Advanced Reactor Stakeholder Public Meeting

August 20, 2020

Telephone Bridgeline: (888) 810-4937
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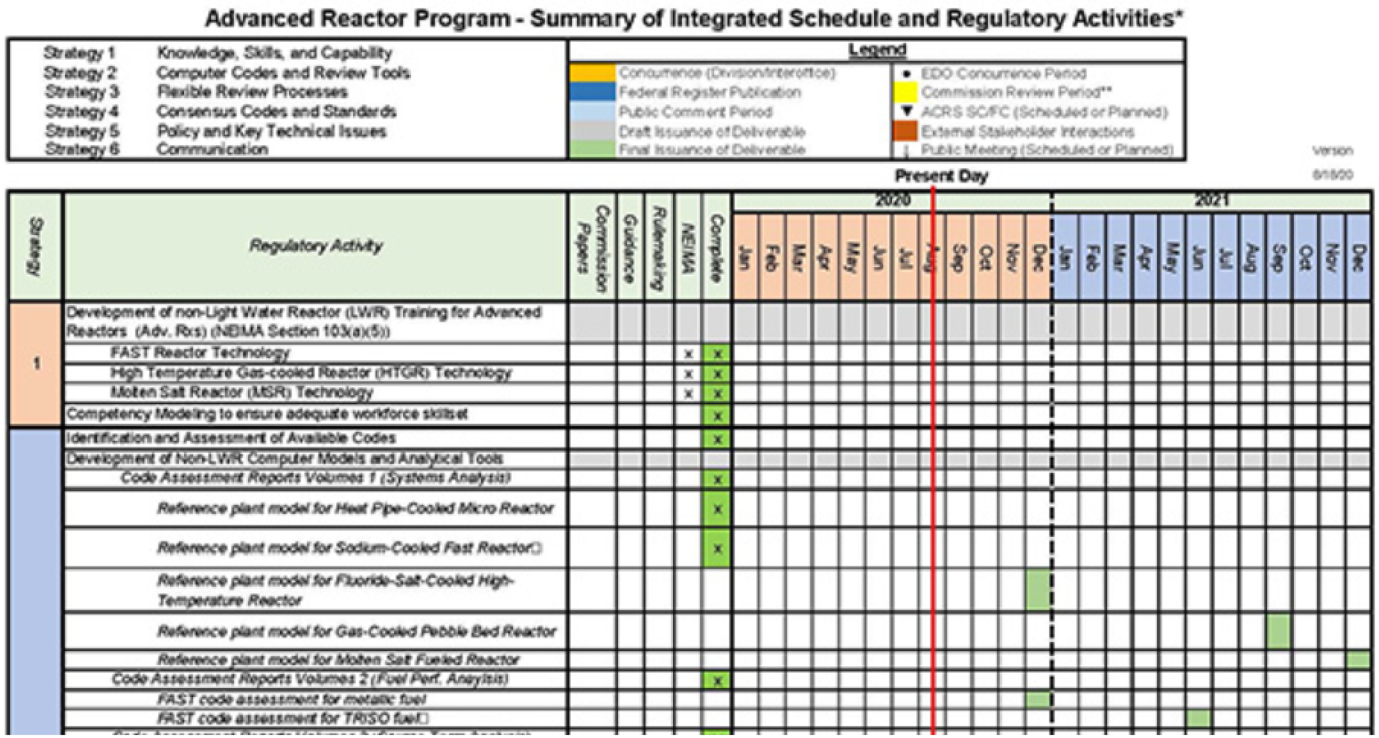
Time	Agenda	Speaker
10:00 - 10:10 am	Opening Remarks	NRC
10:10 - 11:10 am	Presentation on ANL Report, “The Assessment of Tritium Detection and Control in Molten Salt Reactors”	NRC/ANL
11:10 - 12:10 pm	Presentation on INL Report, “Technology-Inclusive Determination of Mechanistic Source Terms for Offsite Dose-Related Assessments for Advanced Nuclear Reactor Facilities”	NRC/INL
12:10 – 12:40 pm	NRC discussion of Advanced-Reactor Source Term – Pilot Studies	NRC
12:40 – 1:00 pm	BREAK	All
1:00 - 1:30 pm	Discussion of Considerations for Annual Fee Regulations for Microreactors	NEI
1:30 - 2:15 pm	Discussion of Part 53 Rulemaking Plan and White Paper	NRC
2:15 - 2:30 pm	Industry Stakeholder’s Perspectives on Part 53	NEI/USNIC
2:30 - 2:45 pm	Discussion of Status of Spent Fuel Reprocessing Rulemaking	NRC
2:45 - 3:00 pm	Overview of ORNL Report on Preparing and Reviewing a Molten Salt Non-Power Reactor Application	NRC
3:00 – 3:15 pm	Concluding Remarks and Future Meeting Planning	NRC/All

