the NRC staff and licensees determine what measures and barriers are needed to protect the health and safety of the public is to perform design basis accident dose analyses.
- A key component of these analyses is the determination of the release source term.
- The NRC has developed regulations, source terms and regulatory guidance to provide licensees and the staff with an efficient method of performing these dose analyses.
- Ongoing efforts by the NRC continue to revise these source terms and methods to address modern fuel utilization and the use of accident tolerant fuel.

NRC Analytical Tools and Past Studies-Severe Accident Progression and Source Term

Hossein Esmaili, RES/DSA
Jason Schaperow, NRR/DANU

Key Messages

- Decades of NRC and international investments in the state-of-practice SCALE and MELCOR modeling including development, assessment and application
- Importance of analytical capabilities in a system level code and being ready to resolve regulatory issues and help decision making
- Leverage international collaboration through severe accident research and code sharing programs
- Application to a wide variety of nuclear technologies

Outline

- Introduction
- MELCOR Code Overview
- International Collaboration (Severe Accidents & MELCOR)
- Applications to Regulatory Decision-making
  - Examples: Design Certification, SOARCA, Post-Fukushima activities
- Application to New and Advanced Reactors
  - SCALE/MELCOR demonstration calculations

Introduction

Source Term Development Process

Experimental Basis ← **PIRT process**

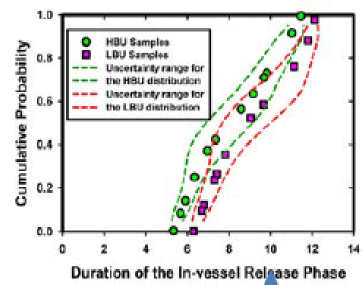
- Oxidation/Gas Generation
- Melt Progression
- Fission Product Release
- Fission Product Transport

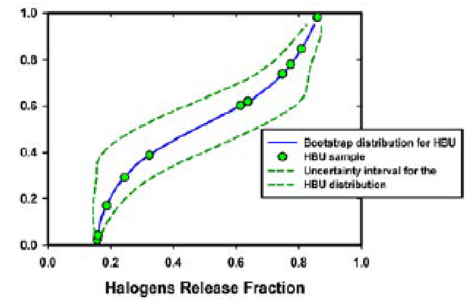
e.g., NUREG/CR-7283 (ATF/HBU)

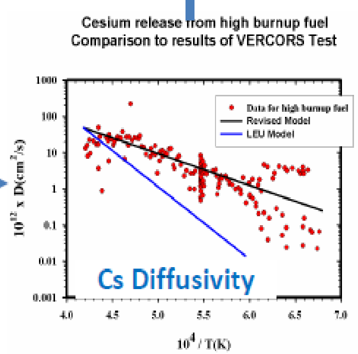
MELCOR → **Accident Analysis**

Scenario # 1	Scenario # 2
…………………	…………………
Scenario # n-1	Scenario # n

→ **Synthesize** timings and release fractions → Design Basis Source Term

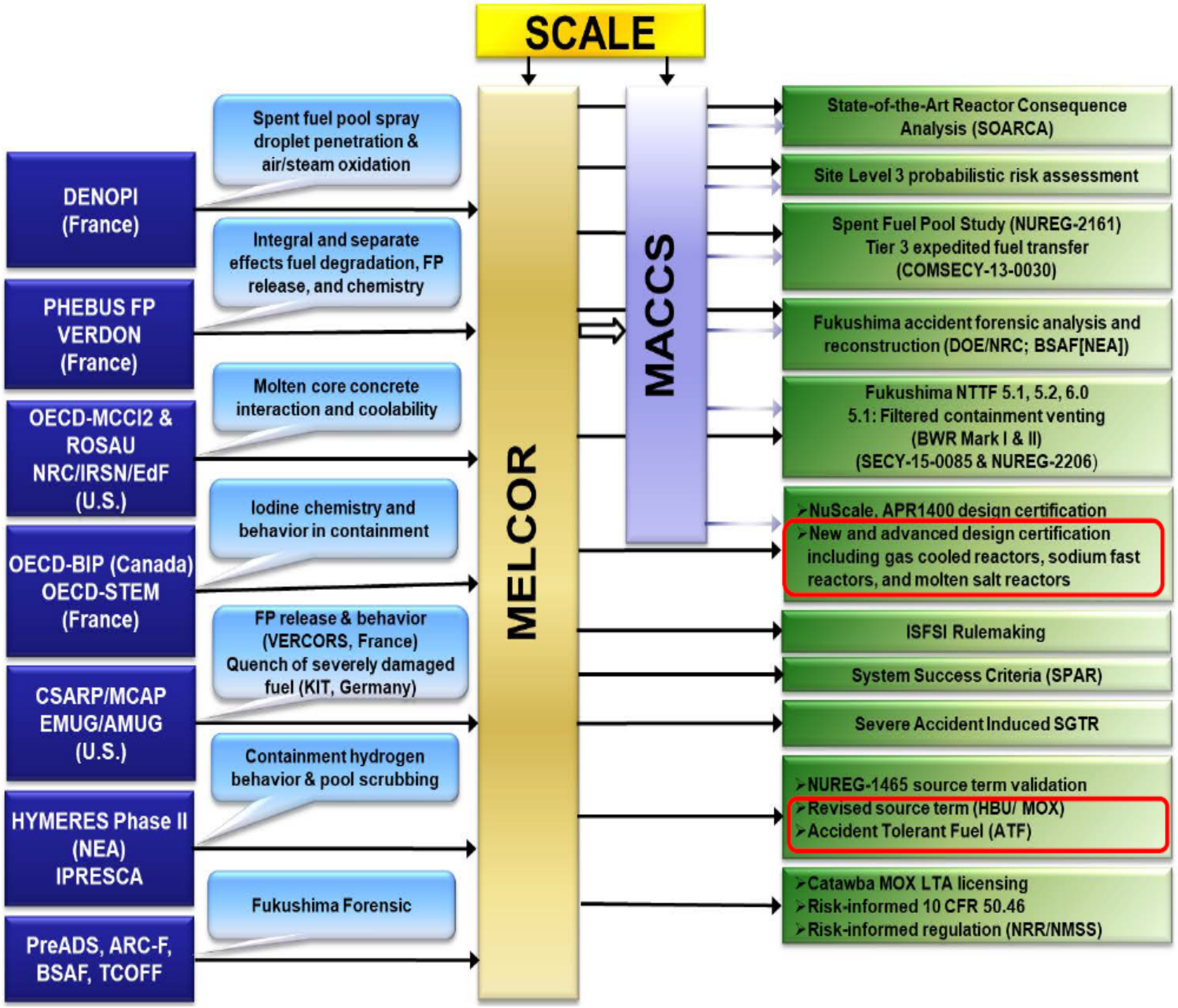






Fission Product class	Gap 0.5 hr	In-vessel 1.3 hr	Ex-vessel 2 hr	Late 10 hr
Noble Gases	5%	95%	~0	~0
Iodine, bromine	5%	35%	25%	1%
Cesium	5%	25%	35%	1%
Tellurium	~0	5%	25%	0.5%
Ba, Sr	~0	2%	10%	~0
Ru, Mo, Pd, etc.	~0	0.25%	0.25%	~0
Lanthanides	~0	0.02%	0.5%	~0
Cerium group	~0	0.05%	0.5%	~0

Code Development & Regulatory Applications



What Is It?
MELCOR is an engineering-level code that simulates the response of the reactor core, primary coolant system, containment, and surrounding buildings to a severe accident.

Who Uses It?
MELCOR is used by domestic universities and national laboratories, and international organizations in around 30 countries. It is distributed as part of NRC's Cooperative Severe Accident Research Program (CSARP).

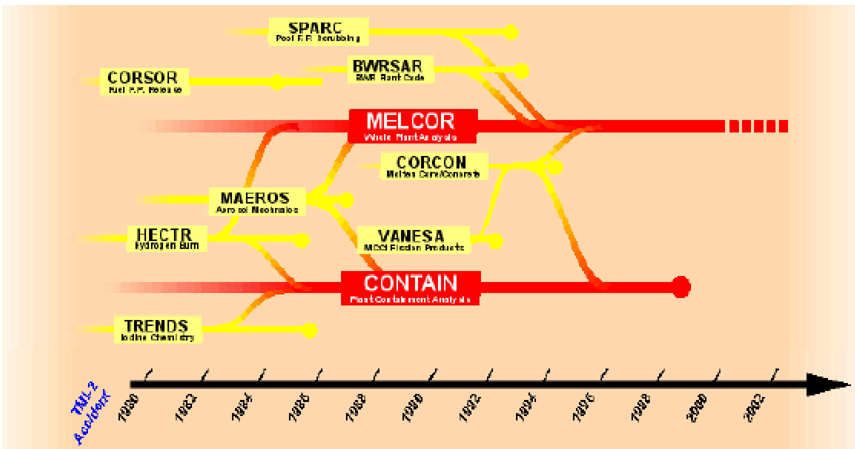
How Is It Used?
MELCOR is used to support severe accident and source term activities at NRC, including the development of regulatory source terms for LWRs, analysis of success criteria for probabilistic risk assessment models, site risk studies, and forensic analysis of the Fukushima accident.

How Has It Been Assessed?
MELCOR has been validated against numerous international standard problems, benchmarks, separate effects (e.g., VERCORS) and integral experiments (e.g., Phebus FPT), and reactor accidents (e.g., TMI-2, Fukushima).

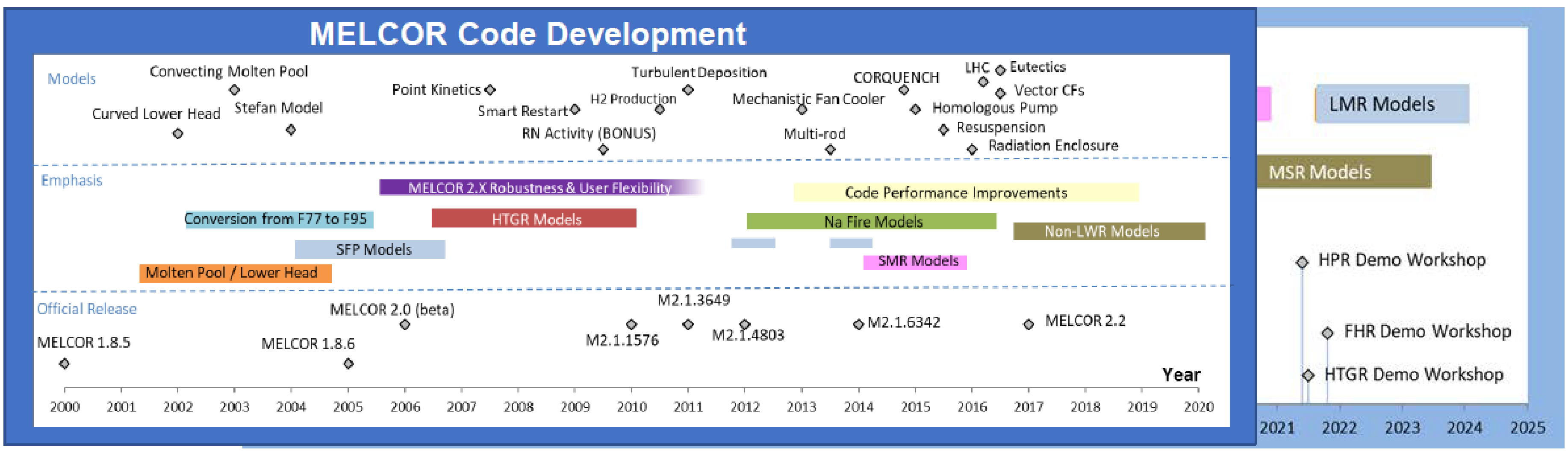
MELCOR Overview

MELCOR History

MELCOR developed at Sandia National Laboratories for NRC since 1982



Version	Date
2.2.21440	December 2021
2.2.18180	December 2020
2.2.14959	October 2019
2.1.11932	November 2018
2.1.9541	February 2017



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MELCOR Development

Fully integrated, engineering-level code

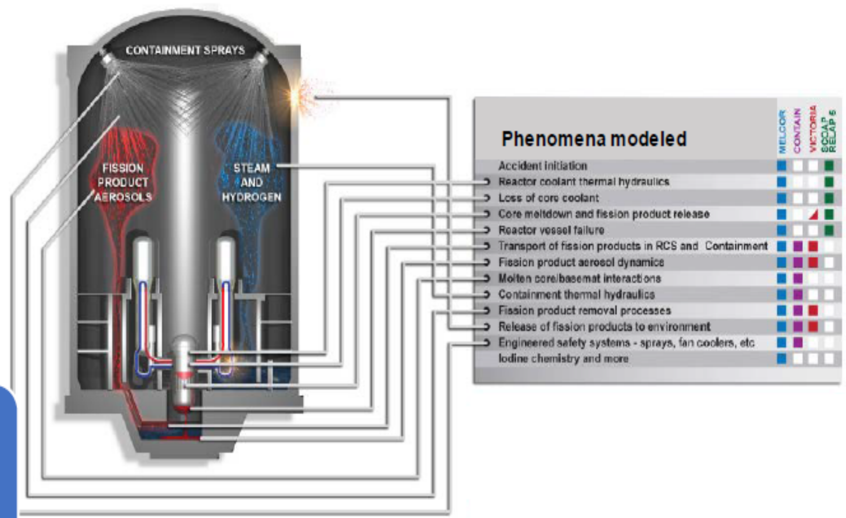
- Thermal-hydraulic response of reactor coolant system, reactor cavity, rector enclosures, and auxiliary buildings
- Core heat-up, degradation and relocation
- Core-concrete interaction
- Flammable gas production, transport and combustion
- Fission product release and transport behavior

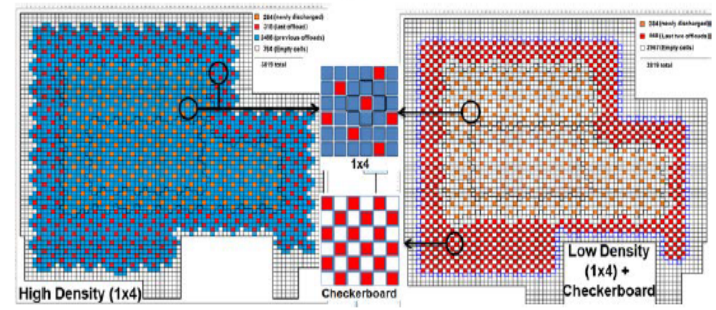
Level of physics modeling consistent with

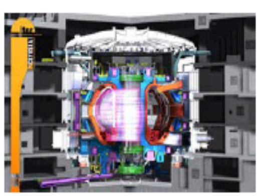
- State-of-knowledge
- Necessity to capture global plant response
- Reduced-order and correlation-based modeling often most valuable to link plant physical conditions to evolution of severe accident and fission product release/transport

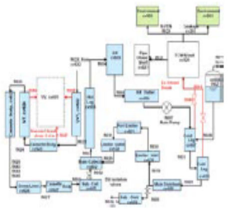
Traditional application

- Models constructed by user from basic components (control volumes, flow paths and heat structures)
- Demonstrated adaptability to new reactor designs – HPR, HTGR, SMR, MSR

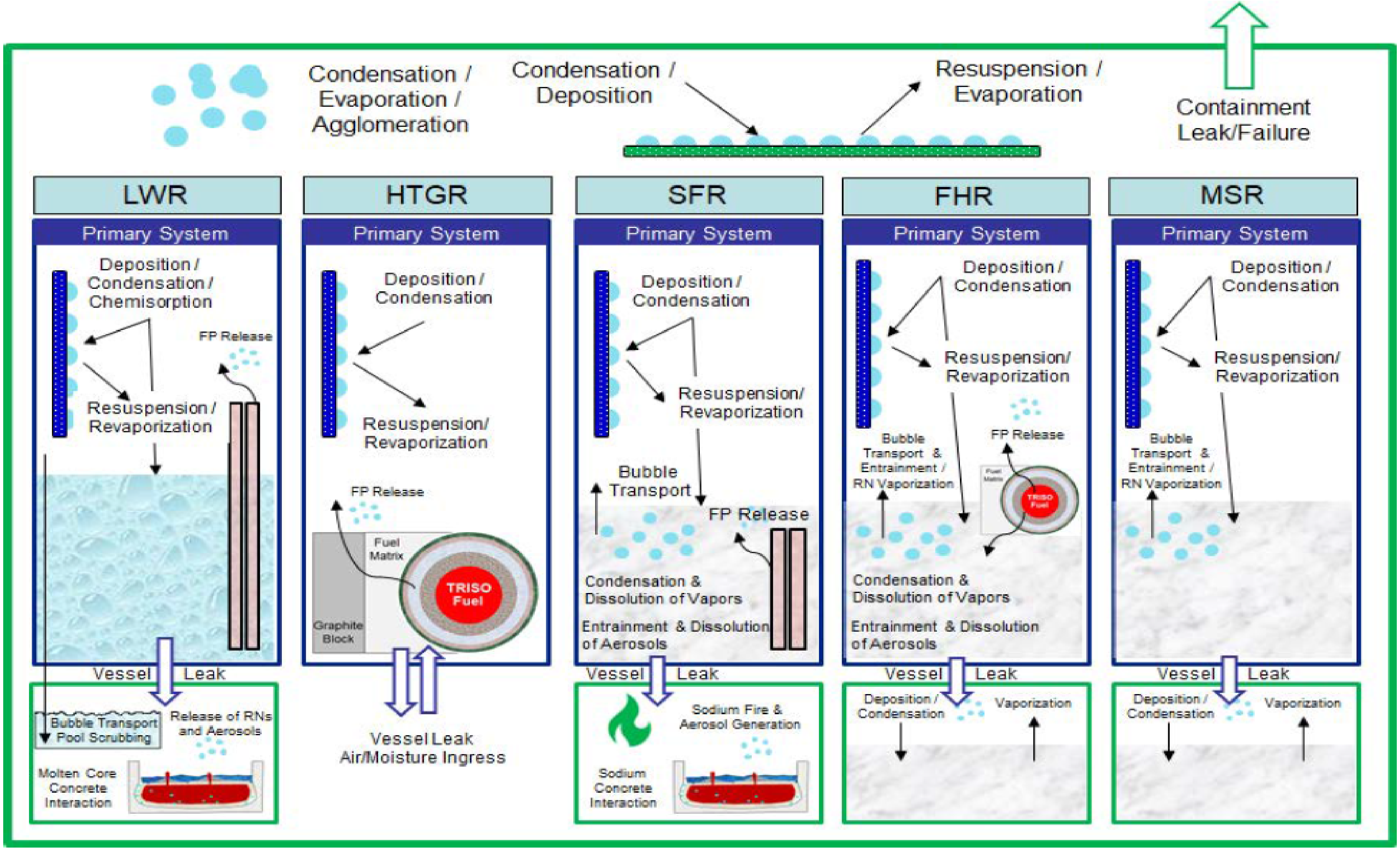






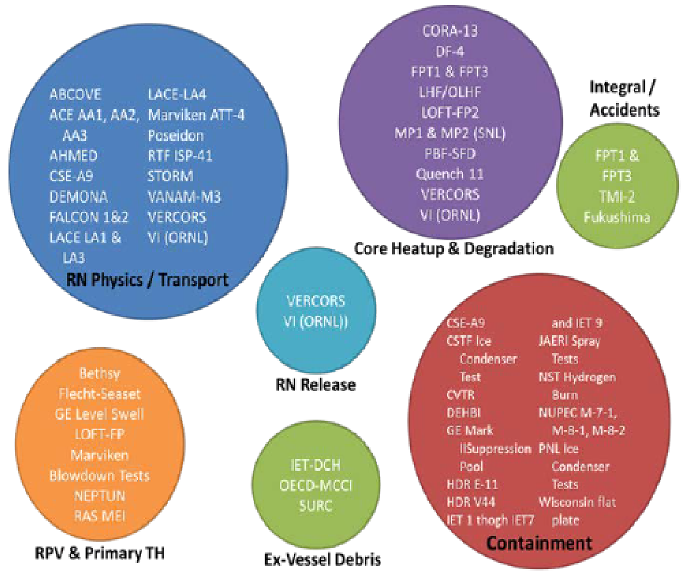


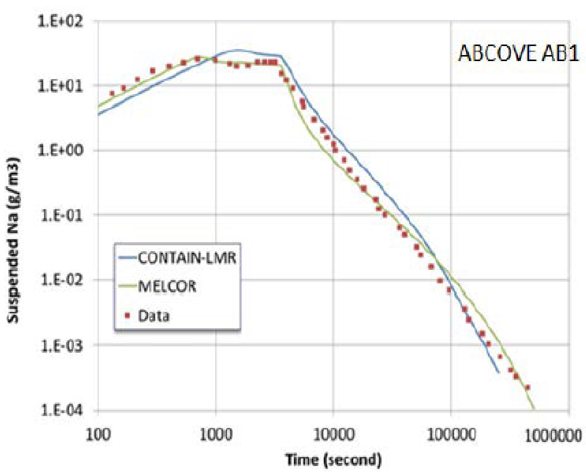
MELCOR Flexibility - Common Phenomena



MELCOR Verification & Validation Basis

Primer & User Guide
Reference Manual
Assessment Problems





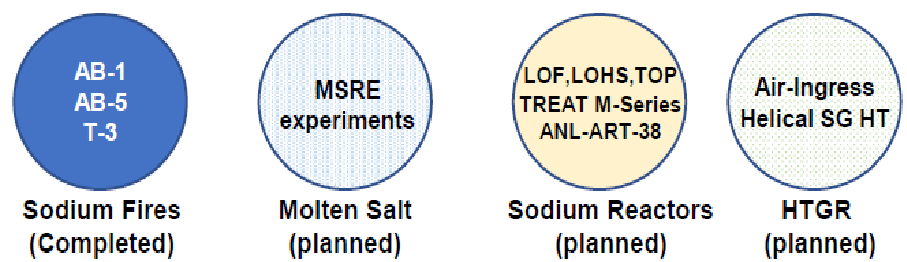
TRISO Diffusion Release

IAEA CRP-6 Benchmark Fractional Release

Case	1a	1b	2a	2b	3a	3b
US/INL	0.467	1.0	0.026	0.996	1.32E-4	0.208
US/GA	0.453	0.97	0.006	0.968	7.33E-3	1.00
US/SNL	**0.465**	**1.0**	**0.026**	**0.995**	**1.00E-4**	**0.208**
US/NRC	0.463	1.0	0.026	0.989	1.25E-4	0.207
France	0.472	1.0	0.028	0.995	6.59E-5	0.207
Korea	0.473	1.0	0.029	0.995	4.72E-4	0.210
Germany	0.456	1.0	0.026	0.991	1.15E-3	0.218

(1a): Bare kernel (1200 °C for 200 hours)
(1b): Bare kernel (1600 °C for 200 hours)
(2a): kernel+buffer+iPyC (1200 °C for 200 hours)
(2b): kernel+buffer+iPyC (1600 °C for 200 hours)
(3a): Intact (1600 °C for 200 hours)
(3b): Intact (1800 °C for 200 hours)

Specific to non-LWR application

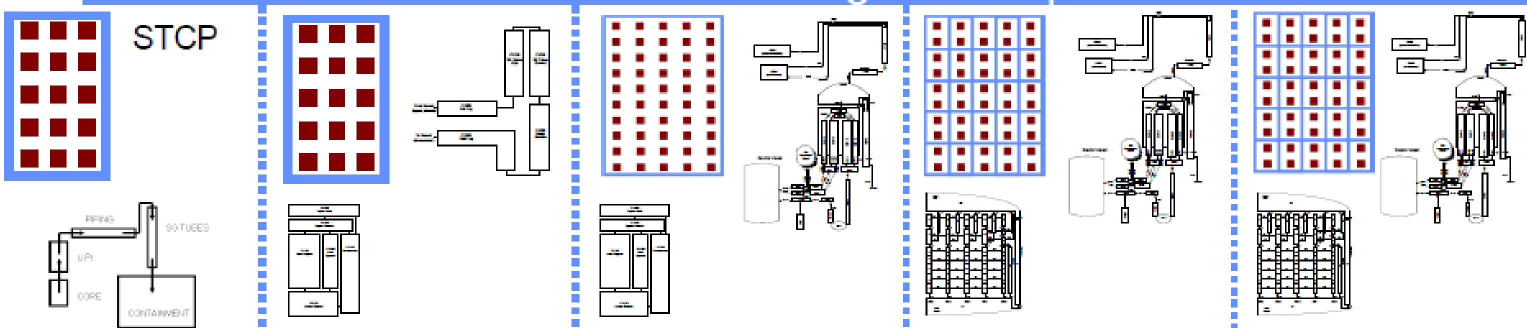


MELCOR State-of-the-Practice Modeling

Timeline for Evolution of MELCOR Modeling Practices

Circa 1985	Circa 1990	Circa 1995	Circa 1998	Present	Future

Modeling Techniques



Example Regulatory Applications

Circa 1985	Circa 1990	Circa 1995	Circa 1998	Present	Future
• NUREG-1150 • Basis for NUREG-1465 revised source term	• AP-600 design certification • ESBWR design certification • AP-1000 design certification	• Begin SGTR	• Finish SGTR	• MOX source terms • High burn-up source terms	• Emerging user needs

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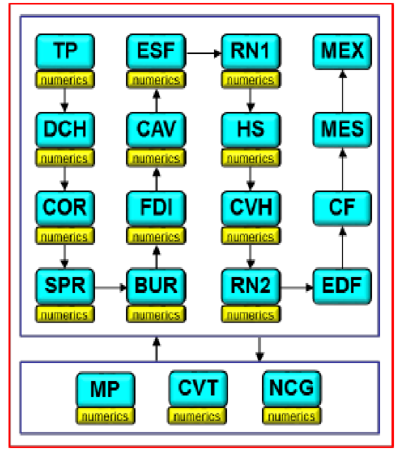
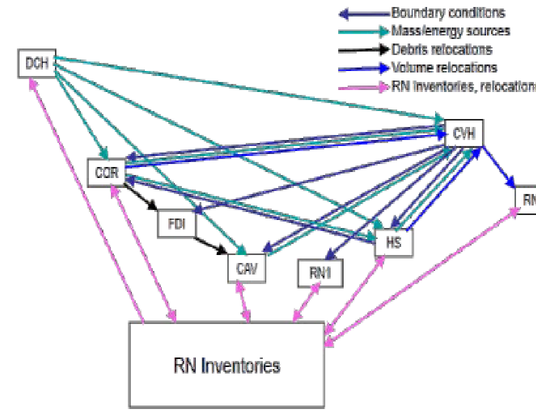
MELCOR Modernization