Advanced Reactor Integrated Schedule of Activities

Advanced Reactor - Summary of Integrated Schedule and Regulatory Activities

Summary of Integrated Schedule and Regulatory Activities (updated 08/18/2020)



https://www.nrc.gov/reactors/new-reactors/advanced.html

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

ASSESSMENT OF TRITIUM DETECTION AND CONTROL IN MOLTEN SALT REACTORS

David Grabaskas, Tingzhou Fei, James Jerden
Argonne National Laboratory

OBJECTIVES

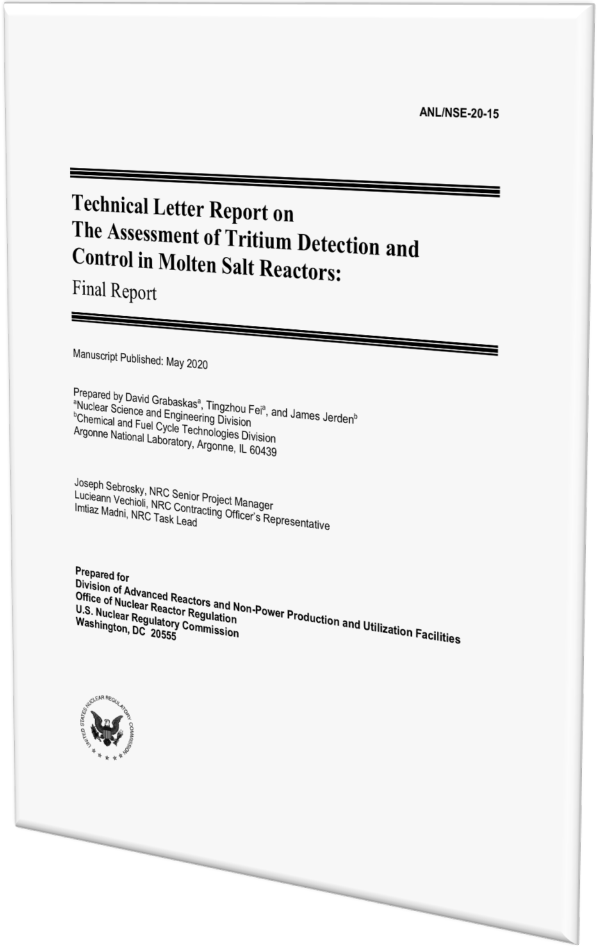
- **Assist NRC:**
  - Expanding capacity and capabilities for licensing non-LWRs through knowledge base and skillset development

- **Technical Assessment of Tritium Behavior in MSRs:**
  - Location and pathways of tritium generation
  - Tritium transport and retention phenomena
  - Barriers to tritium release and mechanisms for tritium control
  - Applicable experience and existing data on tritium behavior and control
  - Available modeling and simulation tools

- **Regulatory Considerations:**
  - Applicability of current regulations
  - Associated limits and constraints on tritium handling and release
  - Areas of consideration during NRC review of MSR licensing applications
  - Assessment of the adequacy of the current regulation and guidance

REPORT

- **ANL/NSE-20-15:**
  - Available on ADAMS and NRC Advanced Reactor Webpage


ANL/NSE-20-15

Technical Letter Report on
The Assessment of Tritium Detection and Control in Molten Salt Reactors:
Final Report

Manuscript Published: May 2020

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Prepared for
Division of Advanced Reactors and Non-Power Production and Utilization Facilities
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