- Generalized numerical solution engine
- Hydrodynamics
- In-vessel damage progression
- Ex-vessel damage progression
- Fission product release and transport

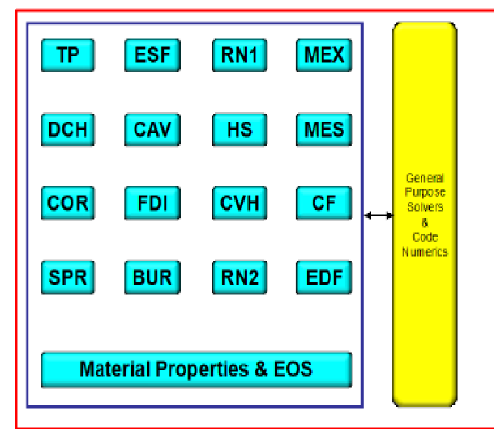


TP = Transfer Process
DCH = Decay Heat
COR = Core
SPR = Containment Spray
BUR = Gas Combustion
FDI = Fuel Dispersal Interaction
CAV = Cavity (MCCI)
ESF = Engineered Safety Features
MP = Material Properties

RN = Radionuclide
HS = Heat Structure
CVH = CV Hydrodynamics
EDF = External Data File
CF = Control Function
MES = Special Messages
MEX = Executive
CVT = CV Thermodynamics
NCG = Non Condensible Gas

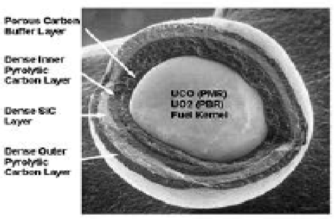


Separate **Physics** & **Numerics**



MELCOR Data Requirements

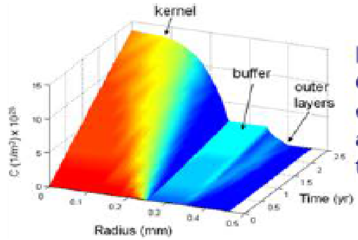
Input Data	HTGR	SFR	MSR	FHR
FP Inventory	SCALE	SCALE	SCALE	SCALE
FP diffusion coefficients (D) and release	Experiments (e.g., AGR) and analysis (e.g., DOE tools)	Experiments		Experiments (e.g., AGR) and analysis (e.g., DOE tools)
Core power shape	Radial/Axial profiles (e.g., SCALE)	Radial/Axial profiles (e.g., SCALE)	Radial/Axial profiles (e.g., SCALE)	Radial/Axial profiles (e.g., SCALE)
Fuel failure	Experiments/other codes (e.g., DOE tools)	Experiments/other codes (e.g., DOE tools)		Experiments/other codes (e.g., DOE tools)
Dust generation & FP transport	Experiments, historical data and other code (e.g., DOE tools)			
FP release under air/water ingress & interaction w/ graphite	Experiments			
Kinetics parameters and reactivity feedback coefficients	Experiments/other codes (e.g., SCALE)	Experiments/other codes (e.g., SCALE)	Experiments/other codes (e.g., SCALE)	Experiments/other codes (e.g., SCALE)
Equilibrium constants for release from pool and vapor pressure data		Experiments/other codes (e.g., DOE tools)	Experiments/other codes (e.g., DOE tools)	Experiments/other codes (e.g., DOE tools)



$$\frac{\partial C}{\partial t}=\frac{1}{r^m}\frac{\partial}{\partial r}\left(r^m \boxed{D}\frac{\partial C}{\partial r}\right)-\lambda C+\boxed{S}$$

SCALE

Experiments/Analysis



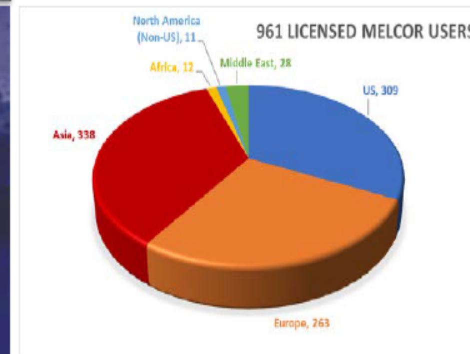
Distribution of Cs-137 in different layers as a function of time

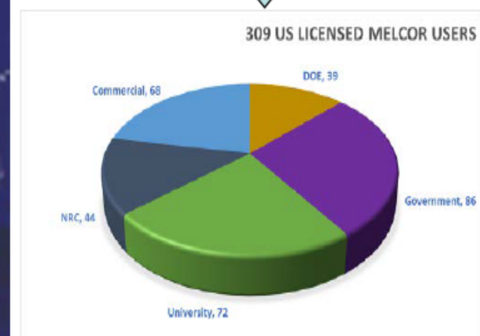
International Collaboration

User Groups & Technical Meetings

Cooperative Severe Accident Research Program (CSARP) – June/U.S.A
MELCOR Code Assessment Program (MCAP) – June/U.S.A
European MELCOR User Group (EMUG) Meeting – Spring/Europe
Asian MELCOR User Group (AMUG) Meeting – Fall/Asia

~1000 Code Users Worldwide



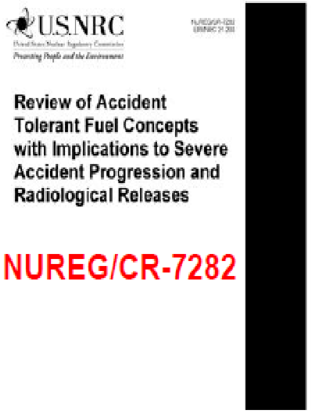


International Severe Accident Projects

- **1988-2010** — IRSN: Phébus-Fission Products Program
- **2005-2010** — IRSN: Phébus-International Source Term Program
- **2006-2019** — NEA: Behavior of Iodine Project (BIP)
- **2005-2016** — IRSN: Experimental Program for Iodine Chemistry Under Radiation (EPICUR)
- **2011-2019** — NEA: Source Term Evaluation and Mitigation (STEM) Project
- **2013–2018** — NEA: Benchmark Study of the Accident at Fukushima (BSAF) Project
- **2017–2021** — NEA: Preparatory Study on Analysis of Fuel Debris (PreADES) Project
- **2019–2021** — NEA: Analysis of Information from Reactor Buildings and Containment Vessels of Fukushima (ARC-F) Project
- **2019–2023** — EU: Management and Uncertainties of Severe Accidents (MUSA)
- **2020–2024** — NEA: Experiments on Source Term for delayed Releases (ESTER)
- **2020–2024** — NEA: Reduction of Severe Accident Uncertainties (ROSAU)

Advanced Fuel Technologies

- Panel of international severe accident experts Phenomena Identification and Ranking Tables (PIRT) that addressed significant phenomenological issues to improve MELCOR
- Source term calculations for HBU/HALEU fuel
- **QUENCH-ATF:** Experiments for ATF cladding materials in the QUENCH facility at Karlsruhe Institute of Technology (KIT) – Near term chromium-coated cladding under design basis accident (DBA) and beyond DBA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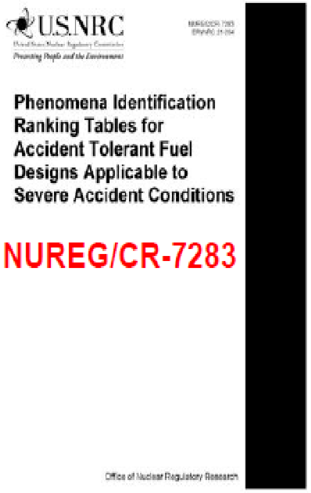
Review of Accident Tolerant Fuel Concepts with Implications to Severe Accident Progression and Radiological Releases

NUREG/CR-7282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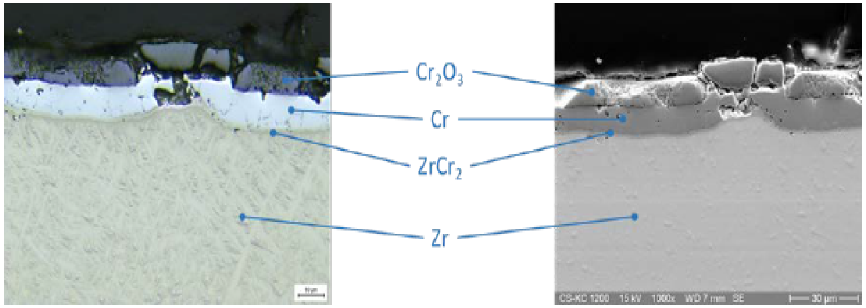
Phenomena Identification Ranking Tables for Accident Tolerant Fuel Designs Applicable to Severe Accident Conditions

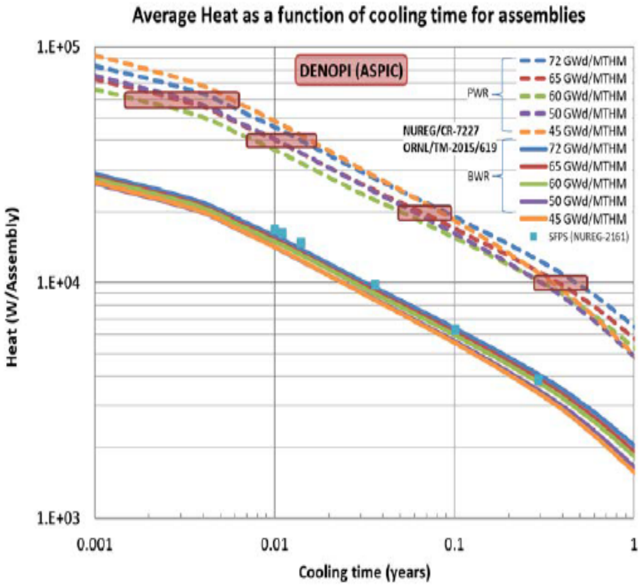
NUREG/CR-7283

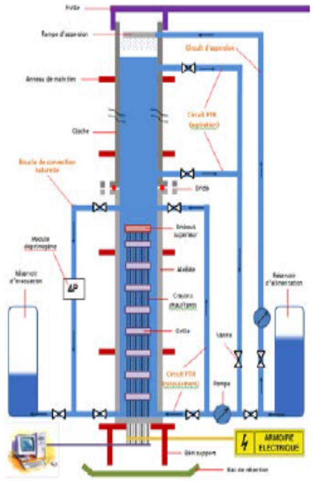
Office of Nuclear Regulatory Research



MELCOR SFP Modeling

- **SECY-16-0100**: "National Academy of Sciences Study of the Lessons Learned from the Fukushima Nuclear Accident for Improving Safety and Security of U.S. Nuclear Power Plants"
- **DENOPI (NRC-IRSN/France):** Provide experimental data to validate spray efficacy on cooling spent fuel bundles, and cladding oxidation under a mixture of steam and air environment.
- Enhance MELCOR SFP capabilities

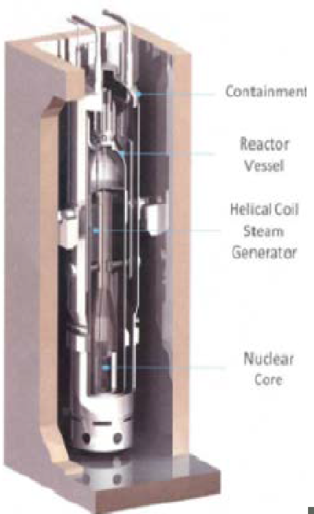




MELCOR Applications

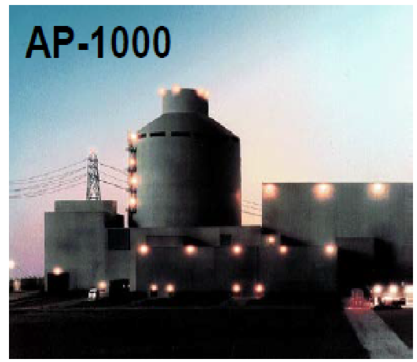
Design Certification

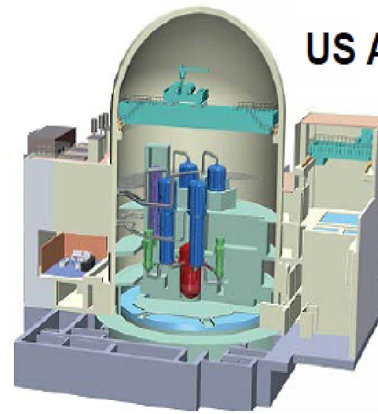
- Severe accident response and source term
- Containment response to design basis accident



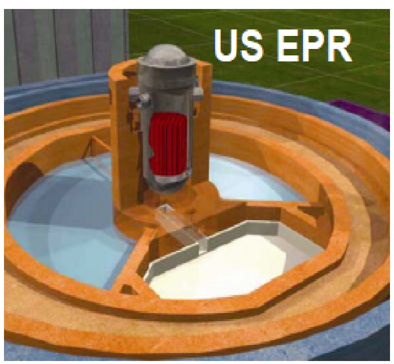
NuScale

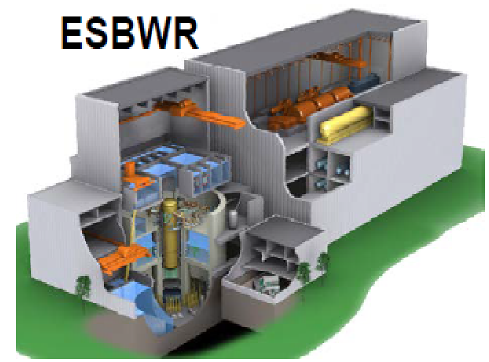
APR-1400





US APWR

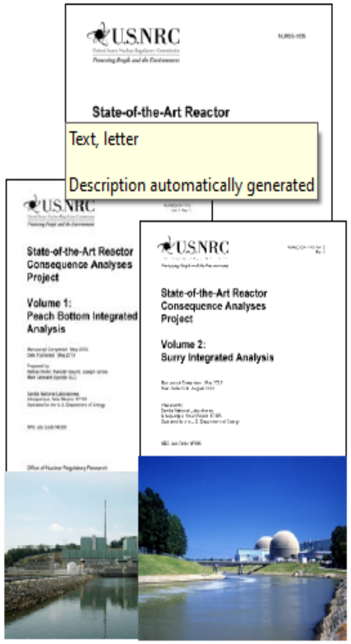


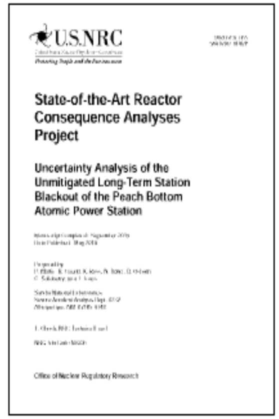


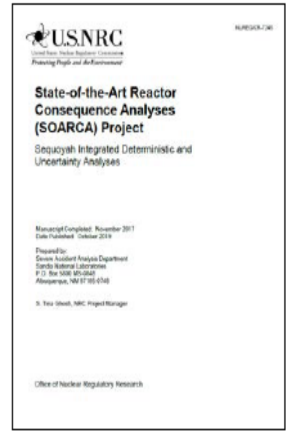
State-of-the-Art Reactor Consequence Analysis (SOARCA)

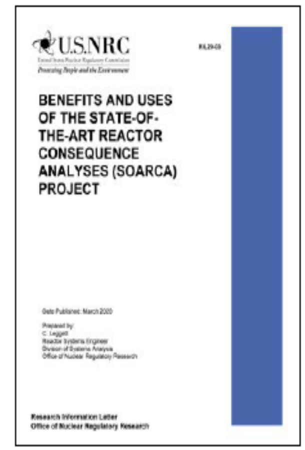
Project Start

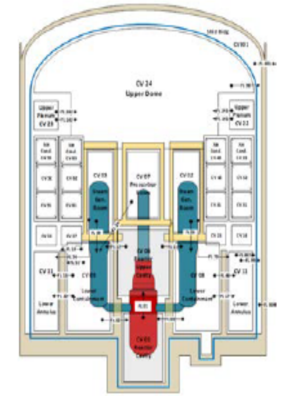
SECY-05-0233

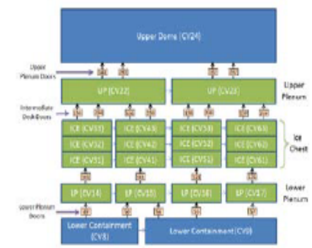












Uncertainty Analysis

2006

2012

2016

2019

2020

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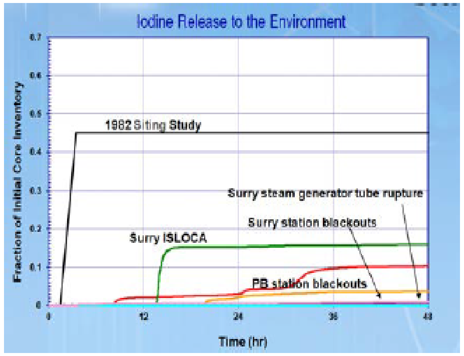
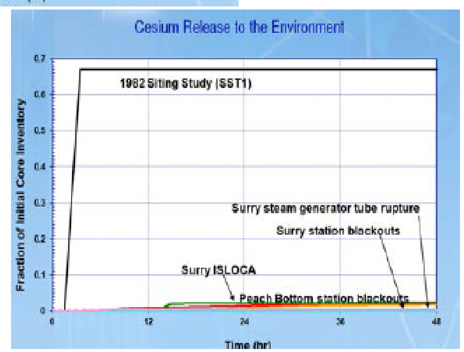
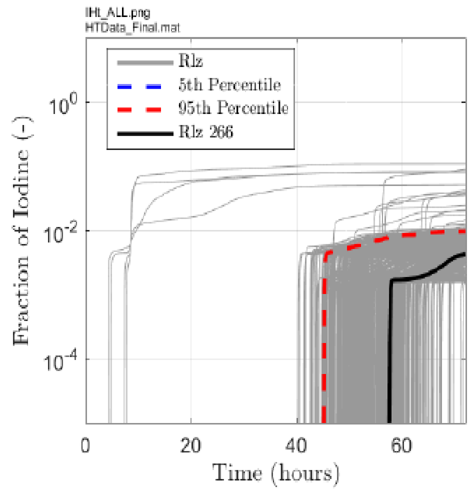
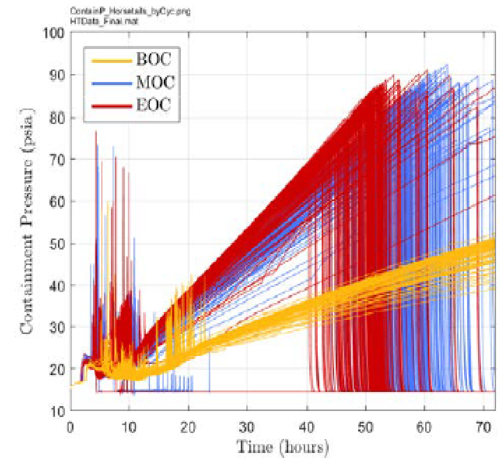
United States Nuclear Regulatory Commission
Protecting People and the Environment

State-of-the-Art Reactor Consequence Analysis (SOARCA)





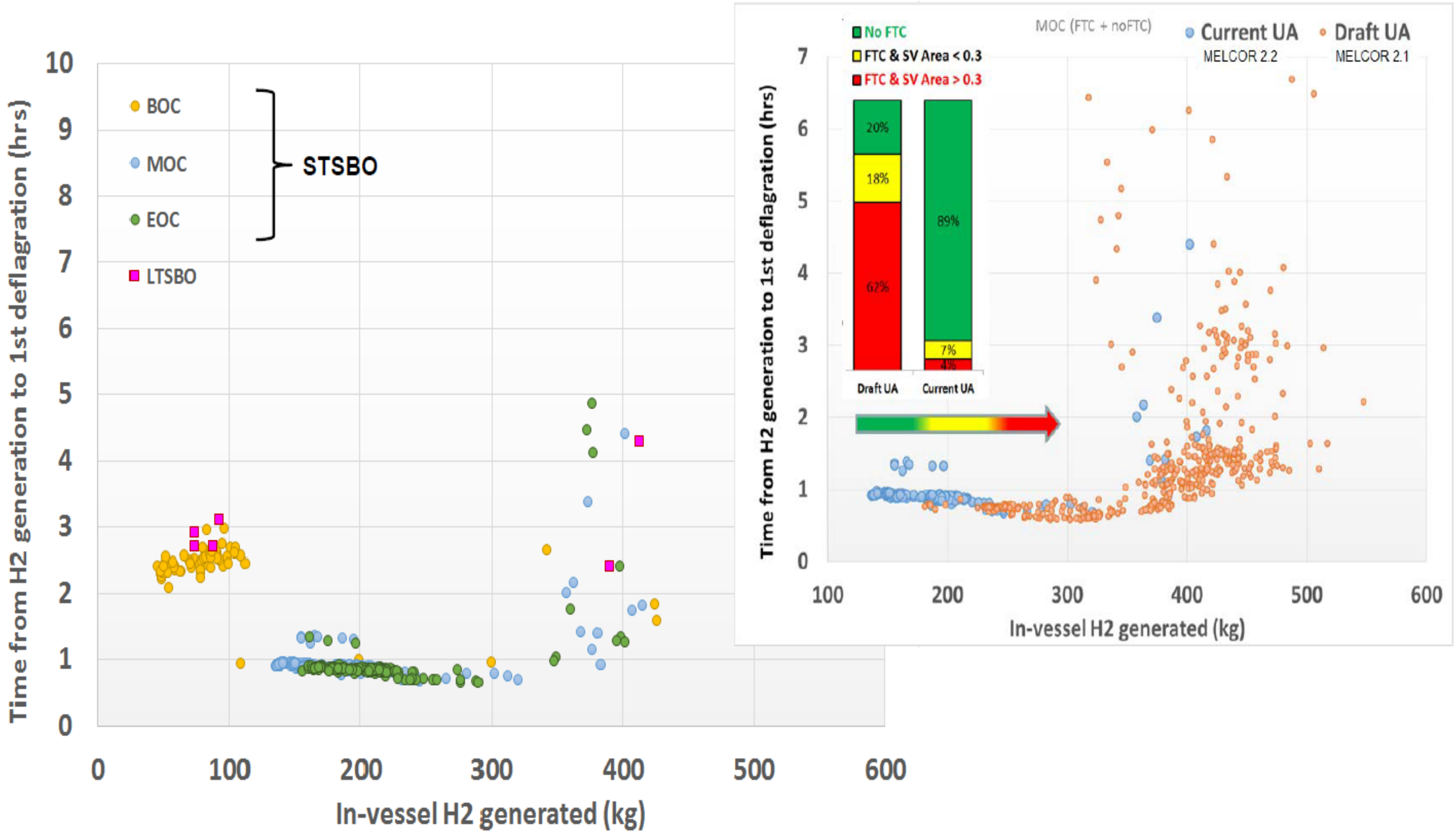




Uncertainty Analysis

2006 2012 2016 2019 2020

Uncertainty Analysis (SOARCA) NUREG/CR-7245



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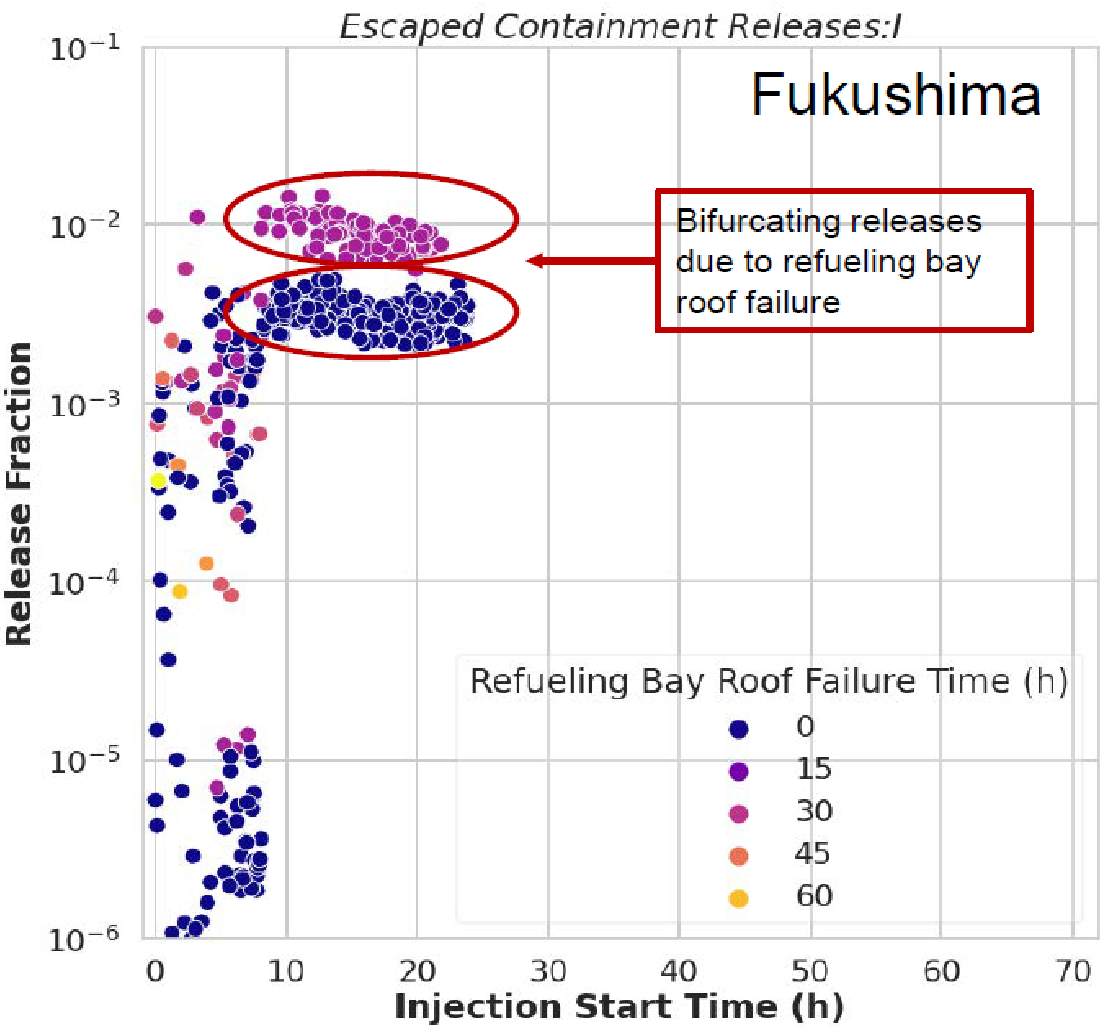
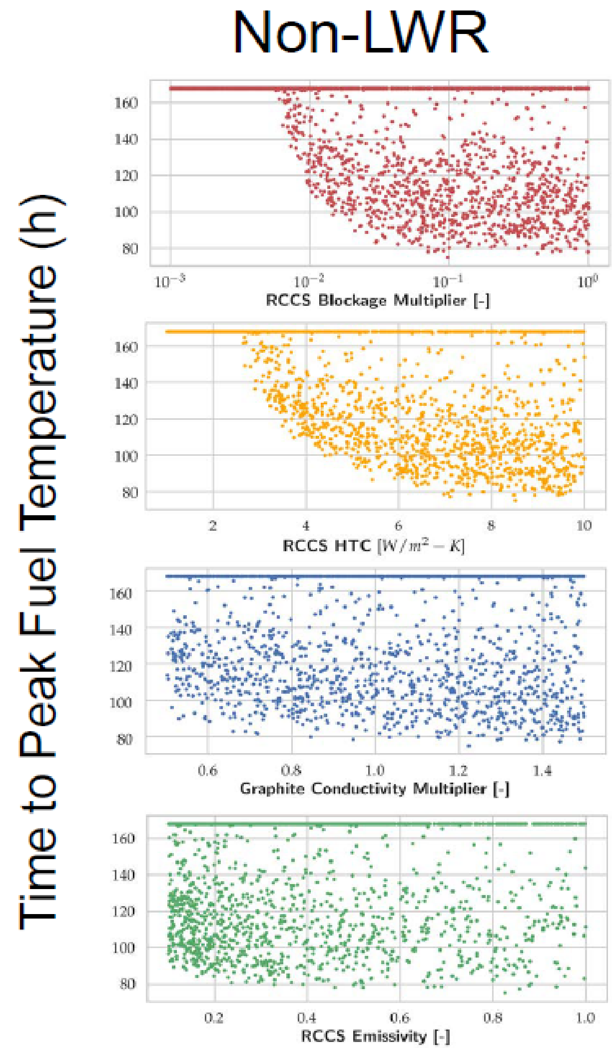
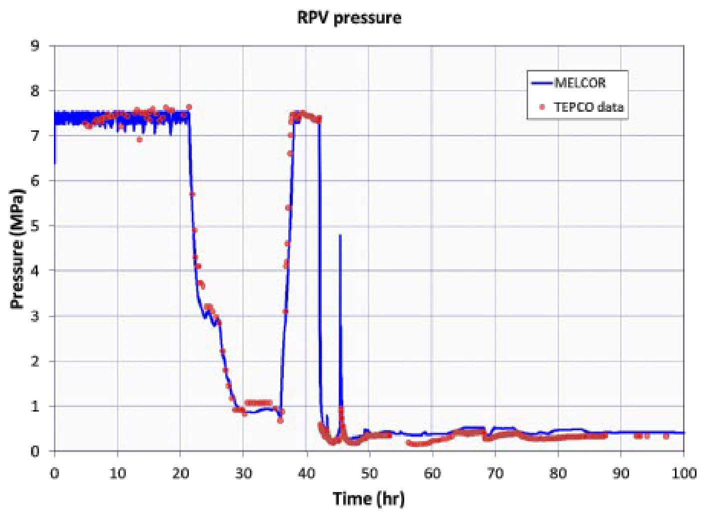
Protecting People and the Environment

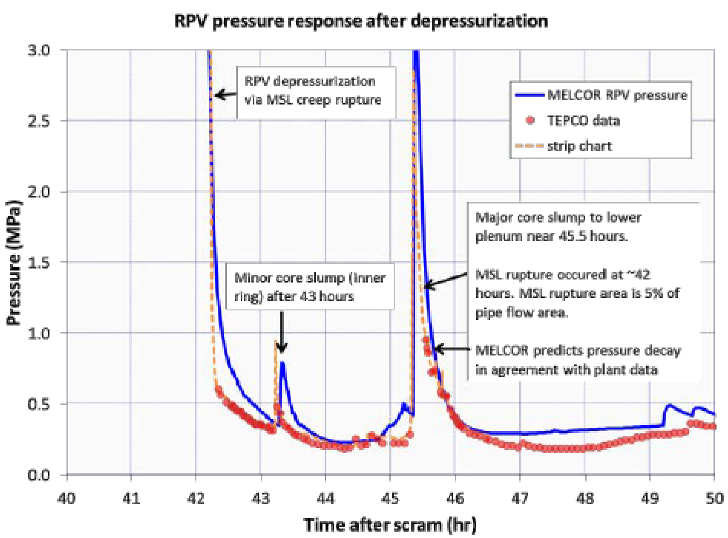
Uncertainty Analysis Applications

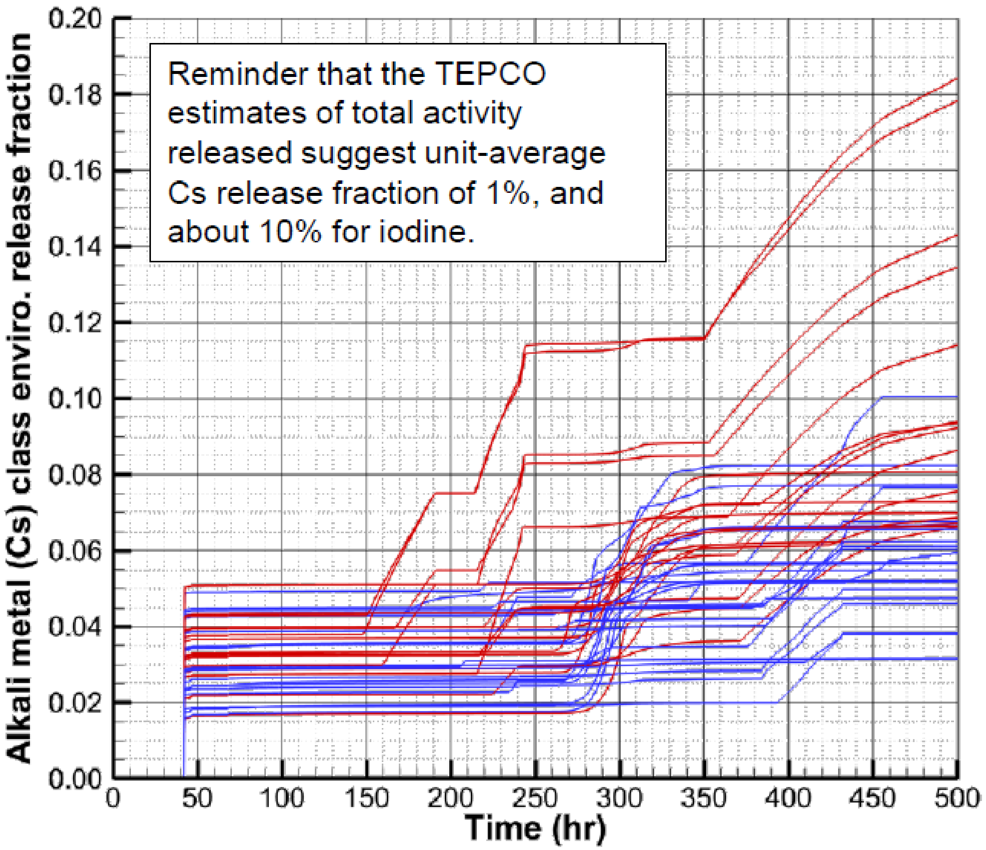




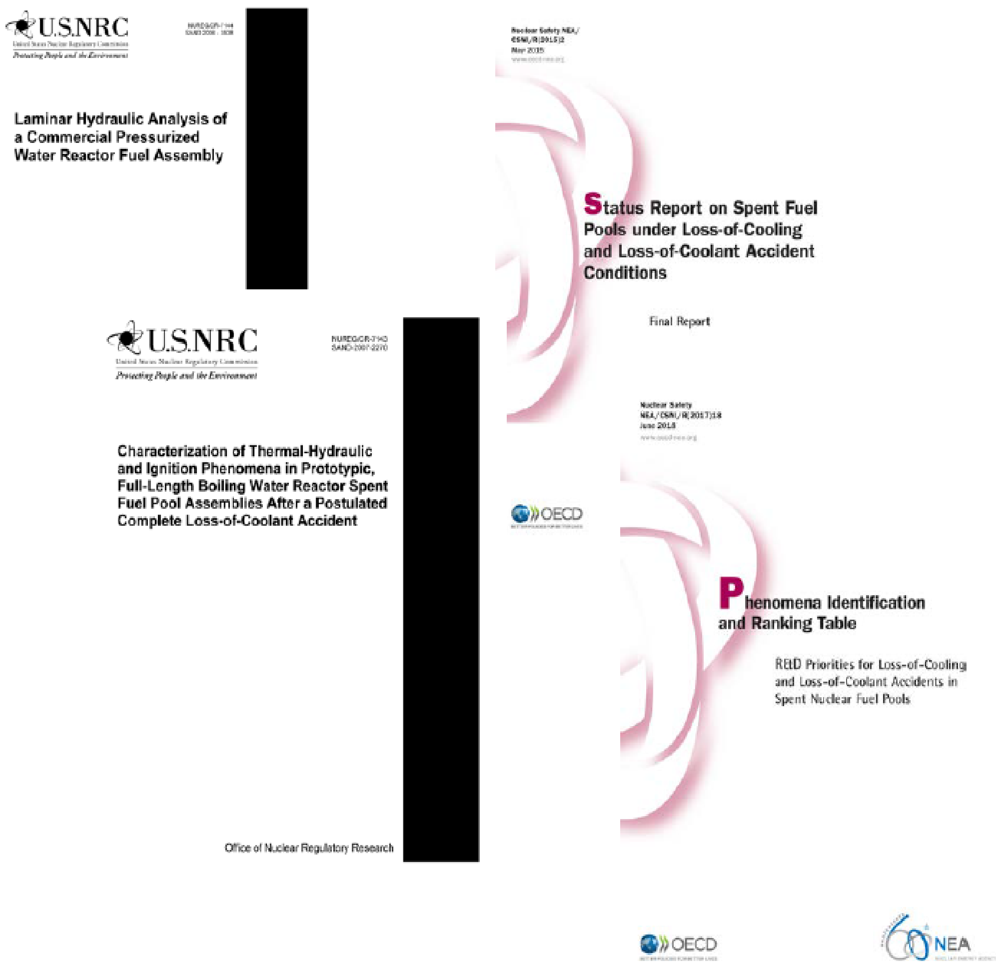
Fukushima Forensics (Unit 3)







MELCOR Spent Fuel Pool Modeling



Convective Heat Transfer Surfaces:

Ring 1 — Clad, Canister / Water Rods, Rack

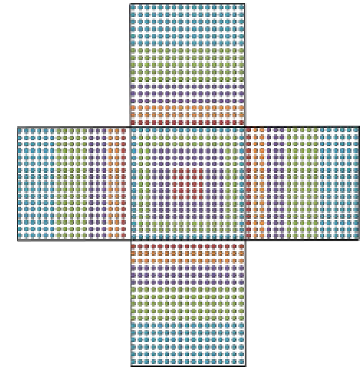
Ring 2 — Clad, Canister / Water Rods, Racks

Radiative Heat Transfer Flow Path:

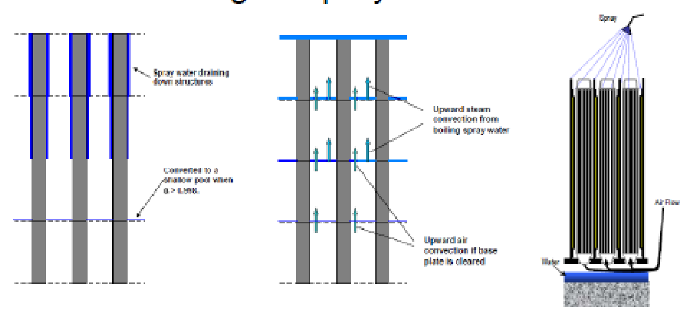
Ring 1: Fuel → Clad → Canister / Water Rods → Rack →

Ring 2: Canister / Water Rods → Clad → Fuel; Canister / Water Rods → Racks

Multiple fuel rod components in the center assembly (Ring 1) and four peripheral assemblies (Ring 2)



Integral Spray Model

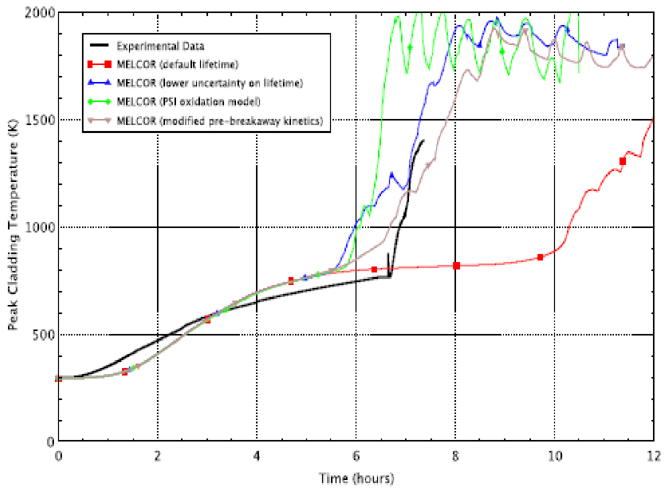


MELCOR SFP Modeling Basis

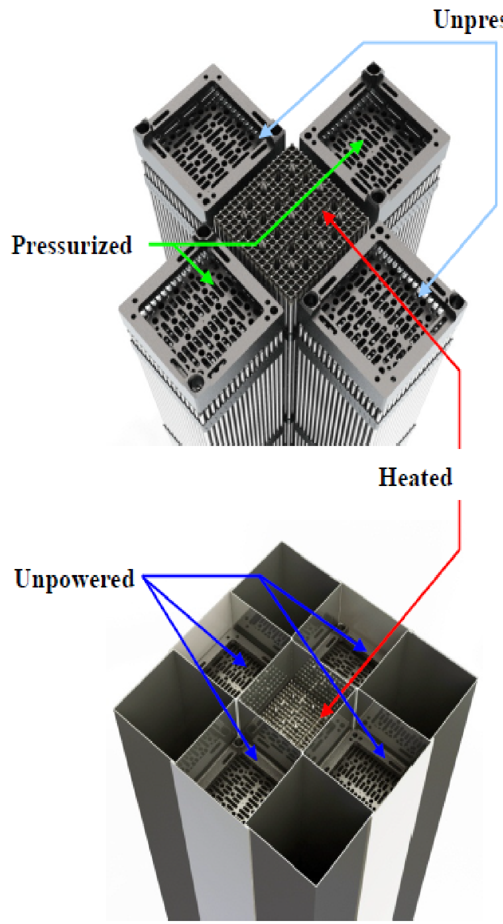
U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NUREG/CR-7216

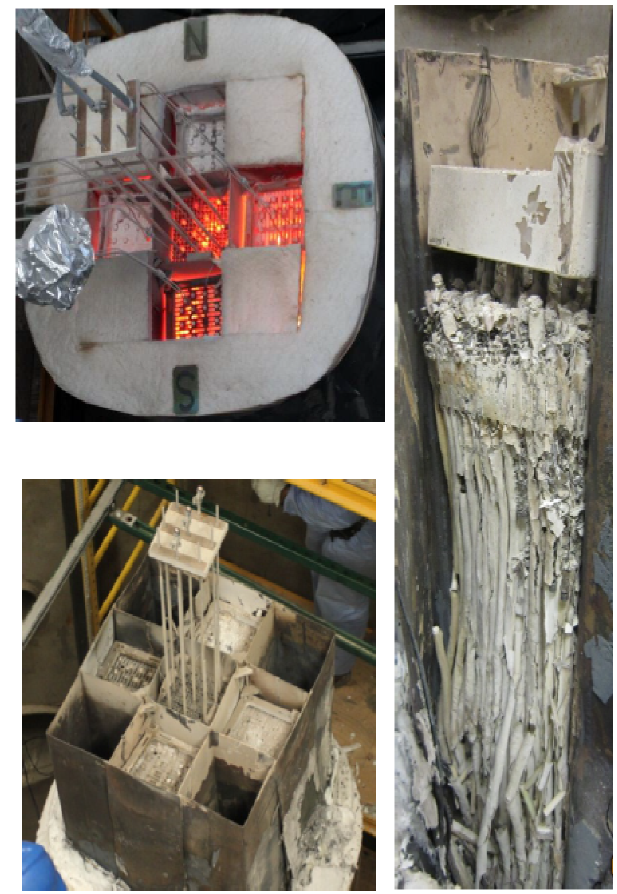
Spent Fuel Pool Project Phase II:
Pre-Ignition and Ignition Testing of a 1x4 Commercial 17x17 Pressurized Water Reactor Spent Fuel Assemblies under Complete Loss of Coolant Accident Conditions



Office of Nuclear Regulatory Research

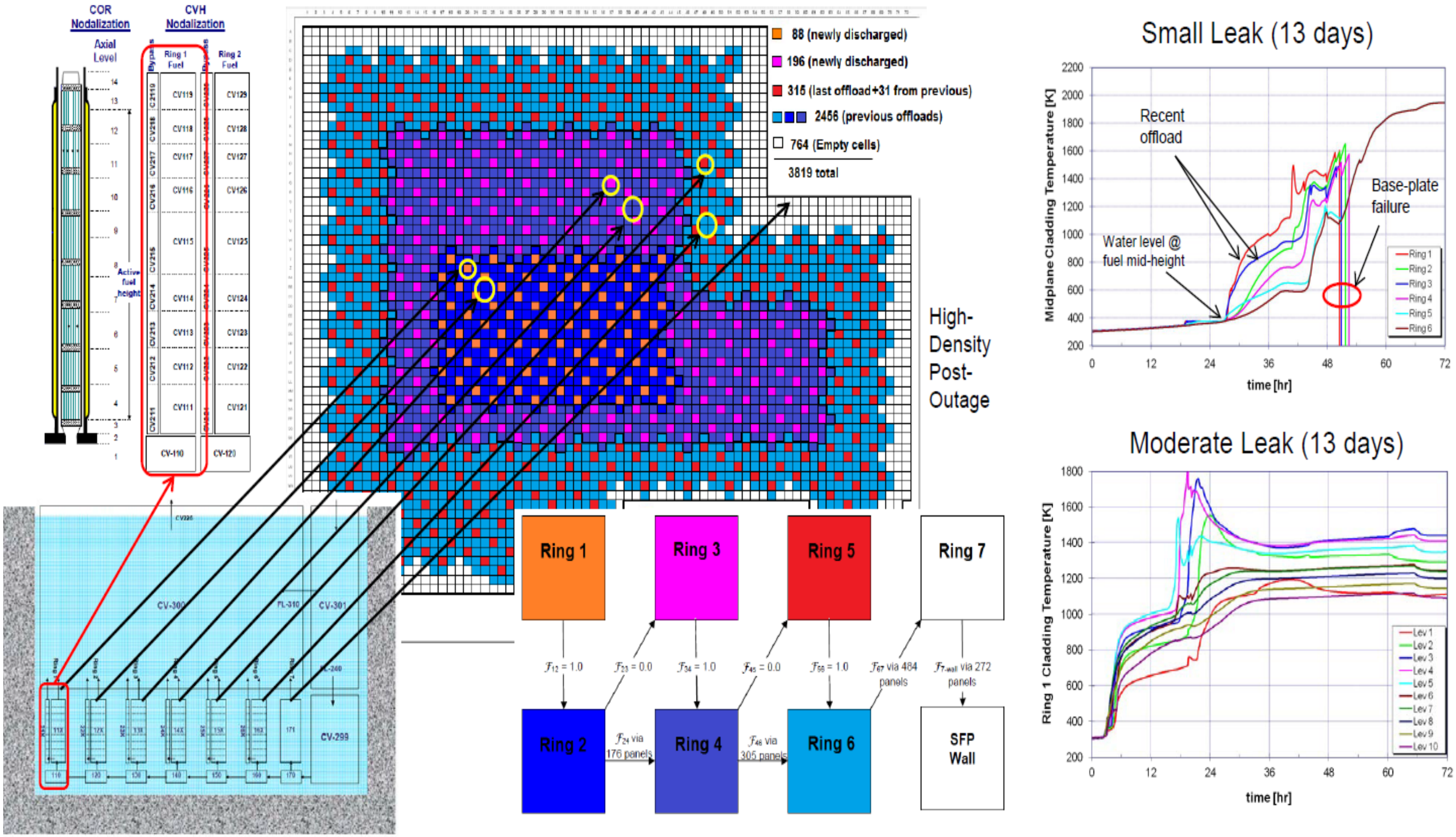


Before



After

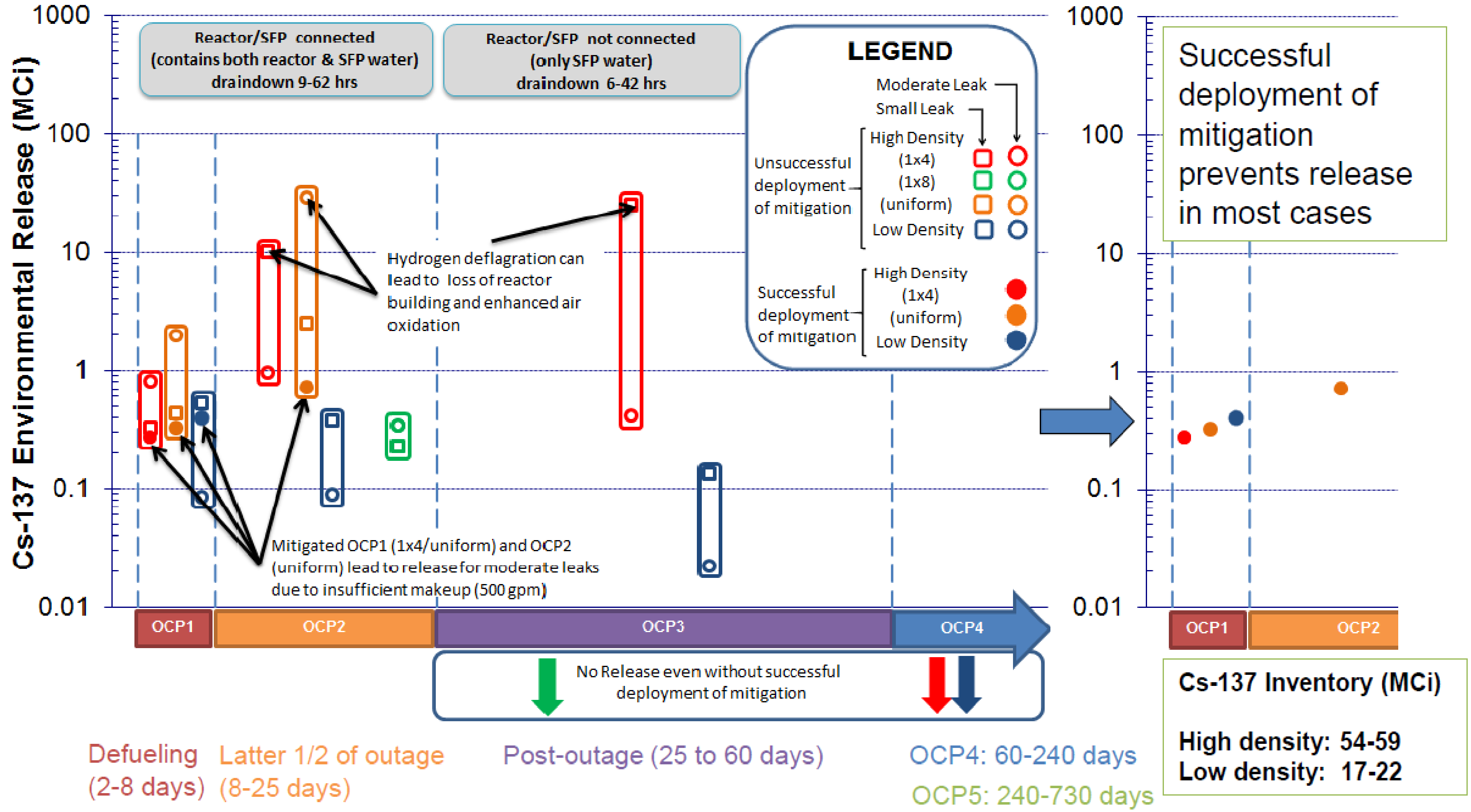
MELCOR SFP Model (NUREG-2161)



BREAK

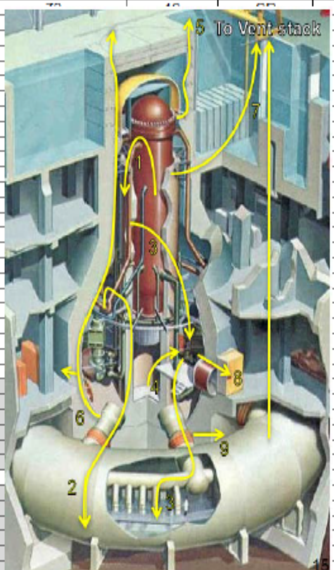
MELCOR Results (NUREG-2161)