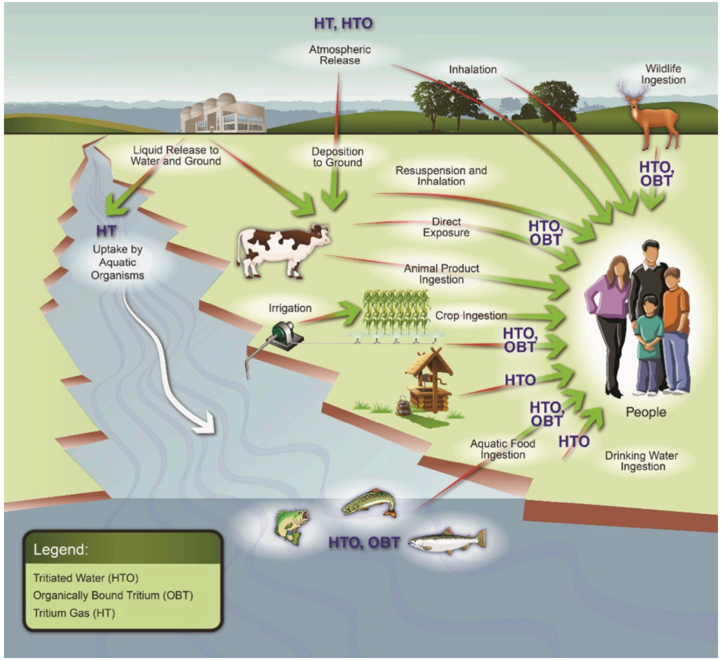
BACKGROUND ON TRITIUM

■ ^{3}H or T

- Radioactive isotope of hydrogen with 12.3 year half-life
- Naturally occurring due to cosmic ray interaction with the atmosphere
- Additional environmental tritium from nuclear weapons tests and nuclear reactor effluents

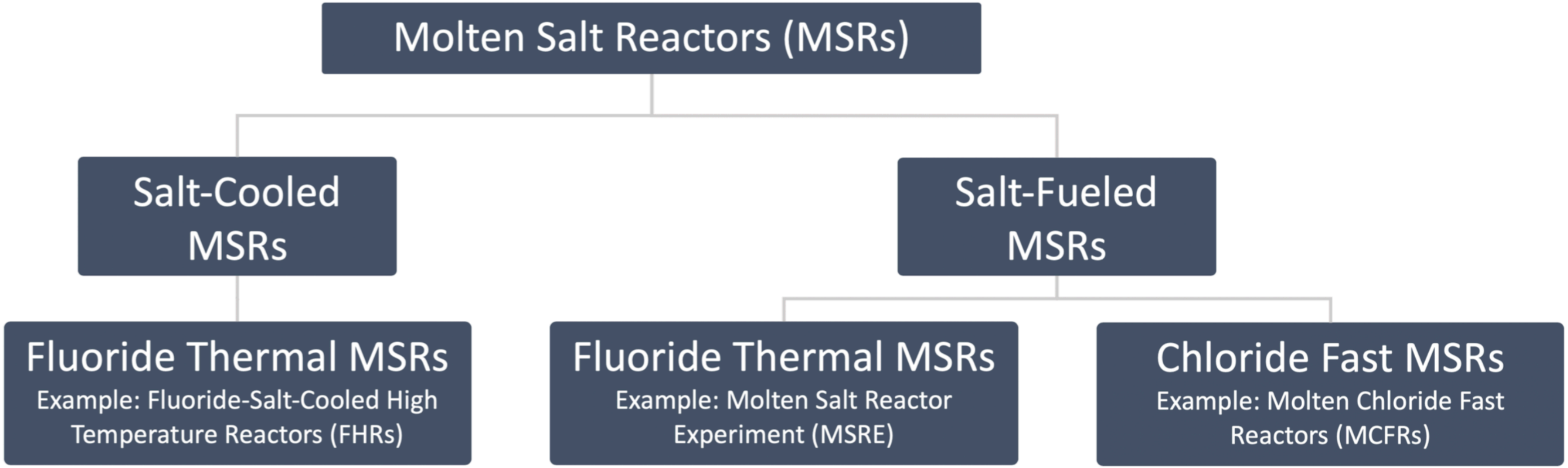
■ Health Hazard

- Low energy beta emitter (max ~18keV):
  - Internal exposure is the only concern as beta has insufficient energy to penetrate dead skin layer
- Differing chemical forms and biological impact:
  - HT/T_2 Gas: Exhaled quickly from the body
  - HTO Water: Mostly eliminated with biological half-life of water (10 days)
  - OBT: Organically bound tritium, can act like carbon in body with longer biological half-life (40 days)