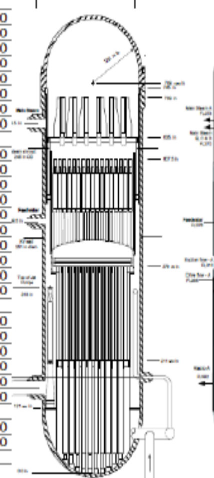
Cases that lead to release (OCP1/2/3)

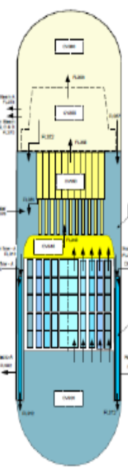


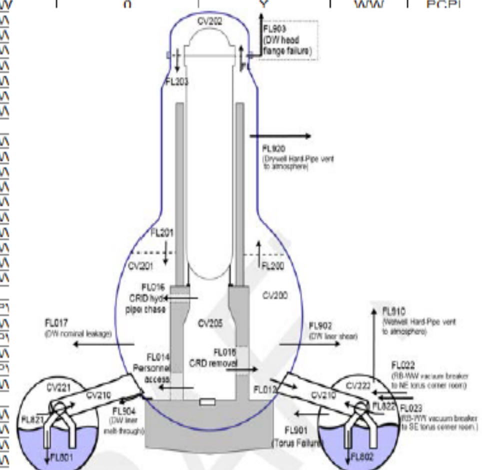
Containment Protection and Release Reduction (NUREG-2206)

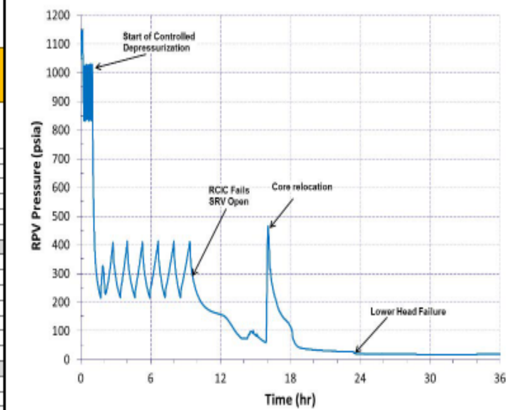
		Pre Core Damage						Post Core Damage				
		RPV Pressure control	RCIC Operation				Anticipatory Venting	Flex Operation		SRV Operation	Venting	
		Availability (hr)	RCIC Availability (hr)	RCIC Suction	Failure Temp (F)	Open SRV after RCIC fails	Setpoint (psig)	Injection @ LH failure	WW Level Control Injection @ 21' (gpm)	Allow SRV stuck open failure?	Location	Setpoint (psig)
Option	Case											
1/2A	1	72	16	SP	230	N	15	-	-	Y	WW	PCPL
1/2A	1S1	72	16	SP	230	N	5	-	-	Y	WW	PCPL
1/2A	2	72	16	SP	230	N	15	-	-	Y	WW	PCPL
1/2A	3	4	4	SP	230	N	15	-	-	N	WW	PCPL
1/2A	4	72	16	SP	240	N	15	-	-	Y	WW	PCPL
1/2A	5	72	16	CST	230	N	15	-	-	Y	WW	PCPL
1/2A	6	72	16	SP	230	N	15	-	-	Y	WW	PSP
3A	7	72	16	SP	230	N	15	RPV	0	Y	WW	PCPL
	7dw	72	16	SP	230	N	15	RPV	0	Y	DW	PCPL
3A	10	72	16	SP	230	N	15	RPV	500	Y	WW/DW	PCPL
3A	11	72	16	SP	230	Y	15	RPV	500	Y	WW/DW	PCPL
4Aii(1)	8	72	16	SP	230	N	15	RPV	throttle	Y	WW	PCPL
4Aii(1)	9	72	16	SP	230	Y	15	RPV	throttle	Y	WW	PCPL
4Aii(1)	12	72	16	SP	230	N	15	RPV	throttle	Y	WW	PSP
4Aii(1)	13	72	16	CST	230	N	15	RPV	throttle	Y	WW	PCPL
4Ai(1)	14				230	N	15	RPV	0	Y	WW	PCPL
4Aiii(1)	15				230	N	15	RPV	throttle	Y	WW	PCPL
4Ai(1)	18				230	N	15	RPV	500	Y	WW/DW	PCPL
4Ai(1)	16				230	N	15	RPV	500	Y	WW/DW	PCPL
3B	21				230			DW	0	Y	WW	PCPL
3B	24				230			DW				
	24dw				230			DW				
4Aii(2)	22				230			DW				
	22dw				230			DW				
4Aii(2)	23				230			DW				
4Aii(2)	25				230			DW				
3B	26				230			DW				
4Ai(2)	27				230			DW				
4Aiii(2)	28				230			DW				
	28dw				230			DW				
4Ai(2)	32				230			DW				
4Ai(2)	30				230			DW				
	30dw				230			DW				
4Aiii(2)	29				230			DW				
	29dw				230			DW				
4Ai(2)	31				230			DW				
	31dw				230			DW				
3A	41				230			RPV				
3B	43				230			DW				
3A	42				230			RPV				
3B	44				230			DW				
4Aii(1)	47				230			RPV				
4Aii(2)	48				230			DW				
3B	45				230			DW				
3B	46				230			DW				
3B	49				-			DW				
4Ai(2)	50				-			DW				
3B	51				230			DW				
3B	52				230	-	15	DW	500	N	DW	15
3B	53	-	16	SP	230	-	15	DW	500	Y	DW	15

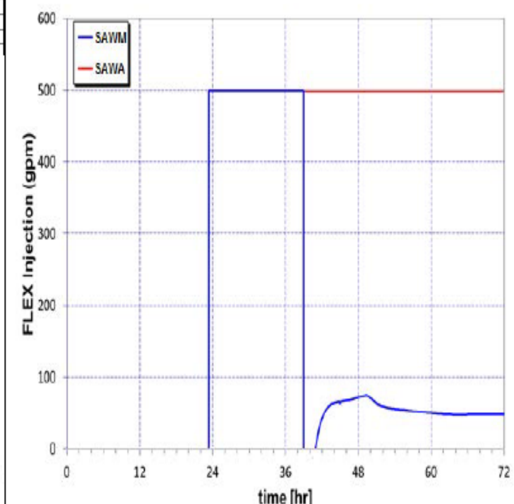




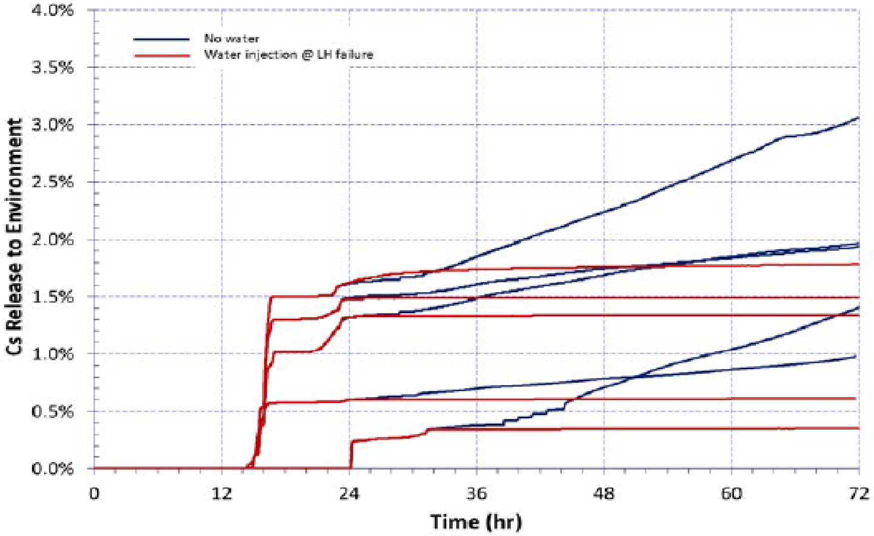






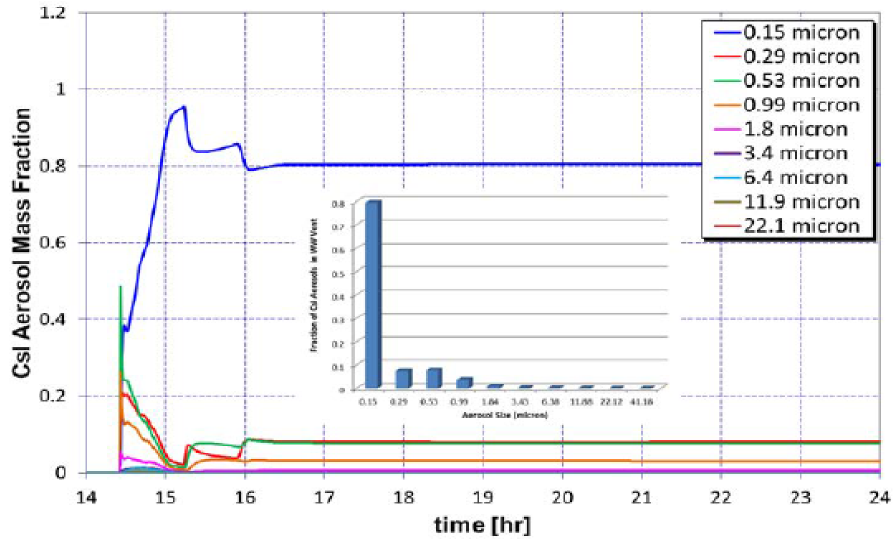


Containment Protection and Release Reduction (NUREG-2206) - Mark I Results

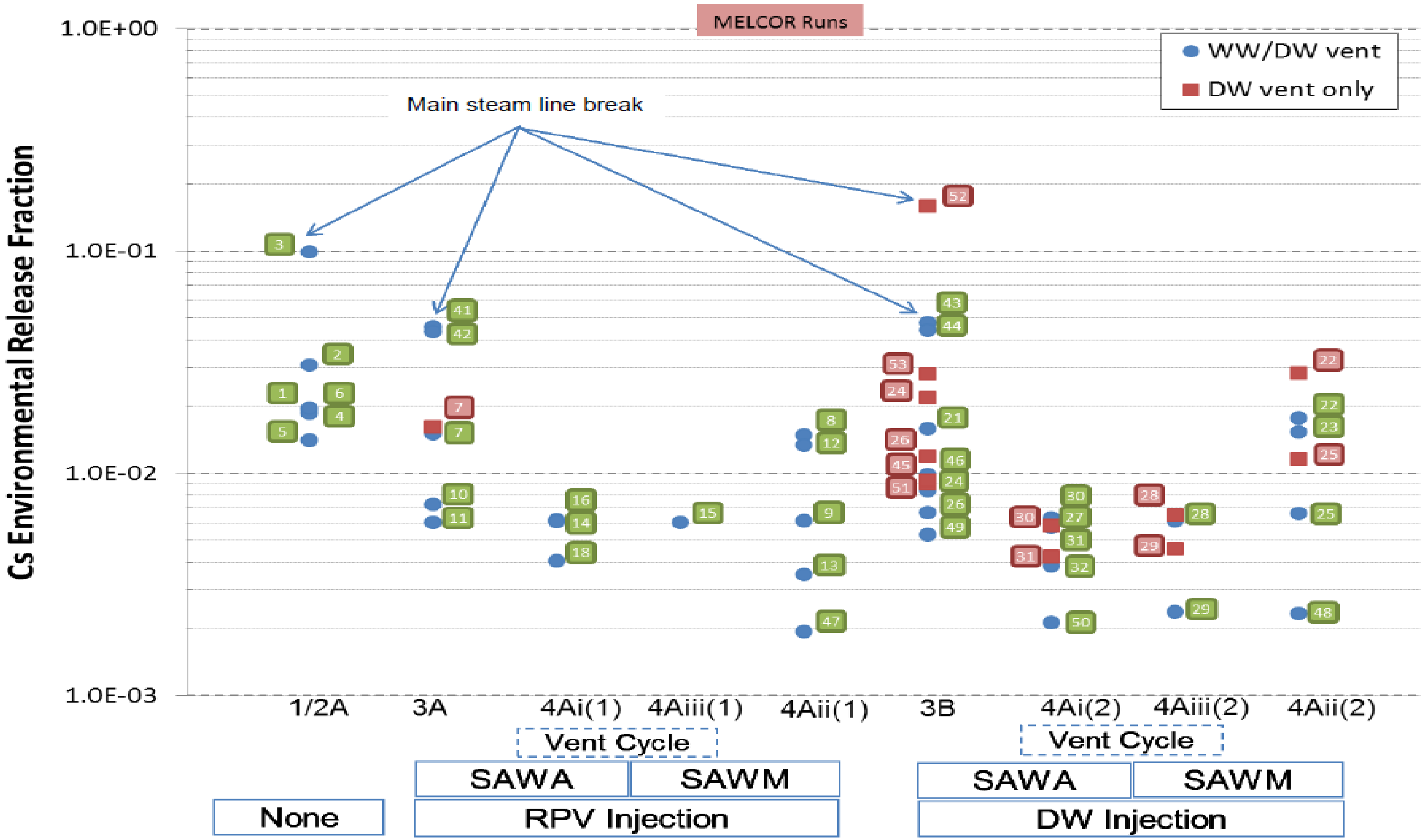


Water addition at lower head failure has the benefit of mitigating further release, but does not affect the release at the time of venting

Particle size distribution dominated by very small aerosols at the time of venting



Containment Protection and Release Reduction (NUREG-2206) - Mark I Results



ACRS meeting on Integration of Source Term Activities in Support of Advanced Reactor Initiatives, 02/17/2022

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Summary

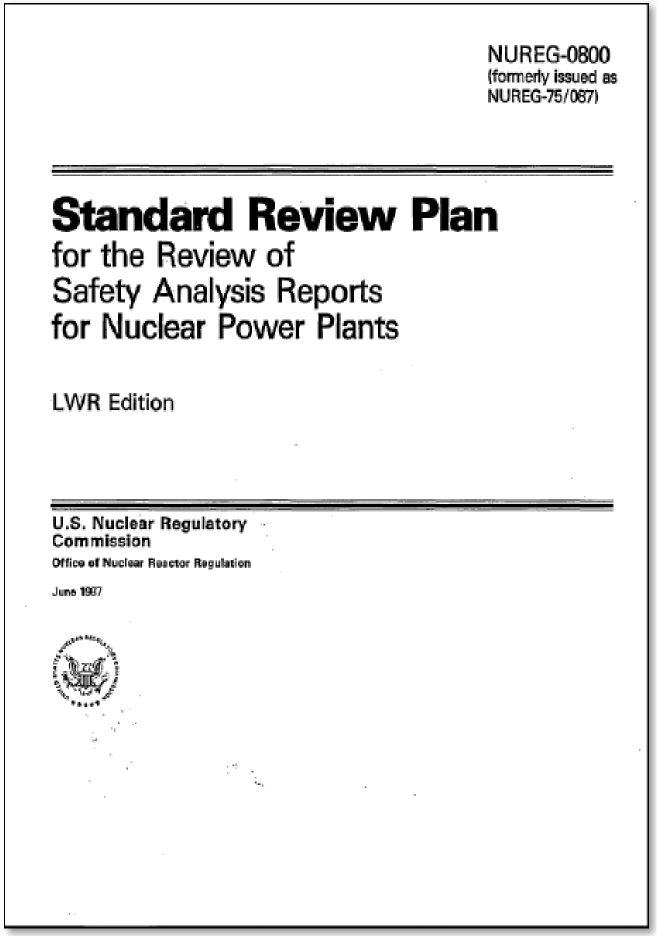
- Decades of experimental and analytical research in severe accident progression and source term
- Validated state-of-practice MELCOR code ready for application to a wide variety of nuclear technologies including advanced designs
- MELCOR has been an essential tool for resolving safety issues and informing regulatory decision making

MELCOR application to new reactors

Standard Review Plan

Staff independent analysis

- Independent assessment of plant response and source term
- Scenarios from the PRA
- Engage with the applicant to resolve differences with the applicant's analysis


NUREG-0800
(formerly issued as
NUREG-75/087)

Standard Review Plan
for the Review of
Safety Analysis Reports
for Nuclear Power Plants

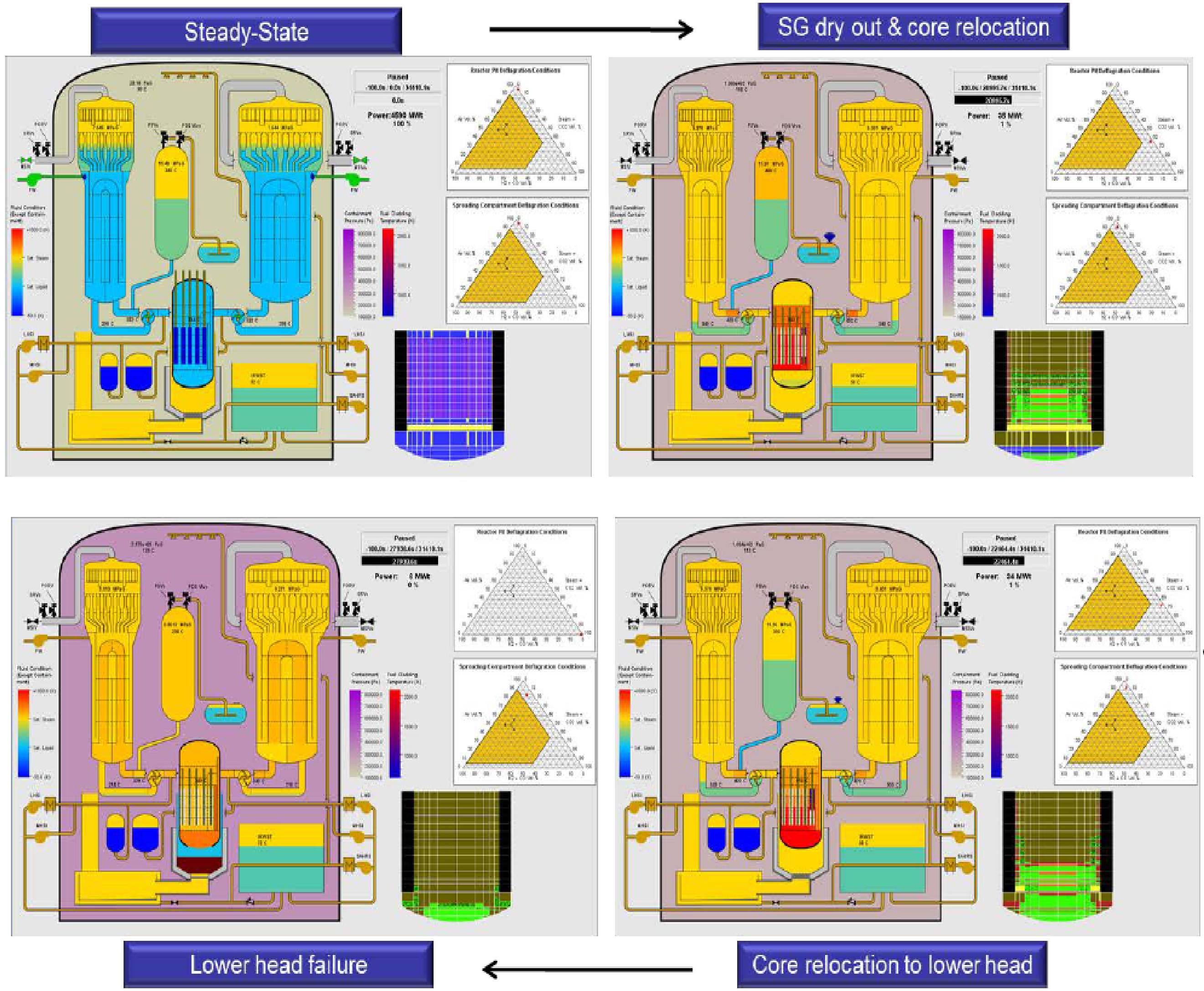
LWR Edition

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation

June 1987

Large LWRs

- ABWR
- AP-600
- System 80+
- AP-1000
- EPR
- APWR
- ESBWR
- APR-1400



SMRs

- NuScale
- mPower
- Westinghouse SMR
- BWRX-300

[redacted]

RES/FSCB 2019-01

Independent MELCOR Confirmatory Analysis for NuScale Small Modular Reactor

S. Campbell
H. Esmaili
J. Schaperow[1]

April 2019

Fuel and Source Term Code Development Branch
Division of Systems Analysis
Office of Nuclear Regulatory Research
United States Nuclear Regulatory Commission

[1] PRA and Severe Accidents Branch, Division of Safety Systems and Risk Assessment, Office of New Reactors, U.S. NRC

[redacted]

NuScale

- Applicant-developed source term for demonstrating EAB/LPZ dose criteria met
  - Replaced RG 1.183 source term
  - MELCOR, STARNAUA
- NRC independent analysis
  - MELCOR, RADTRAD

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Accident Source Term Methodology Topical Report Staff Review

NuScale Design Certification Application Review

Presentation to the ACRS Subcommittee

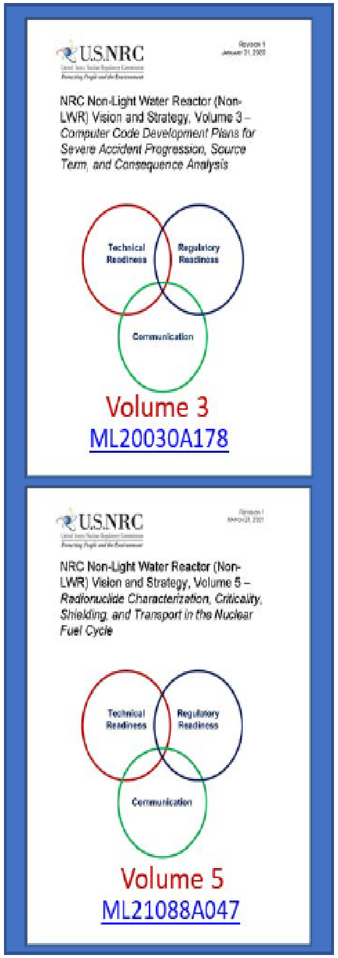
November 20, 2019

SCALE/MELCOR non-LWR source term demonstration project

Outline

- NRC strategy for non-LWR source term analysis
- Project objectives
- Public workshops
- Sample results
  - Heat pipe reactor (HPR)
  - High-temperature gas-cooled reactor (HTGR)
  - Pebble-bed molten-salt-cooled reactor (FHR)
- Summary

NRC strategy for severe accident analysis



U.S.NRC

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 3 – *Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis*

Volume 3
ML20030A178

U.S.NRC

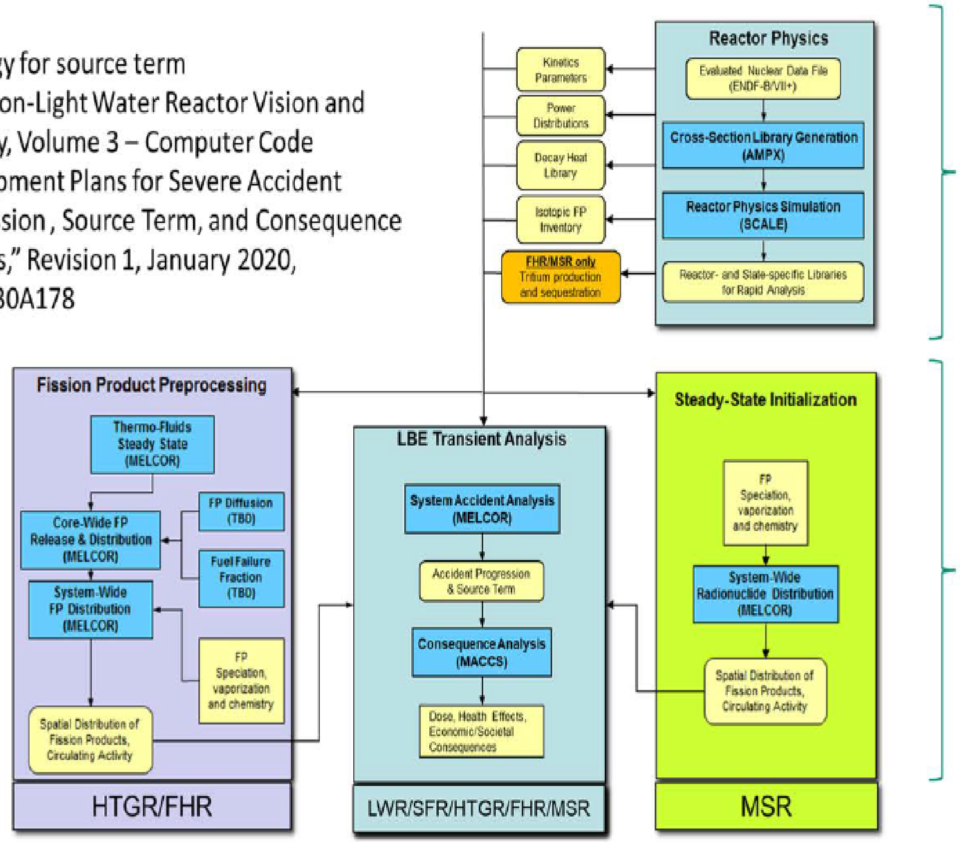
NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 5 – *Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle*

Volume 5
ML21088A047

Evaluation Model and Suite of Codes

Code strategy for source term

"NRC Non-Light Water Reactor Vision and Strategy, Volume 3 – Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis," Revision 1, January 2020, ML20030A178



scale

MELCOR

Project objectives

- Understand severe accident behavior and provide insights for regulatory guidance
- Facilitate dialogue on staff's approach for accident progression and source term
- Demonstrate use of SCALE and MELCOR
  - Identify accident characteristics and uncertainties
  - Develop publicly available input models for representative designs