Courtesy of the Canadian Nuclear Safety Commission (CNSC)

MOLTEN SALT REACTORS: NOMENCLATURE



MSR: SALT SELECTION

- **Considerations:**
  - Neutronics, material compatibility, dissolution properties, stability, and thermophysical properties
  - Lithium- and beryllium-bearing salts are popular choices for thermal reactors due to moderating ability
  - Most past experience with FLiBe

Salt (mole fraction)	T_{melt} (°C)	T_{boil} (°C)	ρ[a] (g/cm^3)	$\rho * C_p$[a] (cal/cm^3-°C)
LiF-BeF_2 (FLiBe) (67-33)	459	1430	1.94	1.12
LiF-NaF-KF (FLiNaK) (46.5-11.5-42)	454	1570	2.02	0.91
NaF-ZrF_4 (59.5-40.5)	500	1350	3.03	0.84
LiF-ZrF_4 (51-49)	509	[b]	3.09	[b]
$NaCl$-$MgCl_2$ (68-42)	445	1465	1.94	0.50

[a] At 700°C temperature.
[b] Data gap.

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MSR Tritium Phenomena

- Production
- Transport
- Control

MSRE Experience

- Design
- Operation
- Tritium

Modeling and Simulation

- Requirements
- Current Capabilities
- Assessment

Regulatory Considerations

- Quantitative Limits
- Regulation
- Assessment

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TRITIUM PRODUCTION: WATER REACTORS

■ Ternary Fission

- All fission nuclear reactors create tritium from ternary fission (fission with three products)
- Approximately 1 in 10,000 fissions
- Largely contained within the fuel in water reactors

■ Other Factors:

- Boron neutron capture in control elements (BWRs) or coolant (PWRs)

$$^{10}B + n \rightarrow\ 2\ ^{4}He + {}^{3}H$$

- Deuterium neutron capture in heavy water reactors (HWRs), such as CANDUs

$$^{2}H + n \rightarrow {}^{3}H$$

TRITIUM PRODUCTION: MSR

MSRs

- Two major factors in the production of tritium: Lithium and Beryllium

^7Li

$$^7Li + n \rightarrow \left\{ \begin{array}{c} ^4He + {}^3H + n \\ ^5He + {}^3H \end{array} \right\}$$

^9Be

$$^9Be + n \rightarrow {}^3H + {}^7Li$$

^6Li

$$^6Li + n \rightarrow {}^4He + {}^3H$$

$$^9Be + n \rightarrow {}^4He + {}^6He$$

$$^6He\ (t_{1/2} = 0.8\ \mathrm{s}) \rightarrow {}^6Li + \beta^- + \bar{\nu}_e$$

TRITIUM PRODUCTION: MSR

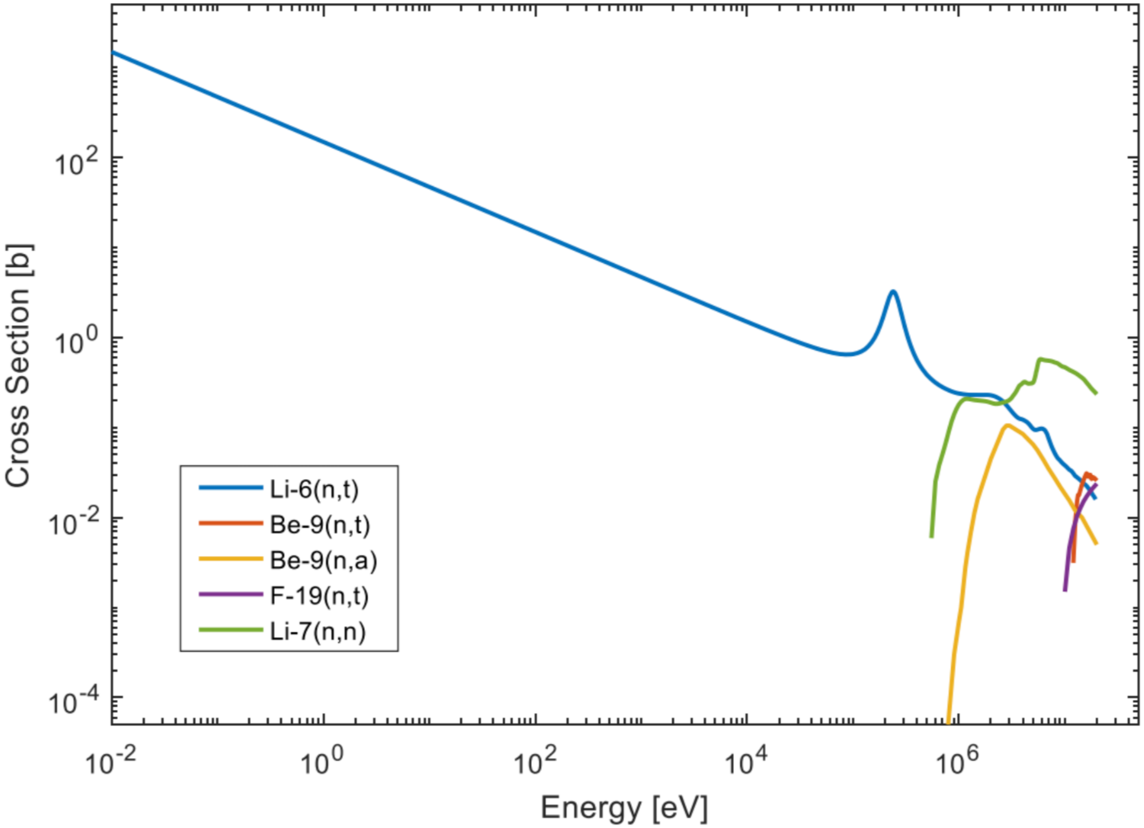
Lithium

- Natural lithium is 92.4% ^{7}Li and 7.6% ^{6}Li
- ^{7}Li has a much smaller tritium-producing cross-section than ^{6}Li (as will be shown)
- Lithium enrichment utilized to reduce ^{6}Li due to tritium concerns
- 99.995% ^{7}Li enrichment is typical, further enrichment may be cost prohibitive

Establishing Equilibrium

- If a molten salt contains both Li and Be, the existing ^{6}Li contained in the salt will be consumed by neutron interactions, but new ^{6}Li is created from neutron interactions with beryllium
- If salt only contains Be, ^{6}Li concentration will build over time until an equilibrium is reached

TRITIUM PRODUCTION: MSR



Source: K. Dolan, "Tritium Thermal Desorption Testing of Nuclear Graphite Irradiated at Fluoride-Salt-Cooled High-Temperature Reactor Conditions," Thesis, Massachusetts Institute of Technology, 2018