Scope

Full-plant models for representative non-LWRs

- Heat pipe reactor – INL Design A
- High-temperature gas-cooled reactor – PBMR-400
- Pebble-bed molten-salt-cooled – UCB Mark 1
- Molten-salt-fueled reactor – MSRE
- Sodium-cooled fast reactor – ABTR

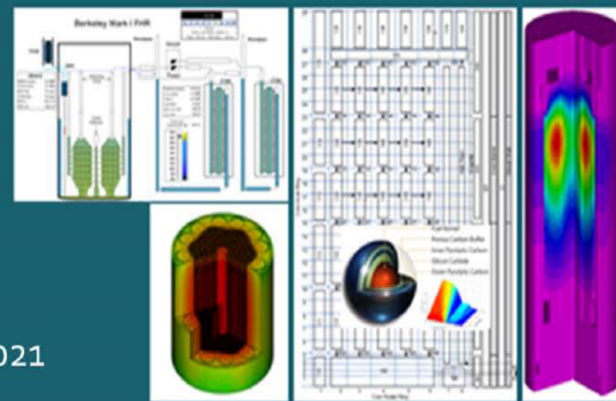
Approach

1. Use SCALE to calculate core decay heat, radionuclide inventory, reactivity feedback
2. Build MELCOR full-plant input model
3. Select accident scenarios
4. Perform MELCOR simulations for the selected scenarios
5. Public workshops to discuss the modeling and sample results

Public Workshops

Public Workshop: SCALE/ MELCOR Non LWR Source Term Demonstration Project

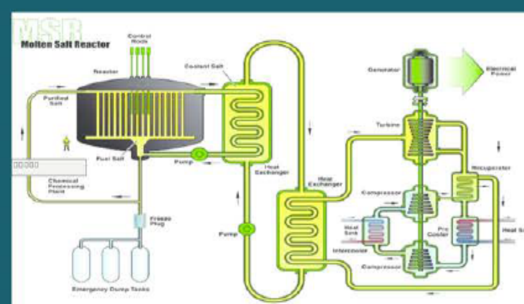
Heat pipe reactor – June 29, 2021
Gas cooled reactor – July 20, 2021
Pebble bed molten-salt-cooled reactor – Sept 14, 2021

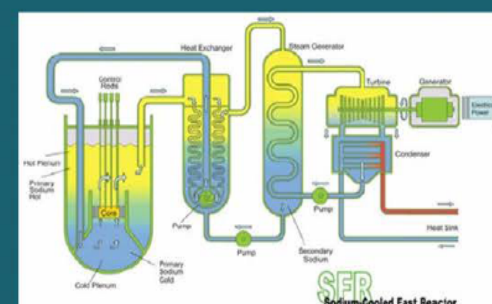


For More Information

Coming in 2022

Molten-salt fueled reactor
Sodium-cooled fast reactor



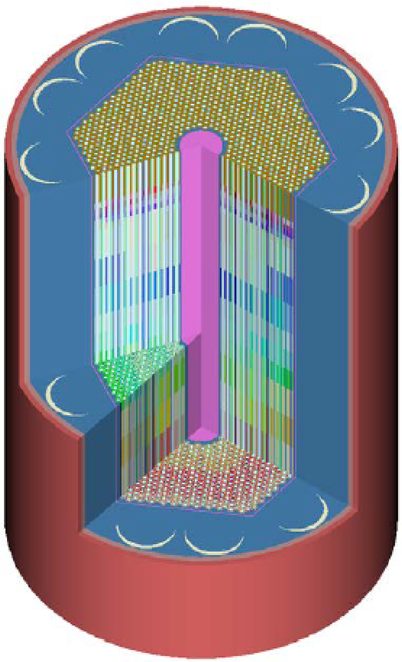


https://www.nrc.gov/reactors/new-reactors/advanced/details.html#non-lwr-ana-code-dev

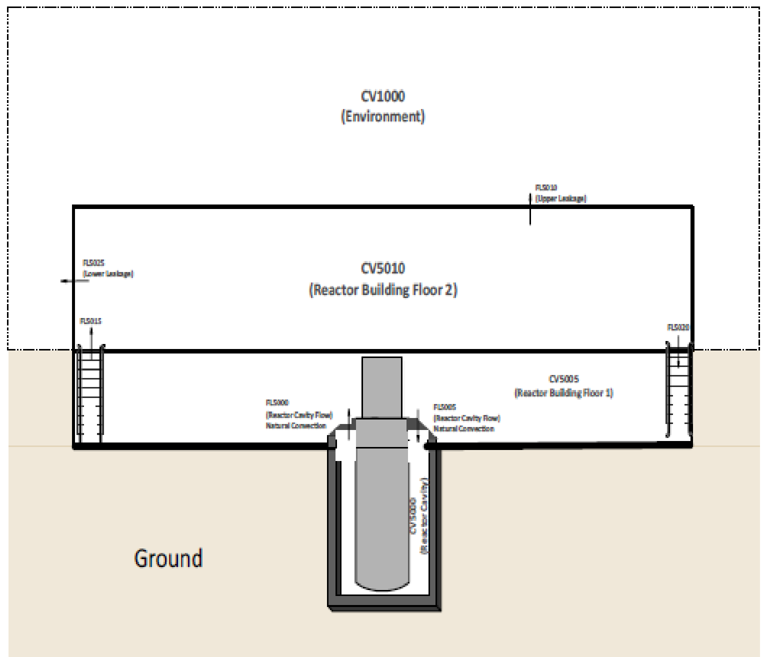
Sample Results

calculations by ORNL and SNL

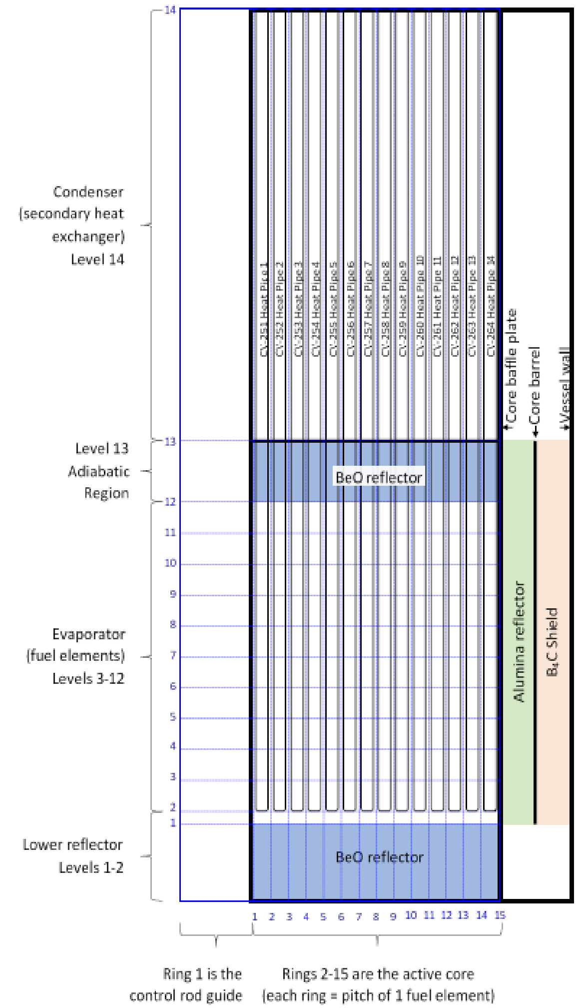
HPR model (INL Design A)



SCALE Model



MELCOR Model



HPR – reactivity addition accident with delayed scram

The control drums start rotating at t=0 sec, which leads to an increase in the core power over 0.9 hr

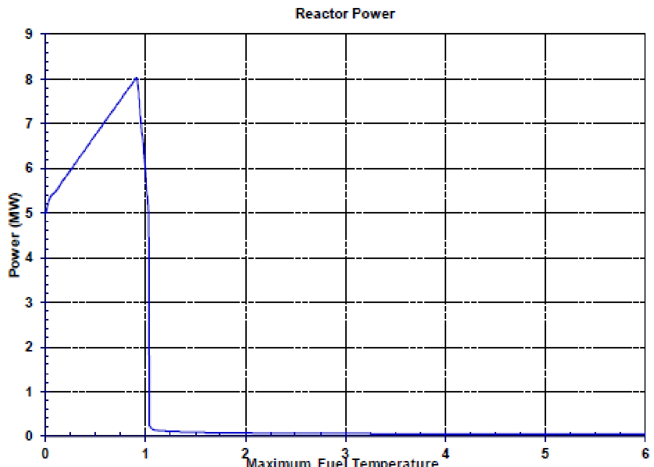
- Negative fuel temperature reactivity feedback limits the rate of power increase

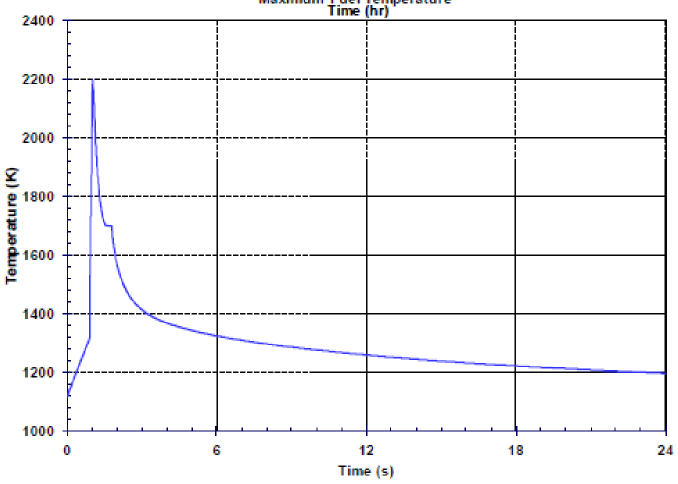
The core steadily heats until the maximum heat flux location reaches the boiling limit

- The heat transfer rate is limited above the boiling limit, which leads to a rapid heatup rate
- The SS cladding is assumed to fail at 1650 K (just below its melting point), which starts the fission product releases into the reactor
- The reactor is assumed to trip at 2200 K

Radial heat dissipation and heat loss to the reactor cavity passively cools the core

- No active heat removal (secondary system trips and isolates)





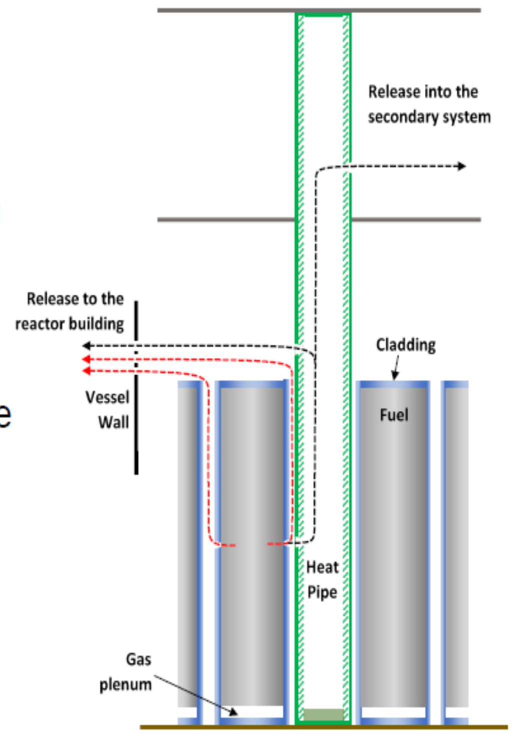
HPR – reactivity addition accident with delayed scram

Cladding failure at 1650 K resulting in fission product release

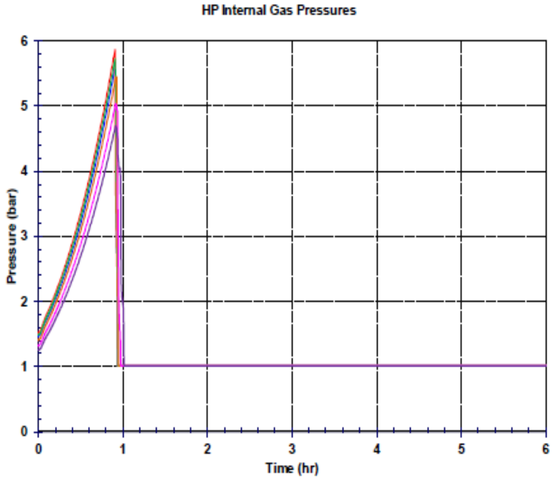
- Heat Pipes (HP) that exceeded the boiling limit rapidly heat to cladding failure (1650 K)
- ~20% of the 1134 HPs and fuel elements failed
- HP depressurization on failure drive release from the vessel

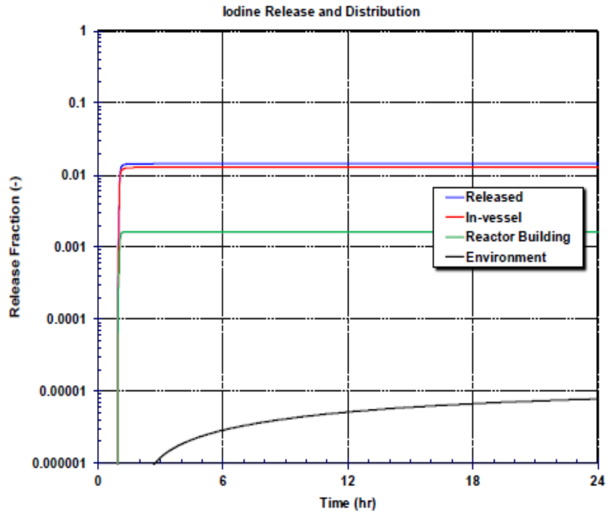
Iodine releases also depend on time at temperature

- Fuel release – 1.4% of core inventory
- Environmental release – 0.0008% of core inventory

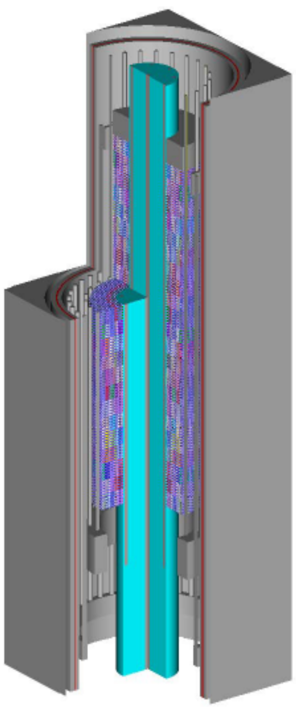


- Vessel leakage is 1.6 in^2
- Building leakage is 1.8 in^2

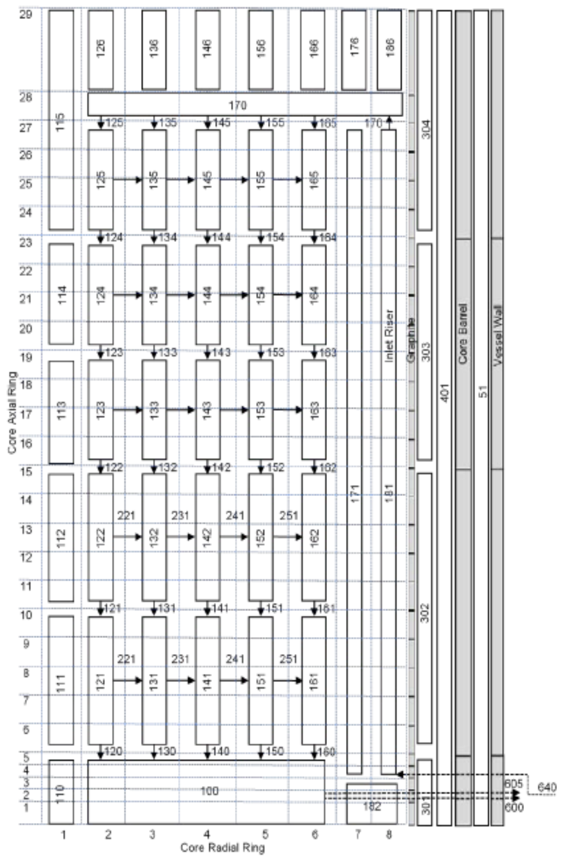




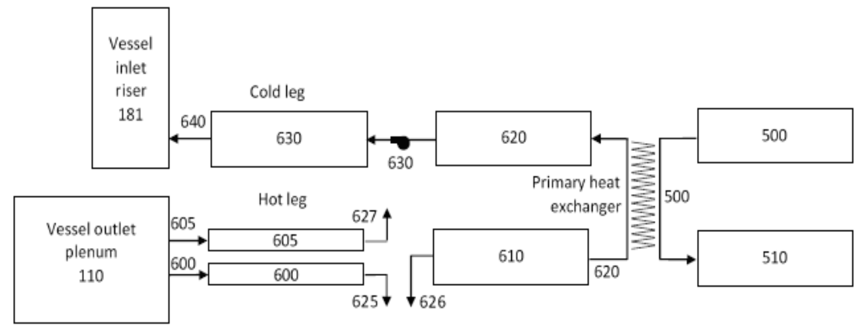
HTGR model (PBMR-400)

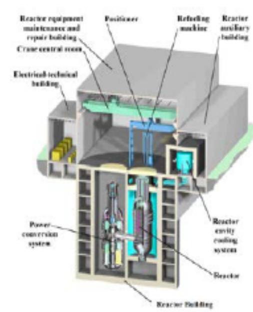


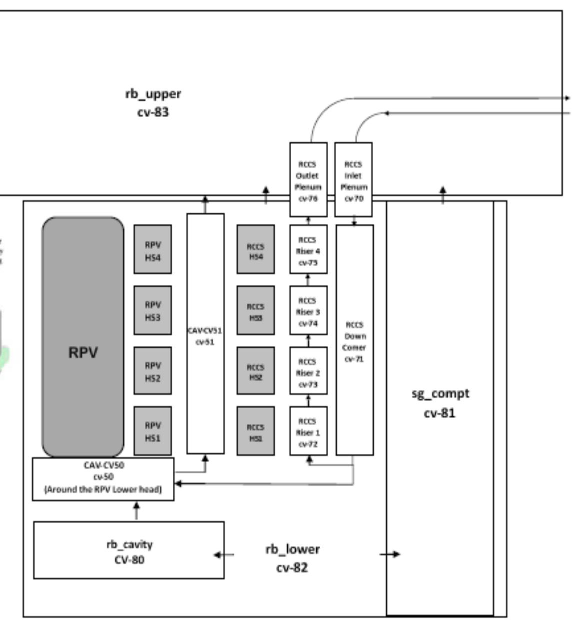
SCALE Model



MELCOR Model







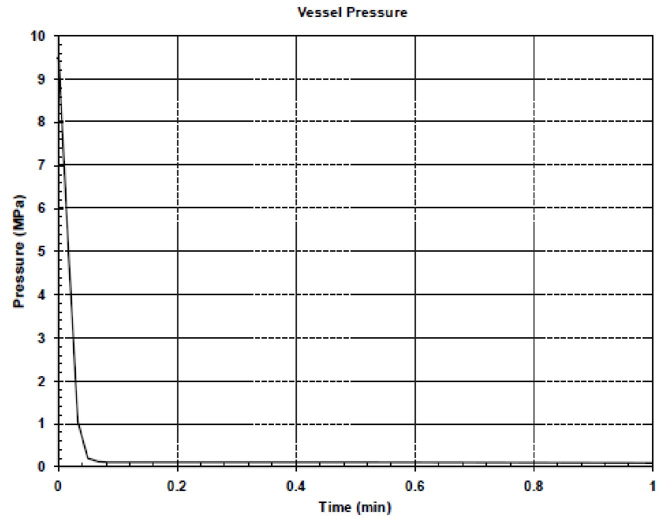
HTGR – loss of coolant accident

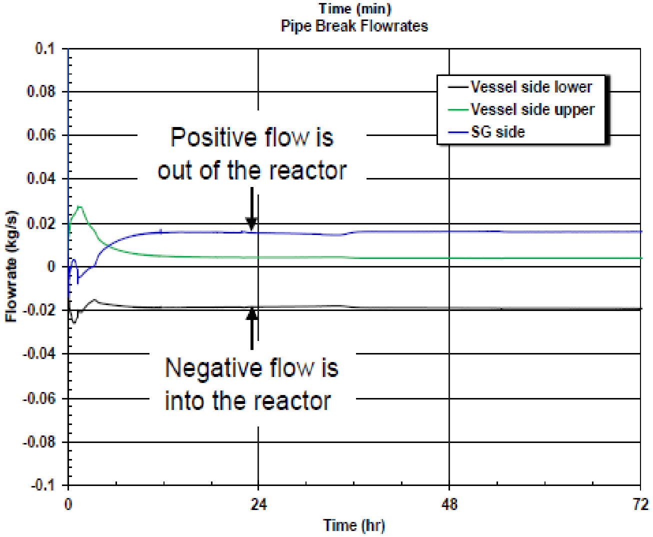
Following pipe break

- Control rods insert to terminate fission
- The vessel depressurizes in seconds as the high-pressure helium escapes out both sides of the broken pipe
- Peak velocity in the pebble bed is 45 m/s (normal flow rate is 11-18 m/s)

Counter-current flow established on the vessel side of the pipe break

- Hot gases from the exit plenum escape on the top side of the broken hot leg pipe and cooler gases enter along the bottom of the pipe





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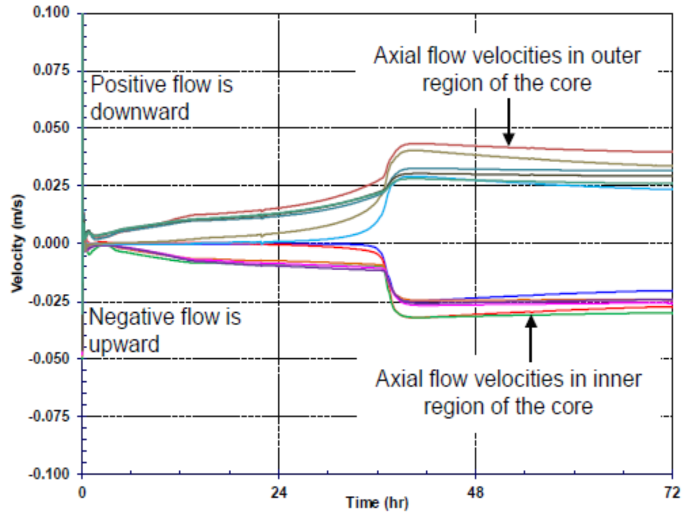
HTGR – loss of coolant accident

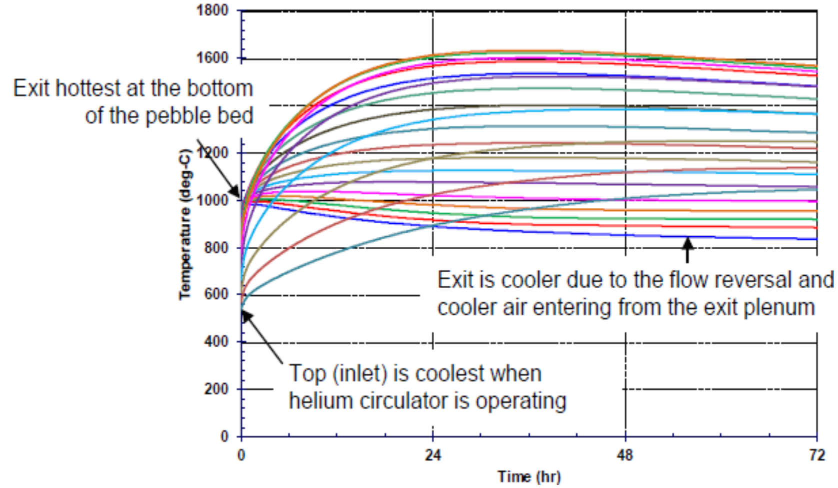
In-vessel natural circulation flow after blowdown

- Upward flow in the inner region of the core where the fuel temperatures and decay power heating are higher
- Downward flow in the outer region of the core where the fuel temperatures and decay power heating are lower
- Flow increases when the fuel starts to cool



The fuel temperatures in the inner region of the pebble bed shift from cooler at inlet and hot at the outlet due to the flow reversal

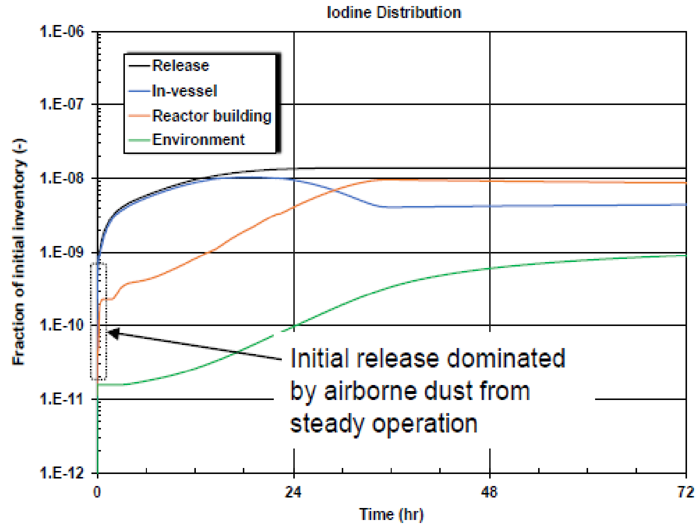
- The axial fuel temperatures are affected by the local decay heat power (highest in the center) and the flow direction
  - During normal operation, the fuel at the exit (bottom) is the hottest
  - The exit becomes the coolest location (low power and cooler gases entering from the exit plenum)