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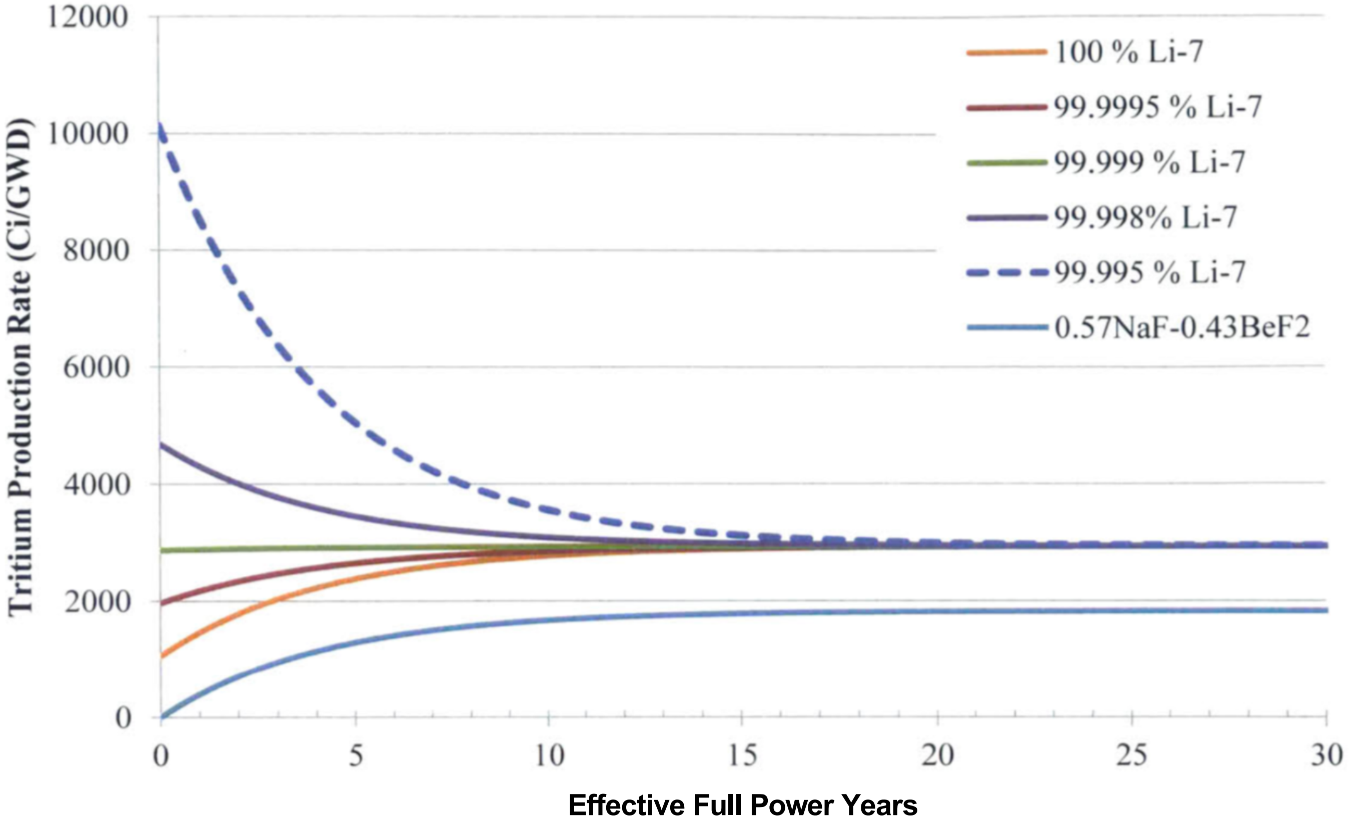
TRITIUM PRODUCTION: PB-FHR EXAMPLE



Source: J. Stempien, "Tritium Transport, Corrosion, and Fuel Performance Modeling in the Fluoride Salt-Cooled High Temperature Reactor (FHR)," Dissertation, Massachusetts Institute of Technology, 2015

TRITIUM PRODUCTION: RATE COMPARISON

- **Example MSR Concepts**:
  - Molten Salt Breeder Reactor (MSBR): A 1000MWe, FLiBe salt-fueled MSR concept studied extensively by ORNL in the 1970s following the operation of MSRE
  - Pebble-Bed Fluoride Salt-Cooled High-Temperature Reactor (PB-FHR): A solid fuel, FLiBe salt-cooled FHR design developed by the University of California-Berkeley, which serves as the basis for much recent FHR research

Reactor Type	**Normalized Tritium Production Rate (Ci/GWe/yr)[a]**			
	Fuel	**Coolant**	**Moderator**	**Control Elements**
PWR	11,000 – 25,000	300 – 1,000		1,000
BWR	11,000 – 25,000	b		3,000 – 5,000
HWR	14,000 – 20,000	50,000	600,000 – 2,400,000	1,000
MSBR	730,000		b	b
PB-FHR	b	2,100,000/720,000[c]	b	b

[a] Unit is curies of tritium produced per GWe during an approximate operating year
[b] Negligible or unknown.
[c] Beginning of life/Steady-state