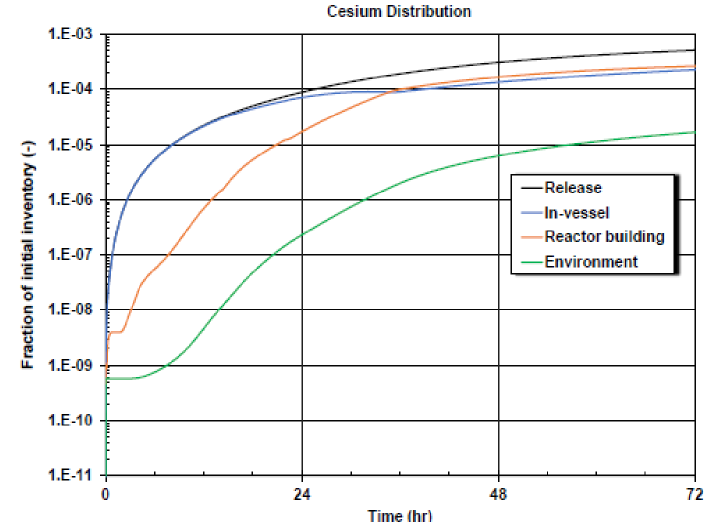


HTGR – loss of coolant accident

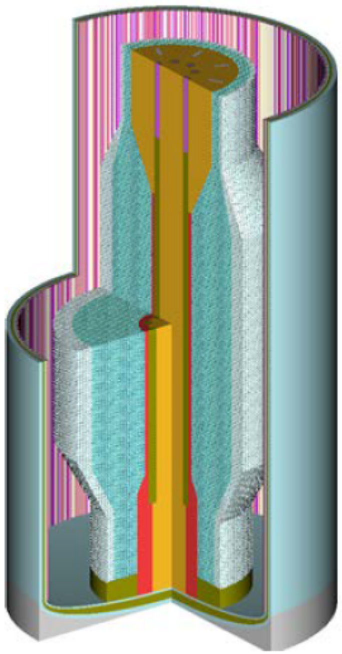
The impact of the low TRISO failure fraction leads to small releases

- Iodine diffusivity assumed to be same as krypton
- Assumes most iodine reacts with cesium
- Larger cesium release due its the higher diffusivity
- Ag release to the environment is 1.2×10^{-3} (highest diffusivity)



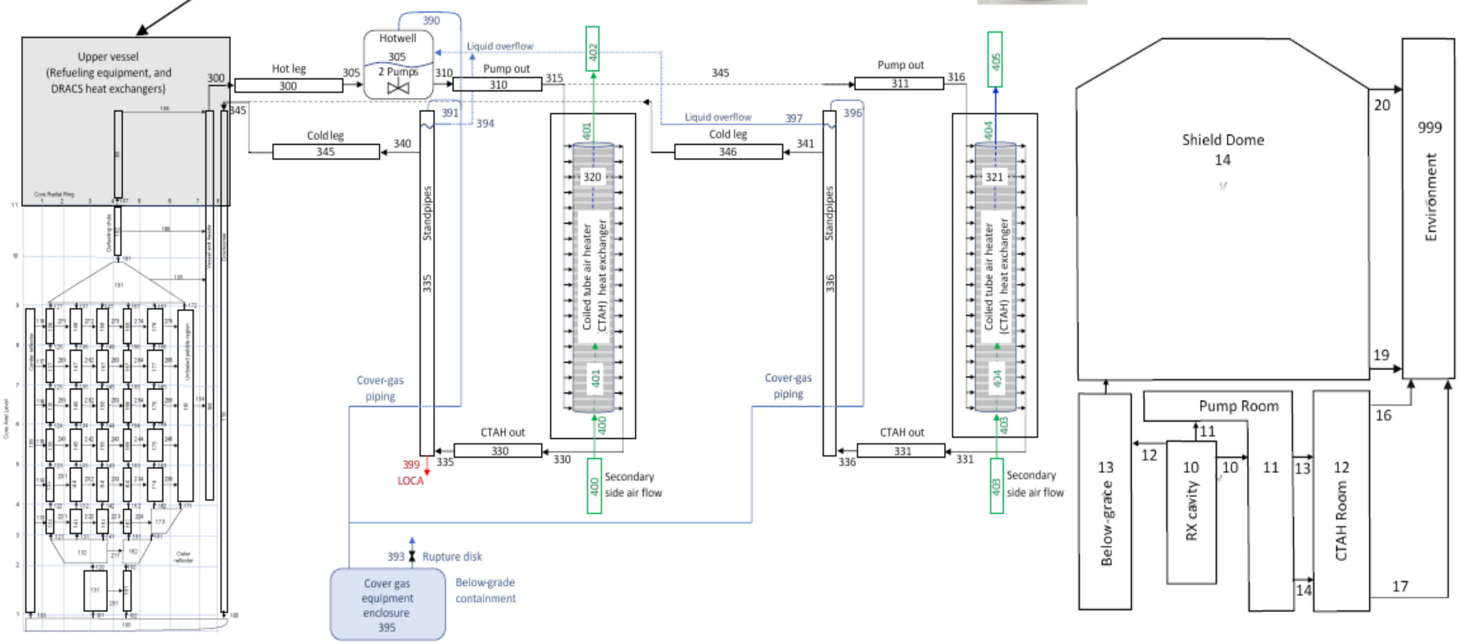


FHR model (UCB Mark 1)



SCALE Model

DRACS (direct reactor auxiliary cooling system)



MELCOR Model

FHR – ATWS

Loss-of-onsite power with failure to SCRAM

- Salt pumps shut off
- Reactor fails to SCRAM
- Secondary heat removal ends
- 0 to 3 trains of DRACS operating

Includes preliminary analysis with xenon transient

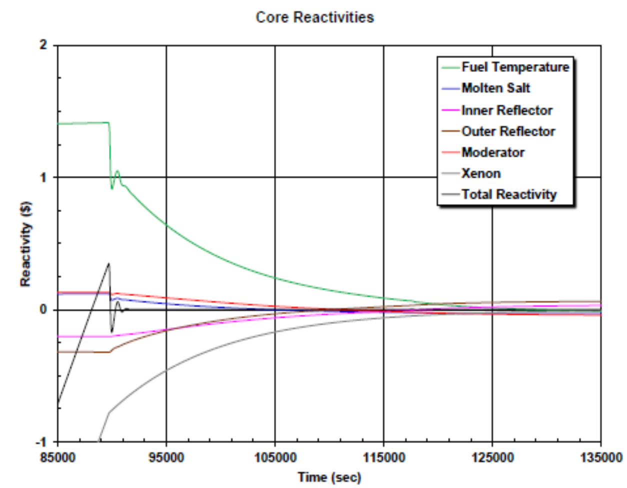
- Guided by ORNL calculations
- Xenon reactivity feedback model being implemented into MELCOR

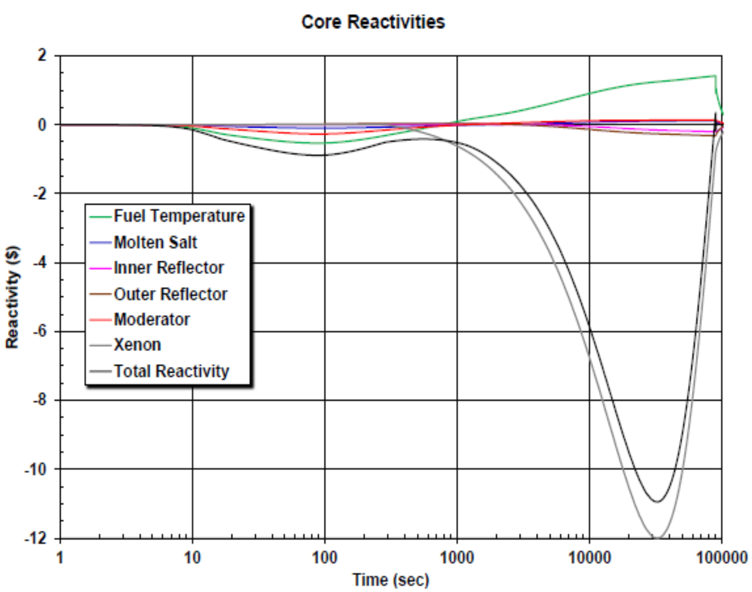
FHR – ATWS

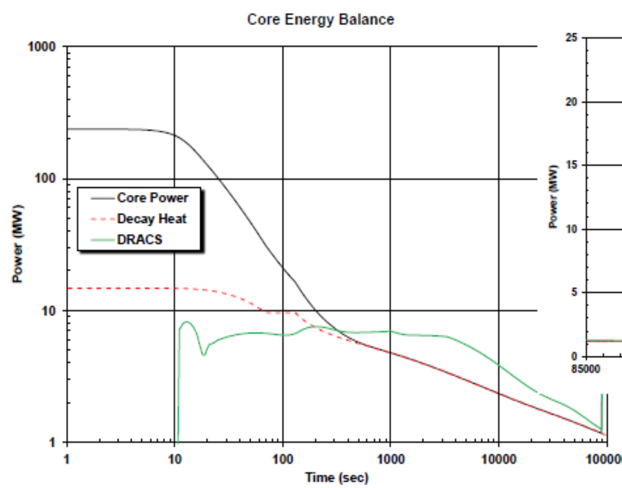
Initial fuel heatup has strong negative fuel and moderator feedback that offsets positive reflector feedbacks

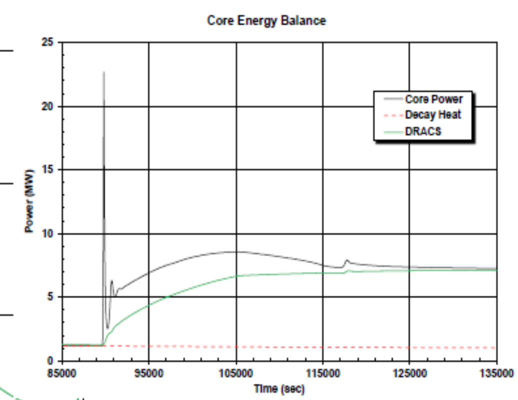
Strong negative xenon transient feedback

3xDRACS exceeds core power after 330 s









FHR – ATWS with variable DRACS

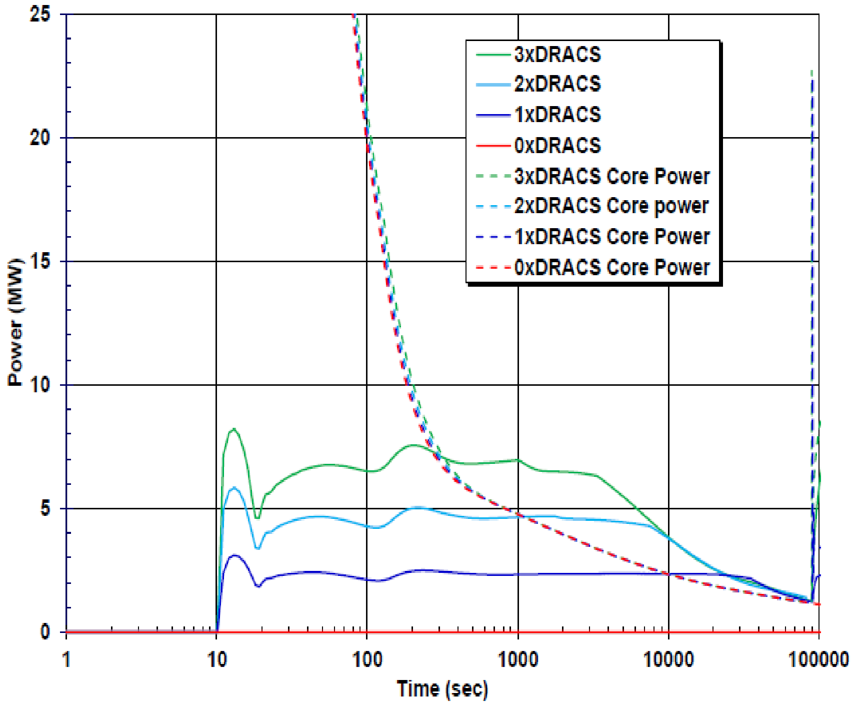
Early power decrease to decay heat level is similar for all cases

- 1xDRACS and 2xDRACS cases exceed decay heat later

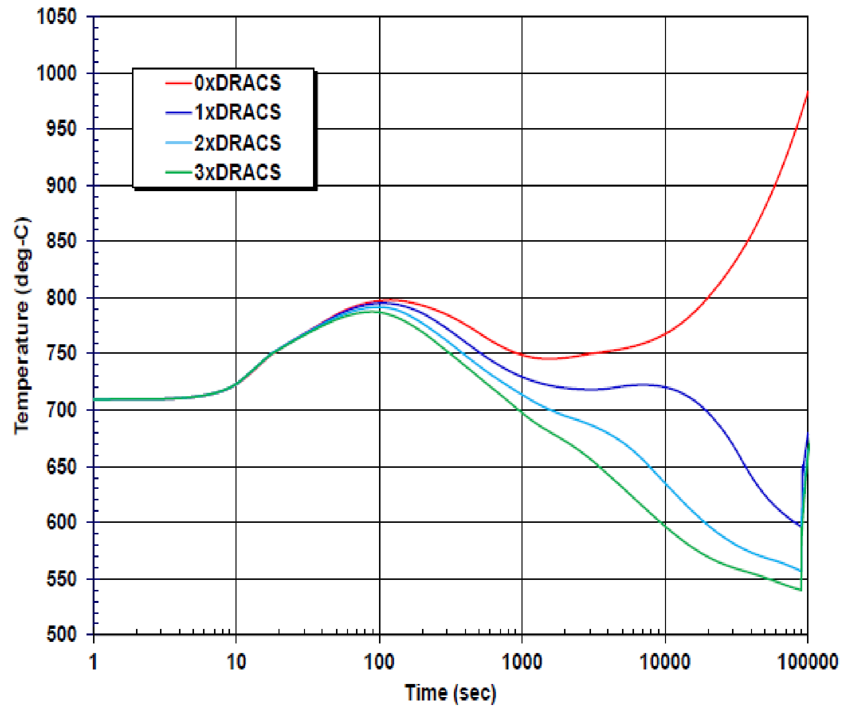
Fuel temperatures cool down according to DRACS heat removal rate

- 0xDRACS peak fuel temperature = 990 ºC at 10^5 s (T_{sat}~ 1350 ºC)

Core Power abd DRACS Heat Removal



Peak Fuel Temperatures



Summary

- Demonstrated use of SCALE and MELCOR for safety analysis for 3 classes of non-LWRs
  - Working on demonstrations for 2 more classes
- Simulated the entire accident starting with the initiating event
  - system thermal hydraulic response
  - fuel heat-up
  - heat transfer through the reactor to the surroundings
  - radiological release
- Evaluated effectiveness of passive mitigation features

References (www.nrc.gov)(1/2)

- **NUREG-2161**, Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor (2014)
- **NUREG-2206**, Technical Basis for the Containment Protection and Release Reduction Rulemaking for Boiling Water Reactors with Mark I and Mark II Containments (2018)
- **NUREG/BR-0524**, Cooperative Severe Accident Research Program (CSARP)(2015)
- **NUREG/CR-7143**, Characterization of Thermal-Hydraulic and Ignition Phenomena in Prototypic, Full-Length Boiling Water Reactor Spent Fuel Pool Assemblies After a Postulated Complete Loss-of-Coolant Accident (2013)
- **NUREG/CR-7144**, Laminar Hydraulic Analysis of a Commercial Pressurized Water Reactor Fuel Assembly (2013)

References (www.nrc.gov)(2/2)

- **NUREG/CR-7216**, Spent Fuel Pool Project Phase II: Pre-Ignition and Ignition Testing of a 1x4 Commercial 17x17 Pressurized Water Reactor Spent Fuel Assemblies under Complete Loss of Coolant Accident Conditions (2016)
- **NUREG/CR-7245**, State-of-the-Art Reactor Consequence Analyses (SOARCA) Project: Sequoyah Integrated Deterministic and Uncertainty Analyses (2019)
- **NUREG/CR-7282**, Review of Accident Tolerant Fuel Concepts with Implications to Severe Accident Progression and Radiological Releases (2021)
- **NUREG/CR-7283**, Phenomena Identification Ranking Tables for Accident Tolerant Fuel Designs Applicable to Severe Accident Conditions (2021)
- **SECY-16-0100**, "National Academy of Sciences Study of the Lessons Learned from the Fukushima Nuclear Accident for Improving Safety and Security of U.S. Nuclear Power Plants"

Abbreviations

ATF	accident tolerant fuel
ATWS	anticipated transient without scram
DBA	design basis accident
DW	drywell
EAB	exclusion area boundary
EU	European Union
DRACS	direct reactor auxiliary cooling system
FHR	fluoride salt-cooled high-temperature reactor
HALEU	high-assay low-enriched uranium (fuel)
HBU	high burnup (fuel)
HPR	heat pipe reactor
HTGR	high temperature gas-cooled reactor
INL	Idaho National Laboratory
IRSN	Institut de radioprotection et de sûreté nucléaire (France)
LPZ	low-population zone
MSR	molten salt reactor
NEA	Nuclear Energy Agency
OCP	operating cycle phase
ORNL	Oak Ridge National Laboratory
PBMR	pebble bed modular reactor
PIRT	phenomena identification and ranking table
PRA	probabilistic risk assessment
RPV	reactor pressure vessel
SAWA	severe accident water addition
SAWM	severe accident water management
SFP	spend fuel pool
SFR	sodium-cooled fast reactor
SMR	small modular reactor
SNL	Sandia National Laboratory
SOARCA	State-of-the-Art Reactor Consequence Analysis

NuScale EPZ Sizing Methodology Topical Report, Rev. 2
Light Water SMR Design Certification Source Term Approach
Source Term Approach for Early non-LWR Movers

United States Nuclear Regulatory Commission
Protecting People and the Environment

Accident Source Term in Recent and Near-term Applications

Michelle Hart
NRR/DANU/UTB2

Outline

- SMR and non-LWR accident source terms recent experience
- Emergency planning zone size justification consequence analyses
- Example: SMR design certification source term approach
- Source term approaches for non-LWR early movers

SMR and Non-LWR Accident Source Terms Recent Experience

- SMR topical report reviews and SMR DC application review
- Advanced reactor pre-application interactions, topical report reviews, and license applications
- Source term development contractor reports

Emergency Planning Zone Size Justification Consequence Analyses

- Concept based on NUREG-0396
  - Technical basis for plume exposure and ingestion pathway EPZ radius of ~10 and ~50 miles, respectively
  - Identification of area within which prompt protective actions may be necessary to provide dose savings in the event of a radiological release
- Calculate dose at distance for a spectrum of accidents
  - Analysis includes design basis accidents and severe accidents

Emergency Planning Zone Size Justification Consequence Analyses

- No separate/unique source terms developed especially for EPZ size analysis
  - Re-use source terms and accident release information developed for safety analysis report and PRA

Emergency Planning Zone Size Justification Consequence Analyses

- Methodology to support exemptions to 10-mile requirement
  - Clinch River ESP EPZ size methodology described in SSAR
- Methodology to support plume exposure pathway EPZ size determination on case-by-case basis for reactors <250 MWt
  - NuScale EPZ sizing methodology topical report (under review)
- EPZ size determination required in EP for SMRs and ONTs alternative framework, once issued
  - SECY-22-0001 issued for Commission review and approval
  - Guidance on analysis in appendices to RG 1.242

NuScale EPZ Sizing Methodology Topical Report

- TR-0915-17772, Revision 2, submitted in 2020, currently under review
  - Not part of DC review
  - Applicable to light-water SMRs such as NuScale, although not limited to the NuScale designs
  - Rev. 3 under development
- Analysis methodology to determine plume exposure pathway EPZ size

NuScale EPZ Sizing Methodology Topical Report

- "Source term" refers to fission product release to the environment as a function of time
- Uses source terms from DBAs (DC FSAR Ch. 15) and PRA severe accident scenarios scoped into analysis
  - No separate/unique source terms developed especially for EPZ size analysis
  - Uses CDF from PRA to categorize severe accidents and select accident sequences to evaluate against relevant dose criteria

Example: SMR Design Certification Source Term Approach

- SECY-19-0079, August 16, 2019
  - Describes staff review approach to evaluate accident source terms for both the TR and the NuScale SMR DC application
  - Provides basis for using source term without core damage for environmental qualification

Example: SMR Design Certification Source Term Approach – NuScale TR

- NuScale TR-0915-17565, "Accident Source Term Methodology," Revision 4, February 2020
  - Methods to develop accident source terms are consistent with RG 1.183 guidance for PWRs except for:
    - Core damage source term for Core Damage Event
    - Iodine spike design basis source term (no fuel damage)

NuScale TR: Core Damage Event

- Derive source term from range of accident scenarios that result in significant damage to the core
  - Informed by NuScale SMR PRA
- NuScale-design-specific analyses using MELCOR to be performed by applicant referencing the TR
- Radionuclide transport phenomena
  - Iodine retention in containment based on pH
  - Aerosol natural deposition in containment

NuScale SMR DC Application: Core Damage Event

- Implemented the NuScale TR methodology to determine the core damage source term
- Core inventory calculated using SCALE code
- Scenario selection
  - Based on NuScale SMR PRA, internal events
  - 5 surrogate scenarios
    - Various failures of ECCS, with decay heat removal system available
    - Intact containment

NuScale SMR DC Application: Core Damage Event

- MELCOR used to estimate release timing and magnitude for each scenario
  - Release onset and duration from scenario with minimum time to core damage
  - Core release fractions taken as median of scenarios
- Time-dependent aerosol removal rates calculated using STARNAUA code
  - Design-specific input thermal hydraulic conditions calculated by MELCOR for surrogate scenario with minimum time to core damage

Source Term Approaches for Non-LWR Early Movers

- Kairos Power
  - MST methodology TR (under review)
    - Methodology for applicants to develop event-specific radiological source terms
      - DBAs for siting and safety analysis
      - AOOs and DBEs for LMP
  - Hermes CP application (under review)
    - Evaluates MHA, deterministic
    - Refers to MST TR

Source Term Approaches for Non-LWR Early Movers

- X-energy
  - Proposed to use developer-made source term code (XSTERM) which includes modeling of radionuclides from generation to release (and dose)
  - TR was submitted, but withdrawn to clarify and resubmit in future (not currently under review)

Source Term Approaches for Non-LWR Early Movers

- Oklo Aurora COL application (review ended)
  - Proposed maximum credible accident without release
- TerraPower
  - Development of source term methodology described in 1/13/2022 public meeting (ML22011A072)
  - Topical report planned for April 2023
- Terrestrial, Westinghouse, Others
  - Source terms to be determined
  - Public website information on non-LWR pre-application activities

Acronyms

AOO	anticipated operational occurrence
CDF	core damage frequency
COL	combined license
CP	construction permit
DBA	design basis accident
DBE	design basis event
DC	design certification
ECCS	emergency core cooling system
EP	emergency preparedness
EPZ	emergency planning zone
ESP	early site permit
FSAR	final safety analysis report
LMP	Licensing Modernization Project
MHA	maximum hypothetical accident
MST	mechanistic source term
MWt	megawatts thermal
Non-LWR	non-light water reactor
ONTs	other new technologies
PRA	probabilistic risk assessment
PWR	pressurized water reactor
RG	regulatory guide
SMR	small modular reactor
SSAR	site safety analysis report
TR	topical report

LUNCH

Accident Consequence-Related Regulation Activities

Michelle Hart
NRR/DANU/UTB2

Petition for Rulemaking

- PRM-50-121, Voluntary Adoption of Revised Design Basis Accident Dose Criteria
  - Received 11/23/2019, docketed 2/19/2020 (85 FR 31709)
  - Under evaluation – no disposition yet
- Requests voluntary rule to allow power reactor licensees to adopt alternative to the accident dose criteria specified in § 50.67, ''Accident source term.''
- Proposes a uniform value of 100 milli-Sieverts (10 rem) for offsite locations and for the control room

Emergency Preparedness for SMRs and Other New Technologies Rulemaking

- Final rule in development
  - New section 10 CFR 50.160, and related/conforming changes
  - ACRS meetings in September and November 2021
- RG 1.242 (to be issued with final rule)
  - Appendices
    - Generalized analysis methodology
    - Information on source terms

Emergency Preparedness for SMRs and Other New Technologies Rulemaking

- Appendix A, “General Methodology for Establishing Plume Exposure Pathway Emergency Planning Zone Size”
  – Provides general guidance on the consequence analysis to support plume exposure pathway EPZ size determination
  – Discusses event selection and consideration of accident likelihood

Emergency Preparedness for SMRs and Other New Technologies Rulemaking

- Appendix B, “Development of Information on Source Terms”
  - Provides guidance to develop source terms for plume exposure pathway EPZ size evaluations

Alternative Physical Security for Advanced Reactors Rulemaking

- Draft rule and guidance in development
- Voluntary alternative physical security requirements commensurate with potential safety and security consequences
- Analyses (guidance under development)
  - Develop relevant scenarios
  - Site-specific potential offsite radiological consequences

Acronyms

CFR	Code of Federal Regulations
EPZ	emergency planning zone
FR	Federal Register
PRM	petition for rulemaking
RG	Regulatory Guide
SMR	small modular reactor

Guidance and Information for Developing Source Terms for Non-LWRs

Michelle Hart, NRR/DANU/UTB2
Bill Reckley, NRR/DANU/UARP
Tim Drzewiecki, NRR/DANU/UTB1

Outline

- Accident consequence analysis for advanced reactors
- Mechanistic source term
- Recent reports on Non-LWR source term development
- Non-LWR PRA standard and source term
- Licensing Modernization Project and source term
- Overview of method in NUREG-2246, “Fuel Qualification for Advanced Reactors”
- Non-LWR accident source term information website

Accident Consequence Analysis for Advanced Reactors

- Regulatory nexus
  - Siting and safety analysis regulatory requirement
  - Newer uses for advanced reactors
    - LMP
    - Plume exposure pathway EPZ size determination
    - Alternative security requirements – ongoing rulemaking
    - Part 53 – ongoing rulemaking

Accident Consequence Analysis for Advanced Reactors

- Accident source term development considerations
  - Event selection, scenarios
  - Balance of prevention vs. mitigation
  - Relationship to functional containment
    - A barrier, or set of barriers taken together, that effectively limit the physical transport of radioactive material to the environment (SECY-18-0096)
  - Relationship to PRA
  - Uncertainty

Accident Consequence Analysis for Advanced Reactors

- Mechanistic or deterministic evaluation
  - LMP assumes MST and use of PRA
  - Some non-LWRs may choose to provide a postulated MHA, similar to non-power reactor licensees
- No current specific RG on MST or non-LWR source terms, however
  - RG 1.183, regulatory position C.2, "Attributes of an Acceptable AST," may be useful
  - SECY-93-092 included staff recommendations on non-LWR source terms

Mechanistic Source Term

- SECY-93-092 definition of MST

  A mechanistic source term is the result of an analysis of fission product release based on the amount of cladding damage, fuel damage, and core damage resulting from the specific accident sequences being evaluated. It is developed using best-estimate phenomenological models of the transport of the fission products from the fuel through the reactor coolant system, through all holdup volumes and barriers, taking into account mitigation features, and finally, into the environs.

SECY-93-092: Provisions for Staff Assurance

- *The performance of the reactor and fuel under normal and off-normal conditions is sufficiently well understood to permit a mechanistic analysis. Sufficient data should exist on the reactor and fuel performance through the research, development, and testing programs to provide adequate confidence in the mechanistic approach.*
- *The transport of fission products can be adequately modeled for all barriers and pathways to the environs, including specific consideration of containment design. The calculations should be as realistic as possible so that the values and limitations of any mechanism or barrier are not obscured.*
- *The events considered in the analyses to develop the set of source terms for each design are selected to bound severe accidents and design-dependent uncertainties*

National Lab Non-LWR Source Term Reports

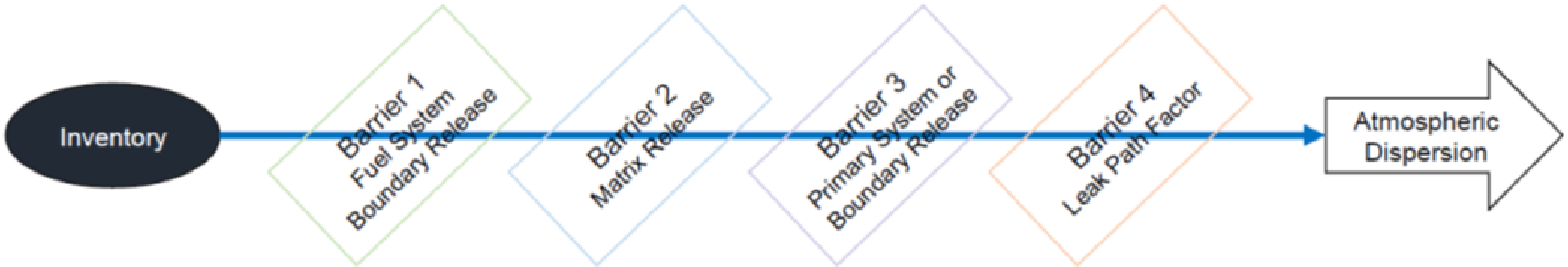
- Technology inclusive, what to do to develop accident source terms, not specific on how to do it
- No specific methods or phenomenological models
- Do not provide technology-related source terms or releases

Technology-Inclusive Determination of Mechanistic Source Terms for Offsite Dose-Related Assessments for Advanced Nuclear Reactor Facilities

INL/EXT-20-58717, Revision 0, June 2020, ML20192A250

- Summarizes a risk-informed, performance-based, and technology-inclusive approach to determine source terms
- Graded process
  - Conservative non-mechanistic approach
  - MST calculation methods
    - Design-specific scenarios for a range of licensing basis events
    - Best-estimate models with uncertainty quantification

MST Formulation



$$I(RN_j) * F(S_i, RN_j, t) * MR(S_i, RN_j, t) * PSR(S_i, RN_j, t) * LPF(S_i, RN_j, t) = ST(S_i, RN_j, t)$$

Figure 1-2 INL/EXT-20-58717, Revision 1. From Illustration of radionuclides retention and removal process for one non-LWR concept (reproduced from SAND2020-0402)

Technology-Inclusive Source Term Methodology Determination

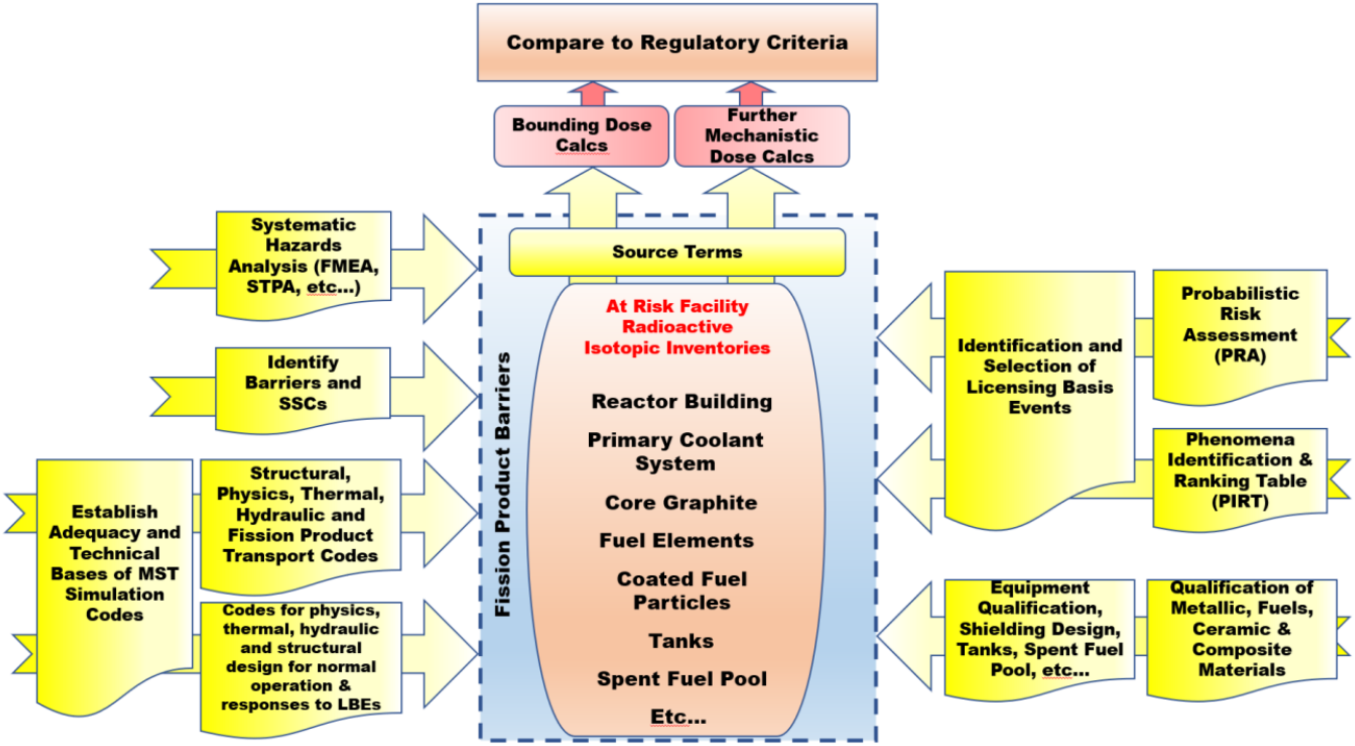


Figure 3-1 Technology-inclusive source terms determination methodology components (modified from Ref. [18]).

INL Report Methodology Steps

1: Identify Regulatory Requirements

2: Identify Reference Facility Design

3: Define Initial Radionuclide Inventories

4. Perform Bounding Calculations

5. Conduct SHA and Perform Simplified Calculations

6. Consider Risk-informed System Design Changes

7. Select Initial List of LBEs and Conduct PIRT

8. Establish Adequacy of MST Simulation Tools

9. Develop and Update PRA Model

10. Identify or Revise the List of LBEs

11. Select LBEs to Include Design Basis External Hazard Level for Source Term Analysis

12. Perform Source Term Modeling and Simulation for LBEs

13. Review LBEs List for Adequacy of Regulatory Acceptance

14. Document Completion of Source Term Development

Simplified Approach for Scoping Assessment of Non-LWR Source Terms

SAND2020-0402, January 2020, [ML20052D133]

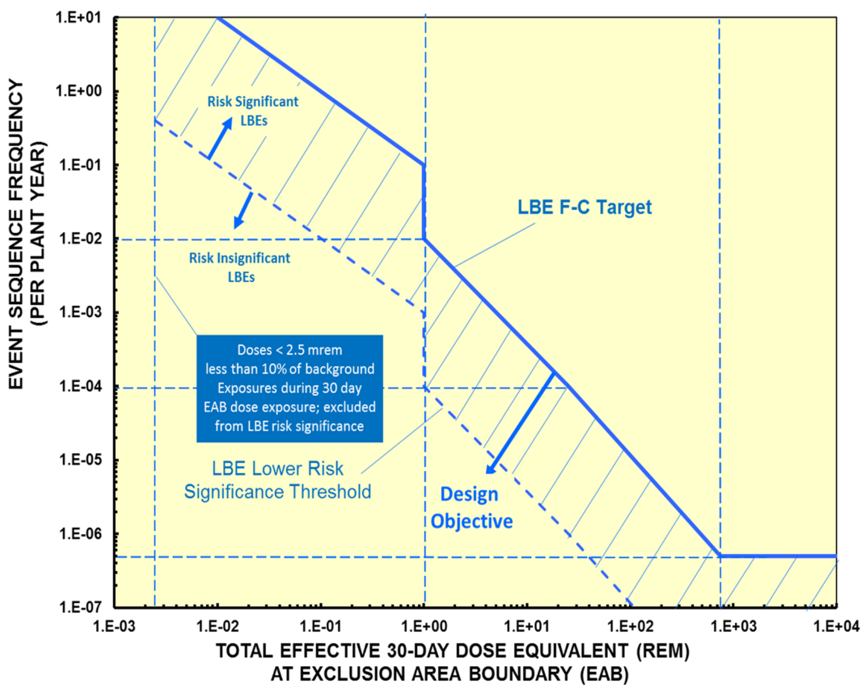
- Primarily qualitative means to identify the dominant considerations that affect a release mitigation strategy
- Classifies release mitigation strategies based on a range of barriers, physical attenuation processes, and system performance under sample accident scenarios
- Did NOT develop quantitative estimates of radiological release magnitudes and compositions to the environment
- Looked at high temperature gas reactors, sodium fast reactors, and liquid fueled molten salt reactors

Non-LWR PRA Standard ASME/ANS RA-S-1.4-2021

- Full scope PRA (includes consequence analysis)
- Mechanistic Source Term Analysis (MS) element provides useful information on what to do to develop mechanistic source terms

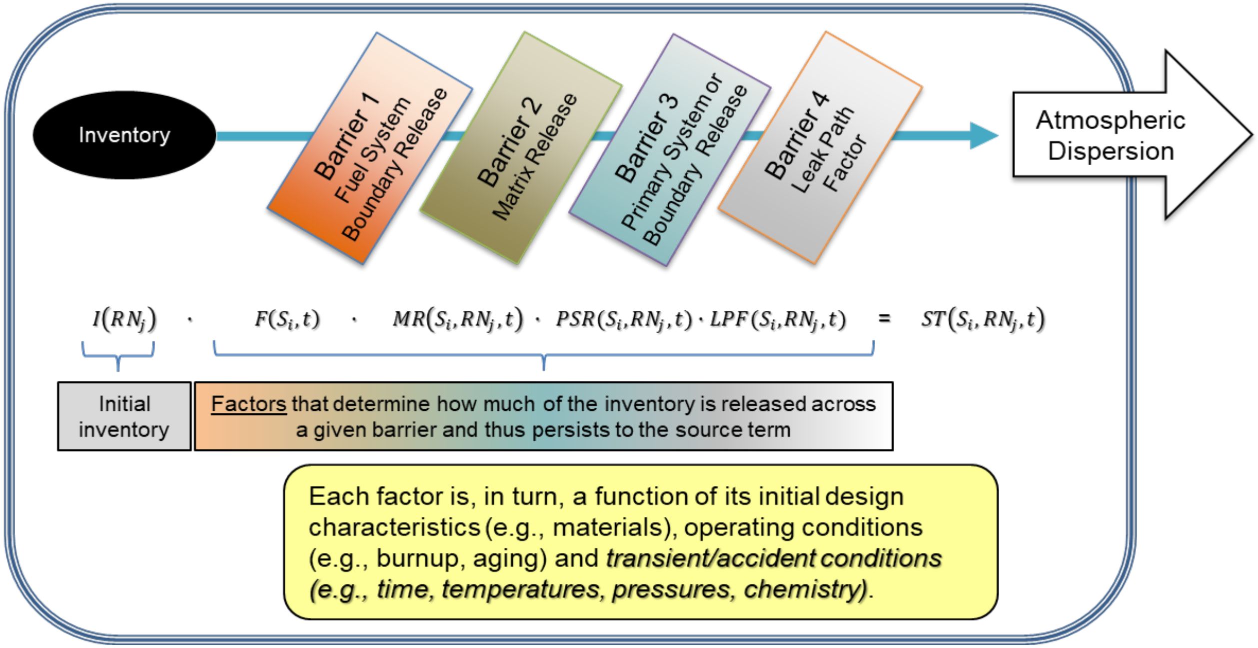
Licensing Modernization

- Risk-informed approach to selection and analysis of licensing basis events
- Combined with assessment of cumulative risks
- Key roles for PRA and MST



Licensing Modernization

Recent NRC activities related to advanced reactors (e.g., functional containment performance criteria, possible changes to emergency planning & security, and DG-1353) recognize the limitations of existing LWR-related guidance, which requires a return to first principles such as fundamental safety functions supporting the retention of radionuclides



See: SECY-18-0096, “Functional Containment Performance Criteria for Non-Light-Water-Reactors,” and INL/EXT-20-58717, “Technology-Inclusive Determination of Mechanistic Source Terms for Offsite Dose-Related Assessments for Advanced Nuclear Reactor Facilities”

Licensing Modernization

- Flexibility provided on how to develop safety case
- NRC Advanced Reactor Policy Statement encourages use of passive and inherent features

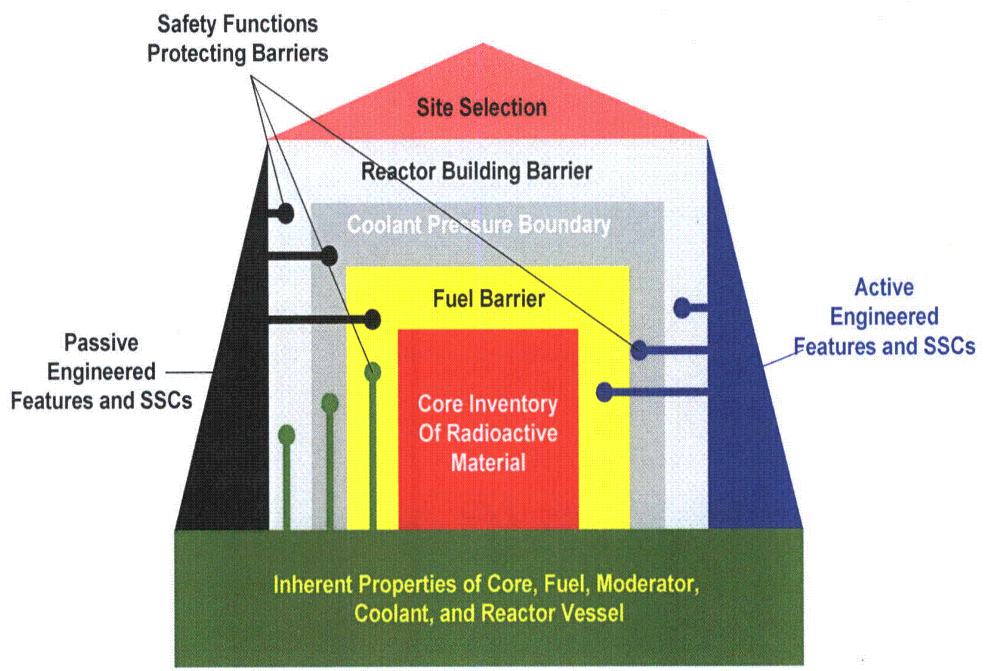
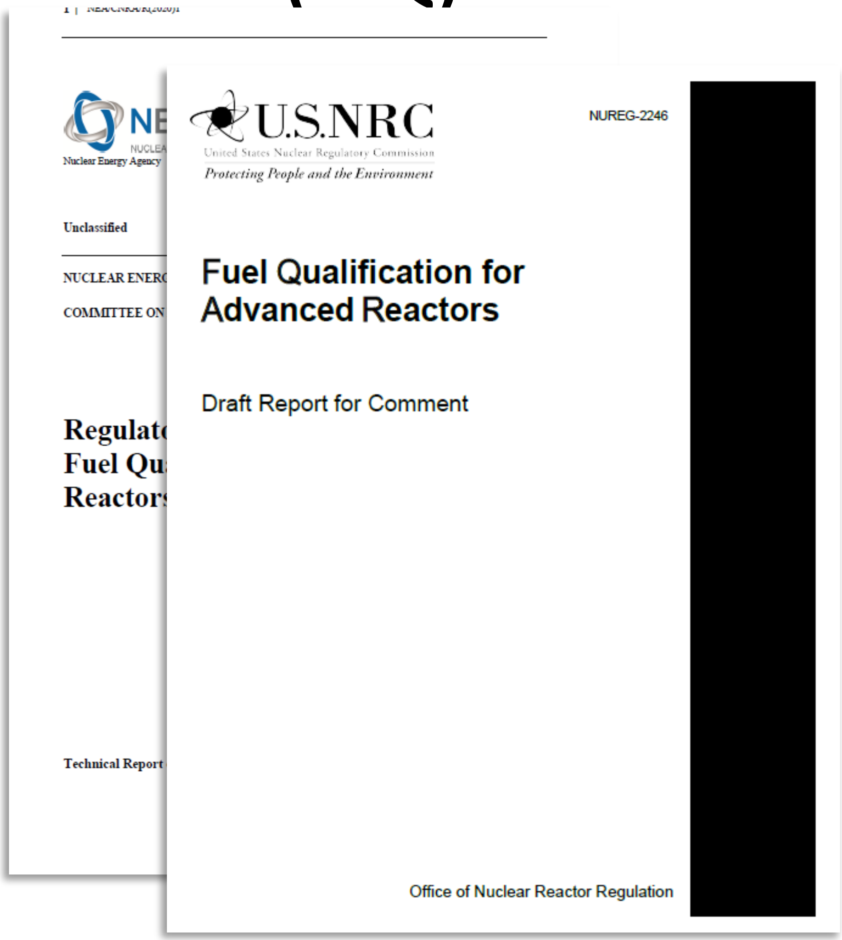


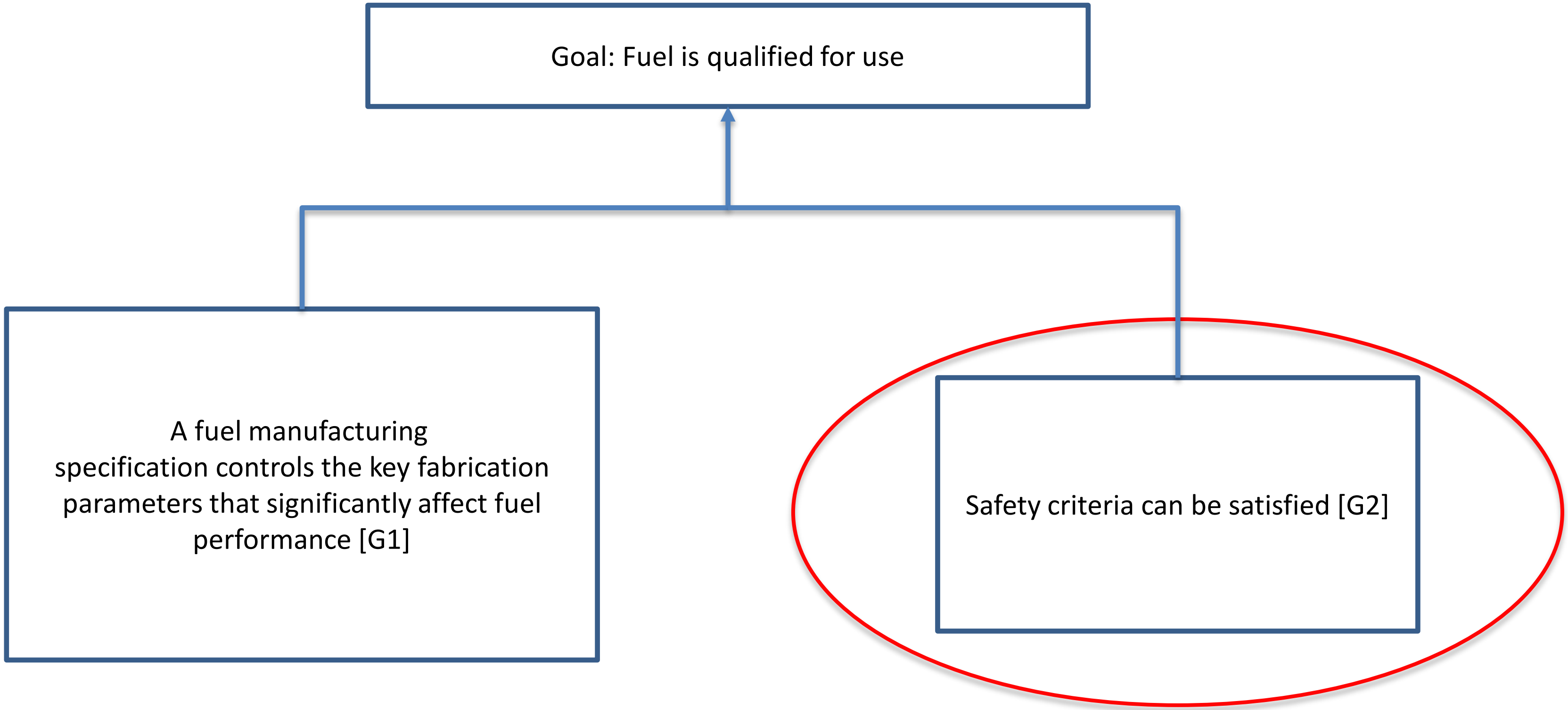
Figure 3-6. Elements of safety design approach incorporated into ***Plant Capability Defense-in-Depth***.