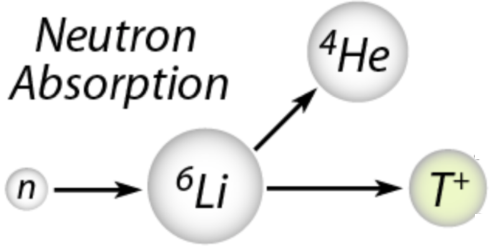
TRITIUM PRODUCTION: SUMMARY

Key Point

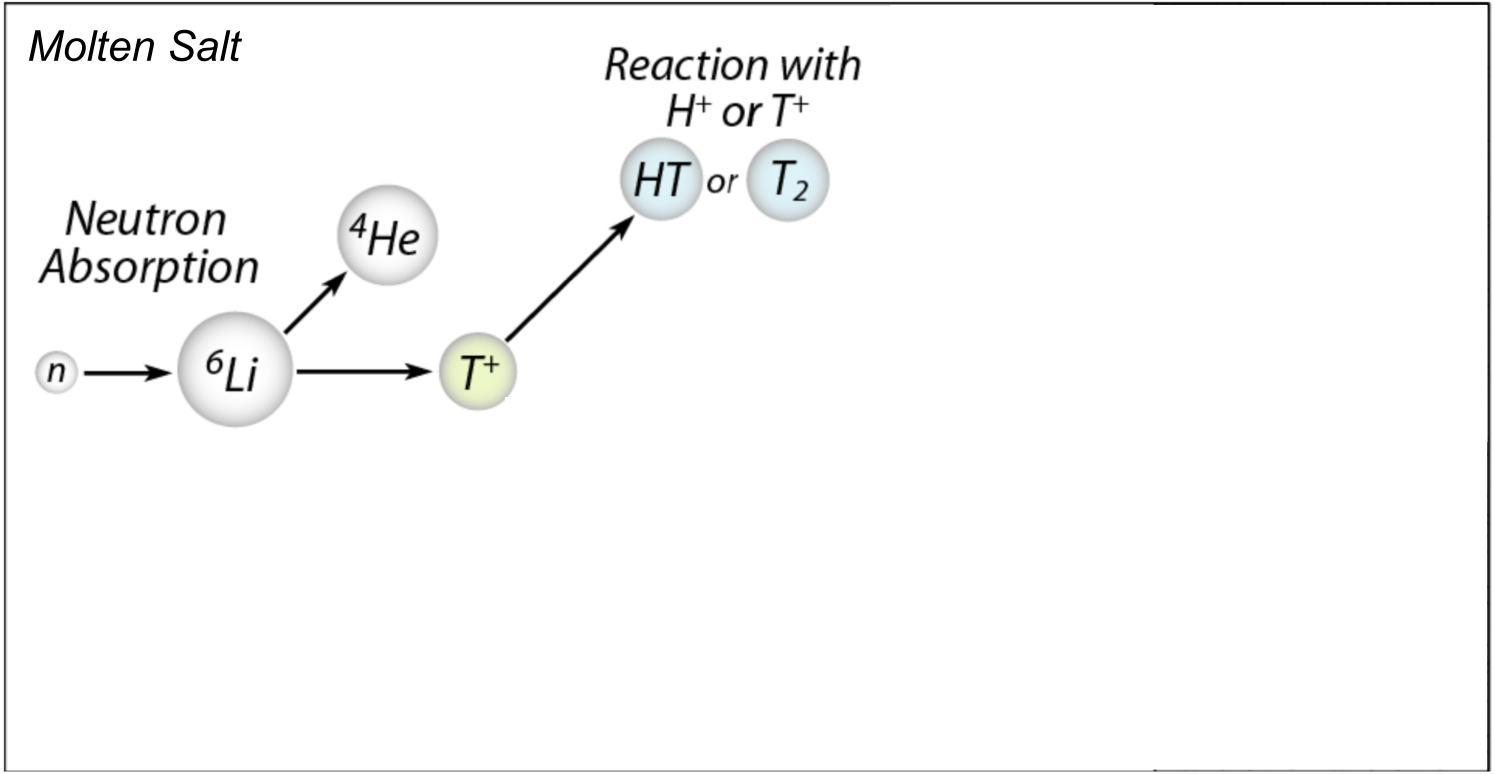
For MSRs that contain lithium or beryllium within the molten salt, it is possible to generate tritium at rates far exceeding current U.S. LWR systems (on a per GWe basis) due to neutron interactions with ^{6}Li. In addition, tritium generated through this pathway will be present within the molten salt and not contained within fuel or control elements