Assessment Frameworks
Fuel Qualification (FQ)

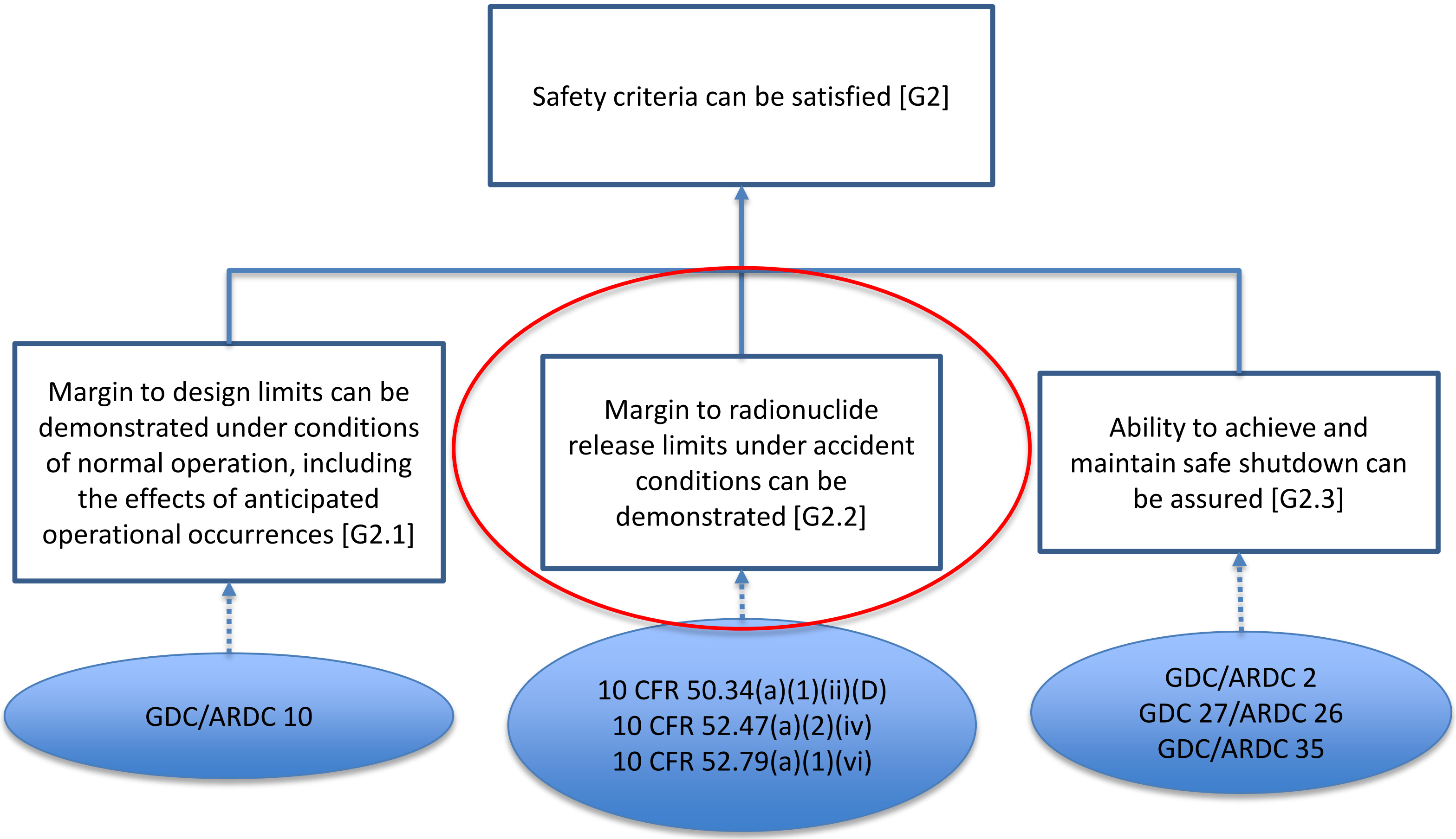
- Top-down approach to identify criteria (goals) to support a finding that "fuel is qualified"



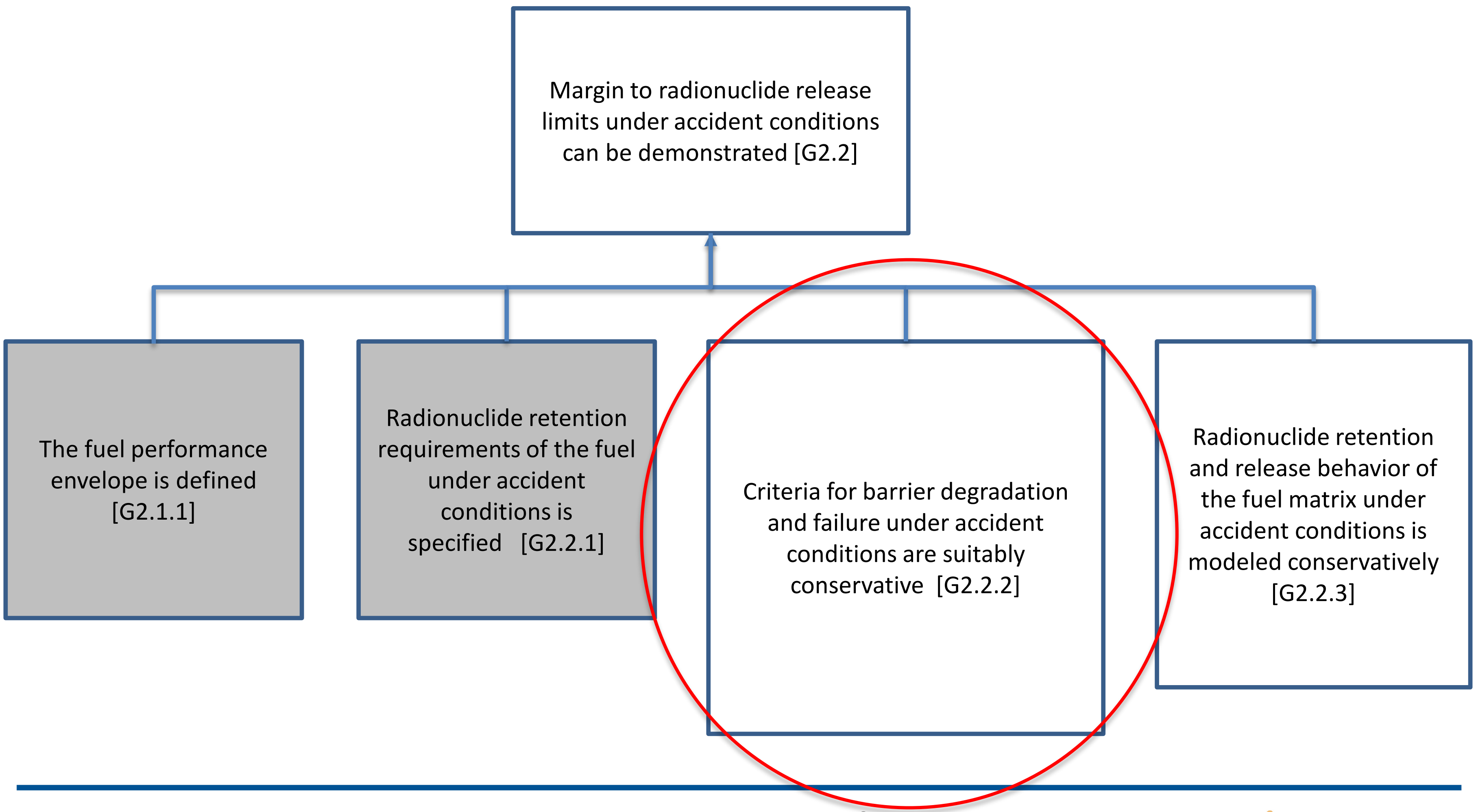
FQ Assessment Framework



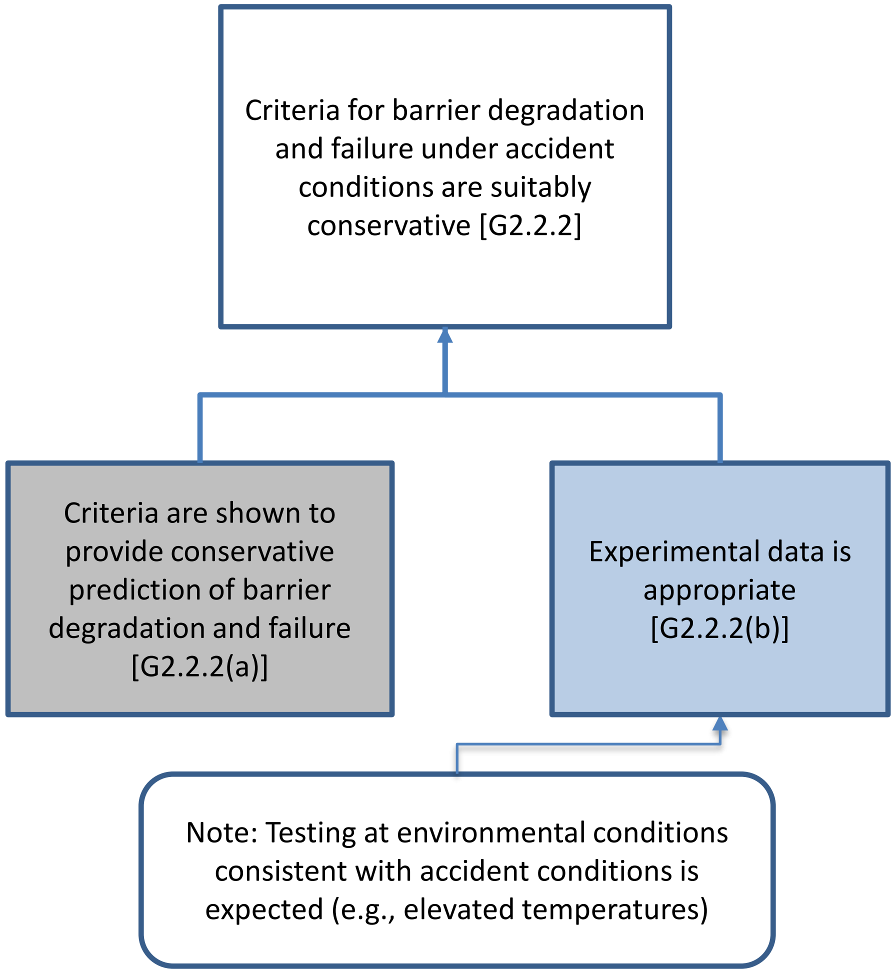
G2: Safety Criteria



G2.2: Radionuclide Release Limits



G2.2.2 Criteria for Barrier Degradation



Complete FQ Assessment Framework

GOAL				Fuel is qualified for use
G1				Fuel is manufactured in accordance with a specification
	G1.1			Key dimensions and tolerances of fuel components are specified
	G1.2			Key constituents are specified with allowance for impurities
	G1.3			End state attributes for materials within fuel components are specified or otherwise justified
G2				Margin to safety limits can be demonstrated
	G2.1			Margin to design limits can be demonstrated under conditions of normal operation and AOOs
		G2.1.1		Fuel performance envelope is defined
		G2.1.2		Evaluation model is available (see EM Assessment Framework)
	G2.2			Margin to radionuclide release limits under accident conditions can be demonstrated
		G2.1.1		Fuel performance envelope is defined
		G2.2.1		Radionuclide retention requirements are specified
		G2.2.2		Criteria for barrier degradation and failure are suitably conservative
			(a)	Criteria are conservative
			(b)	Experimental data are appropriate (see ED Assessment Framework)
		G2.2.3		Radionuclide retention and release from fuel matrix are modeled conservatively
			(a)	Model is conservative
			(b)	Experimental data are appropriate (see ED Assessment Framework)
	G2.3			Ability to achieve and maintain safe shutdown is assured
		G2.3.1		Coolable geometry is ensured
			(a)	Criteria to ensure coolable geometry are specified
			(b)	Evaluation models are available (see EM Assessment Framework)
		G2.3.2		Negative reactivity insertion can be demonstrated
			(a)	Criteria are provided to ensure that negative reactivity insertion path is not obstructed
			(b)	Evaluation model is available (see EM Assessment Framework)

GOAL			Evaluation model is acceptable for use
EM G1			Evaluation model contains the appropriate modeling capabilities
	EM G1.1		Evaluation model is capable of modeling the geometry of the fuel system
	EM G1.2		Evaluation model is capable of modeling the material properties of the fuel system
	EM G1.3		Evaluation model is capable of modeling the physics relevant to fuel performance
EM G2			Evaluation model has been adequately assessed against experimental data
	EM G2.1		Data used for assessment are appropriate (see ED Assessment Framework)
	EM G2.2		Evaluation model is demonstrably able to predict fuel failure and degradation mechanisms over the test envelope
		EM G2.2.1	Evaluation model error is quantified through assessment against experimental data
		EM G2.2.2	Evaluation model error is determined throughout the fuel performance envelope
		EM G2.2.3	Sparse data regions are justified
		EM G2.2.4	Evaluation model is restricted to use within its test envelope

GOAL		Experimental data used for assessment are appropriate
ED G1		Assessment data are independent of data used to develop/train the evaluation model
ED G2		Data has been collected over a test envelope that covers the fuel performance envelope
ED G3		Experimental data have been accurately measured
	ED G3.1	The test facility has an appropriate quality assurance program
	ED G3.2	Experimental data are collected using established measurement techniques
	ED G3.3	Experimental data account for sources of experimental uncertainty
ED G4		Test specimens are representative of the fuel design
	ED G4.1	Test specimens are fabricated consistent with the fuel manufacturing specification
	ED G4.2	Distortions are justified and accounted for in the experimental data

* For illustrative purposes only. Please see Appendix A to NUREG-2246 for a legible list.

Non-LWR Accident Source Term Webpage Information

https://www.nrc.gov/reactors/new-reactors/advanced/related-documents/nuclear-power-reactor-source-term.html

- One-stop shop for existing information, on public website under advanced reactors
  - Discussion of accident source terms
  - Linked list of documents relevant to development of non-LWR accident source terms for licensing
- Staff will keep up to date

Acronyms

AST	alternative source term
EPZ	emergency planning zone
INL	Idaho National Laboratory
LBE	licensing basis event
LMP	Licensing Modernization Project
LWR	light water reactor
MHA	maximum hypothetical accident
MST	mechanistic source term
Non-LWR	non-light water reactor
PIRT	phenomena identification and ranking table
PRA	probabilistic risk assessment
RG	regulatory guide
SHA	system hazard analysis

Guidance for developing advanced reactor source term (long-term)

Bill Reckley
Michelle Hart
John Segala
NRR/DANU

General Approach

- Maintain traditional LWR approach (RG 1.183) as an acceptable option
- Technology-inclusive methodology available as an option
- Actual implementation is technology/design specific
- NRC not planning to provide analytical inputs to applicants (beyond making available NRC developed models)

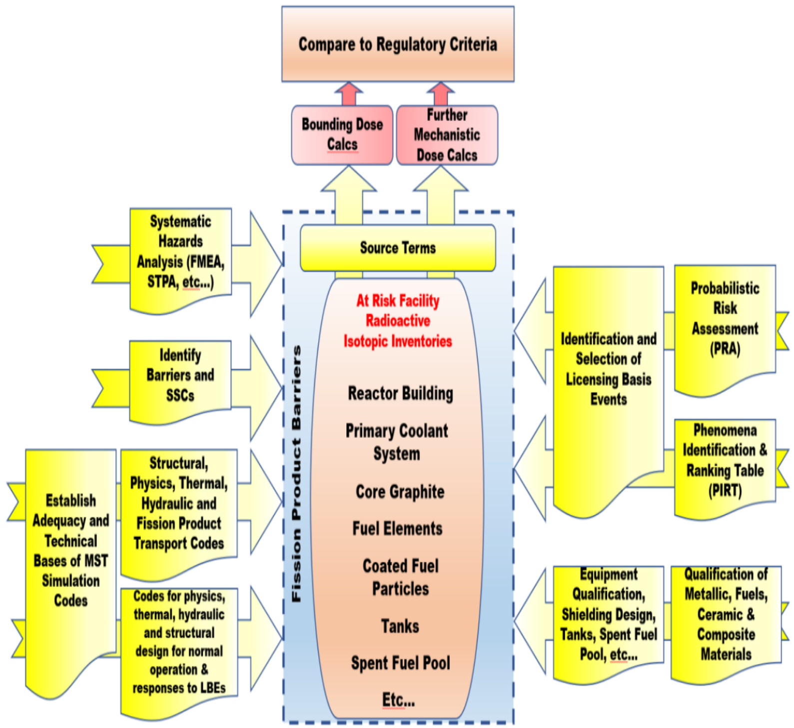
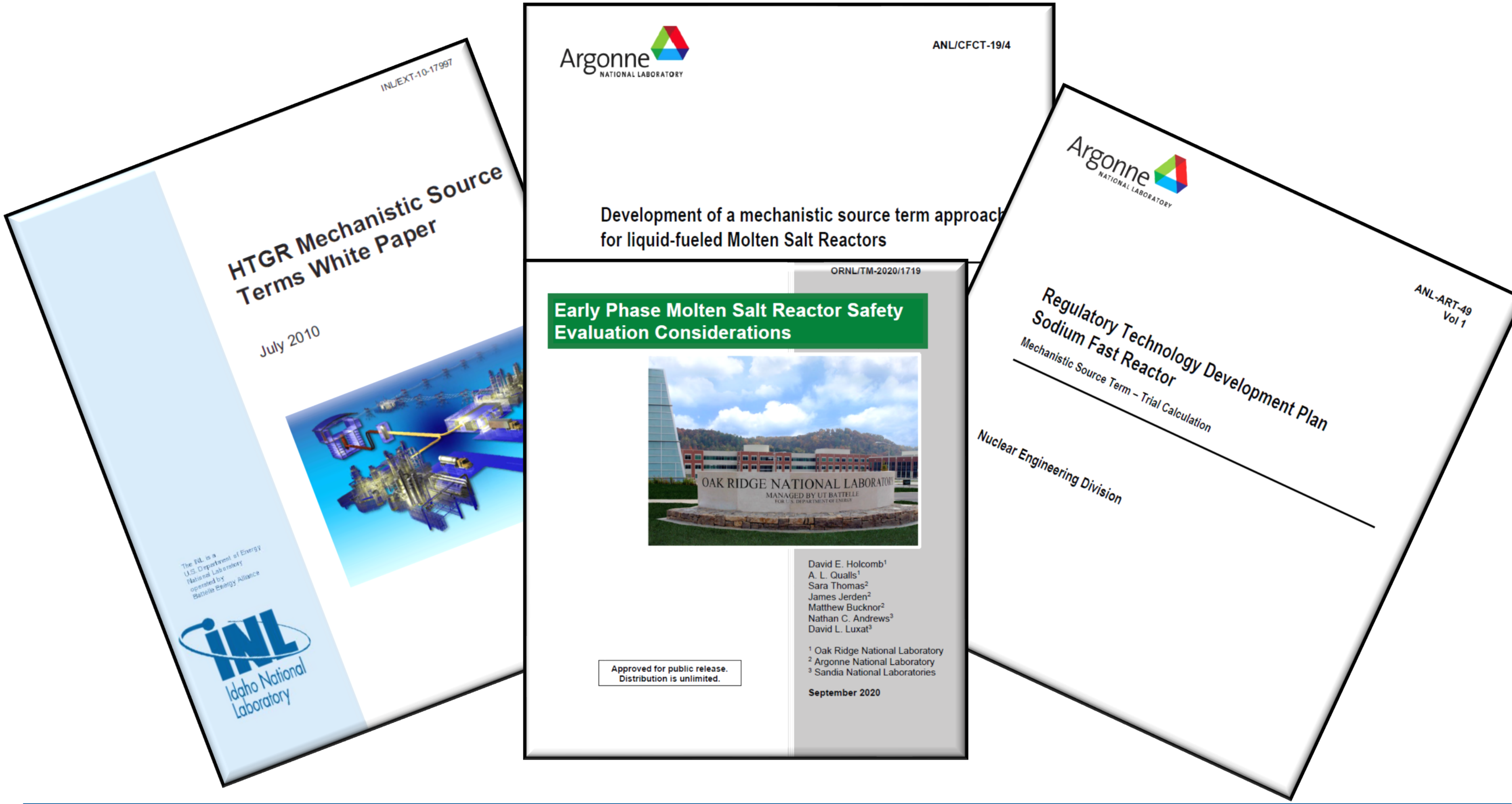
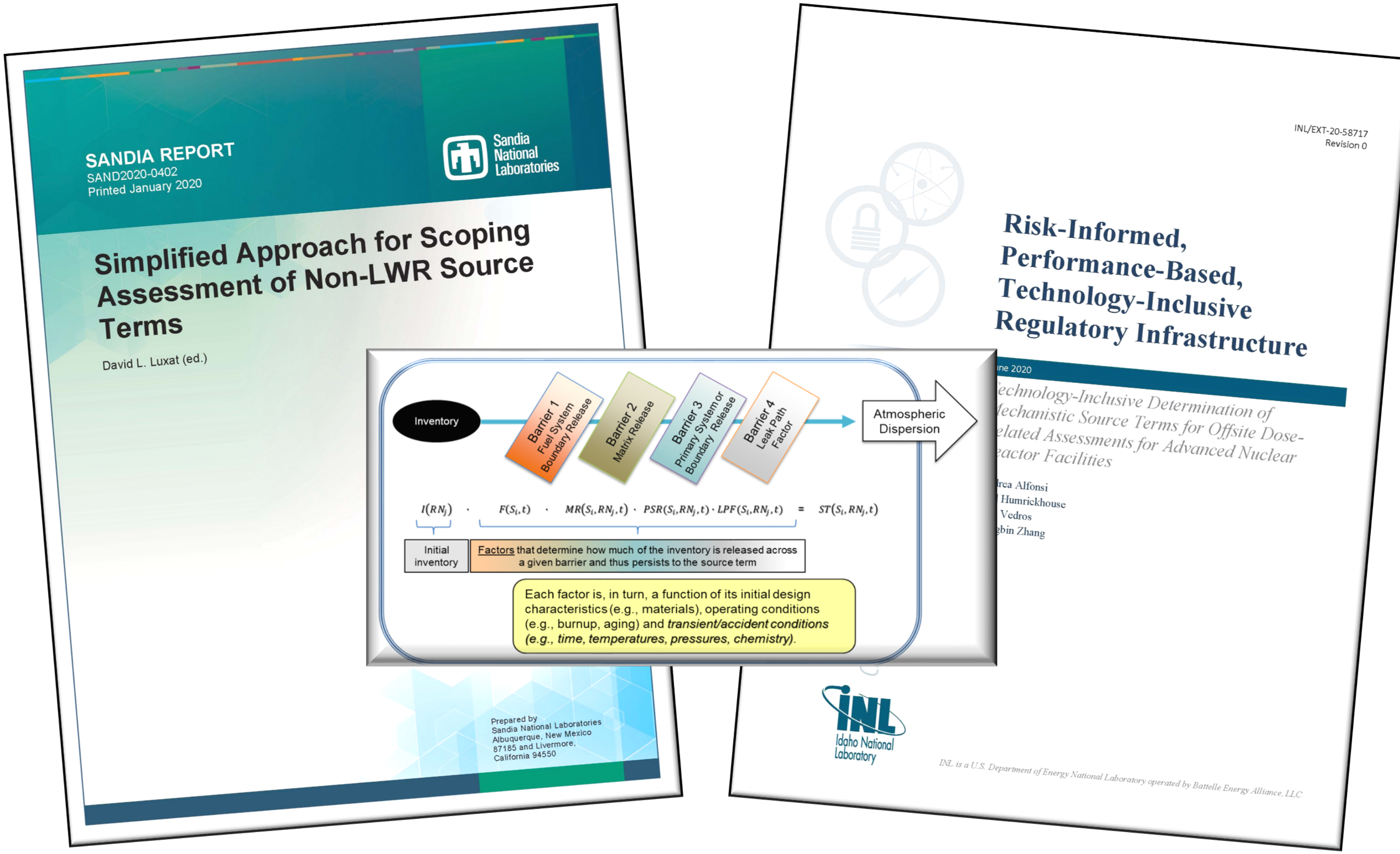


Figure 3-1 Technology-inclusive source terms determination methodology components (modified from Ref. [18]).

DOE/National Laboratories



NRC Activities



Next Generation Nuclear Plant (NGNP)

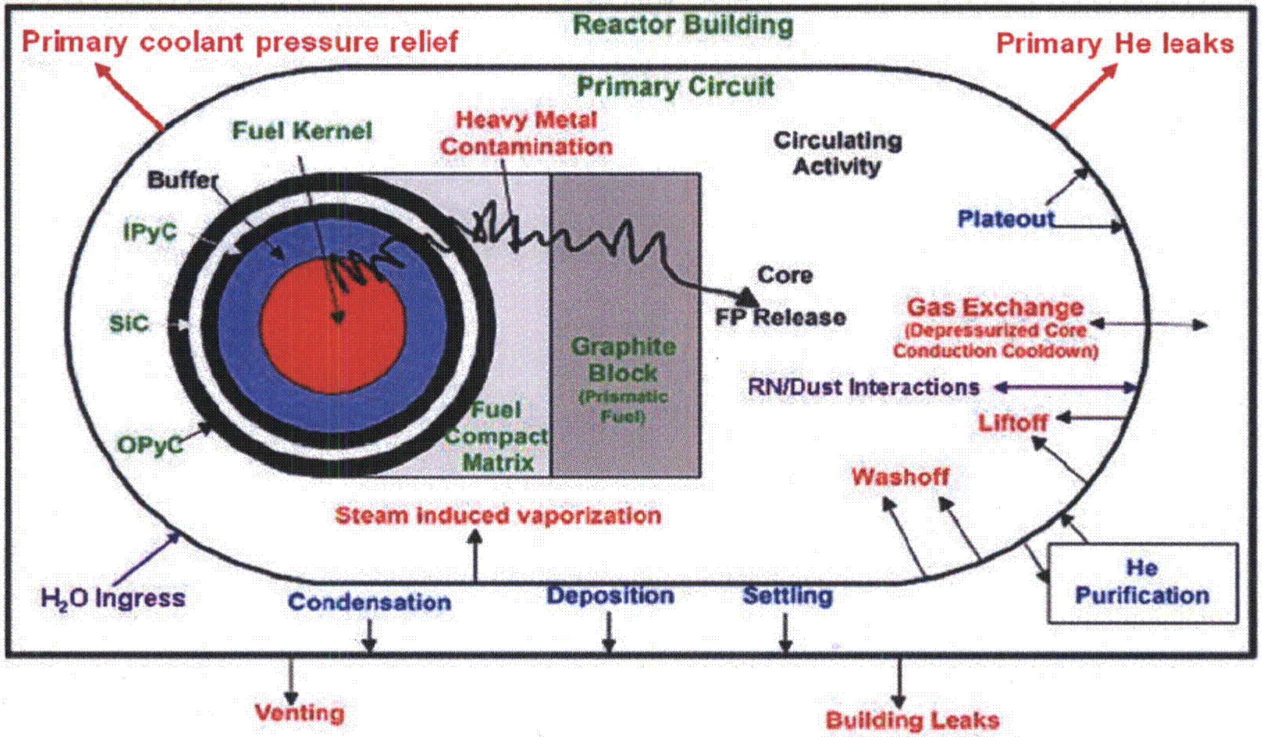
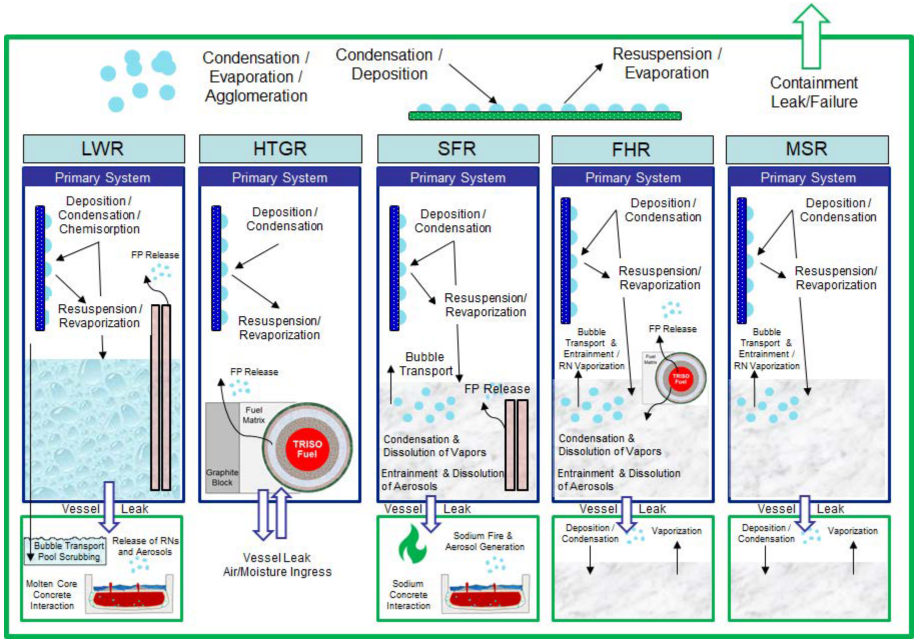


Figure 2-4. HTGR radionuclide retention system.

Model Development



Primer & User Guide
Reference Manual
Assessment Problems

Applications & Pre-App Interactions

NuScale Power

Kairos Power

Westinghouse

X energy

TerraPower

Moving Forward

- Following the scientific work being done by national laboratories and developers
- Engaging with developers
- Continuing to develop NRC models and identify related uncertainties
- Consider additional guidance based on experience with ongoing interactions
- Consider feedback on the new webpage

Opportunity for Public Comment

Member Discussion

Adjourn