TRITIUM TRANSPORT: CHEMICAL FORM

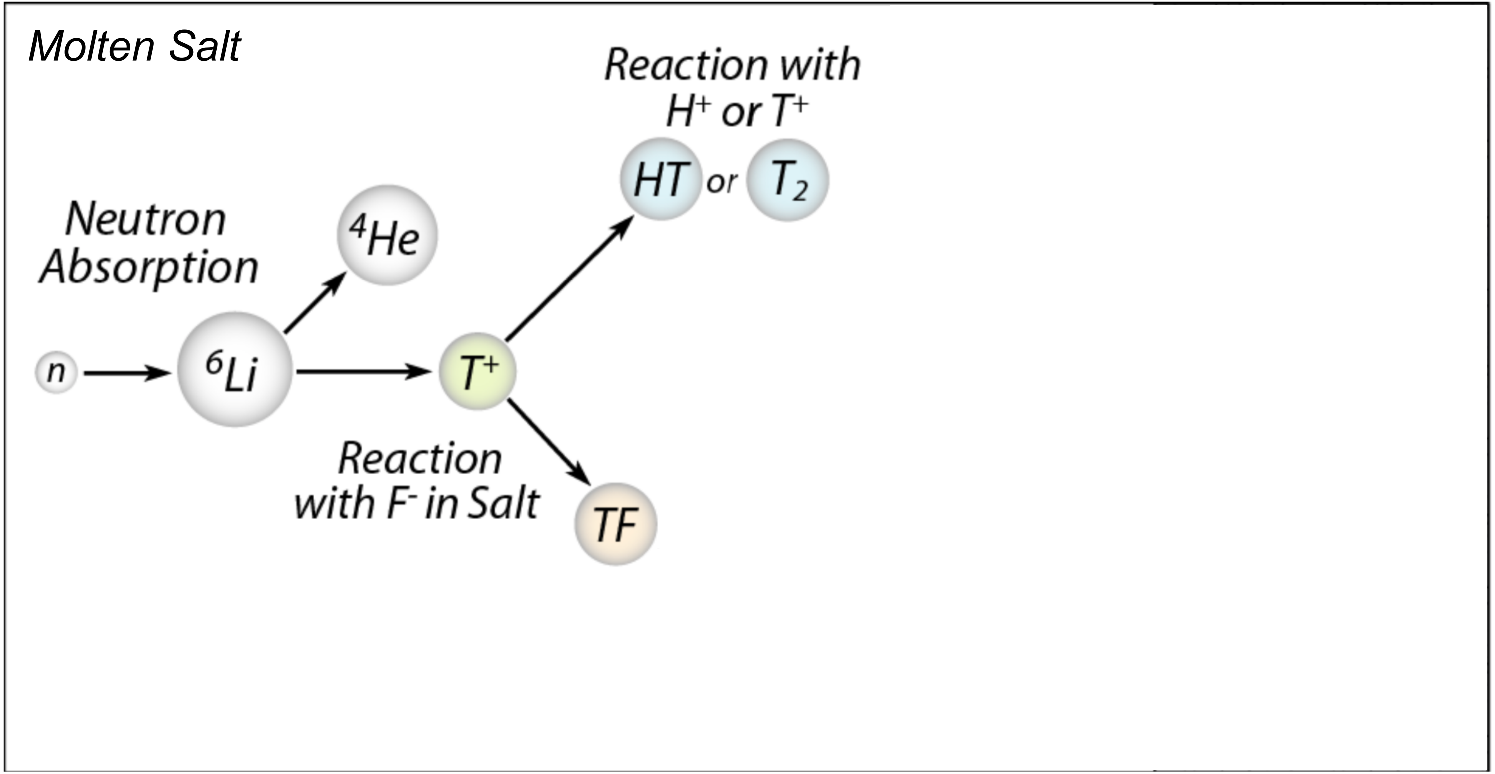
Molten Salt



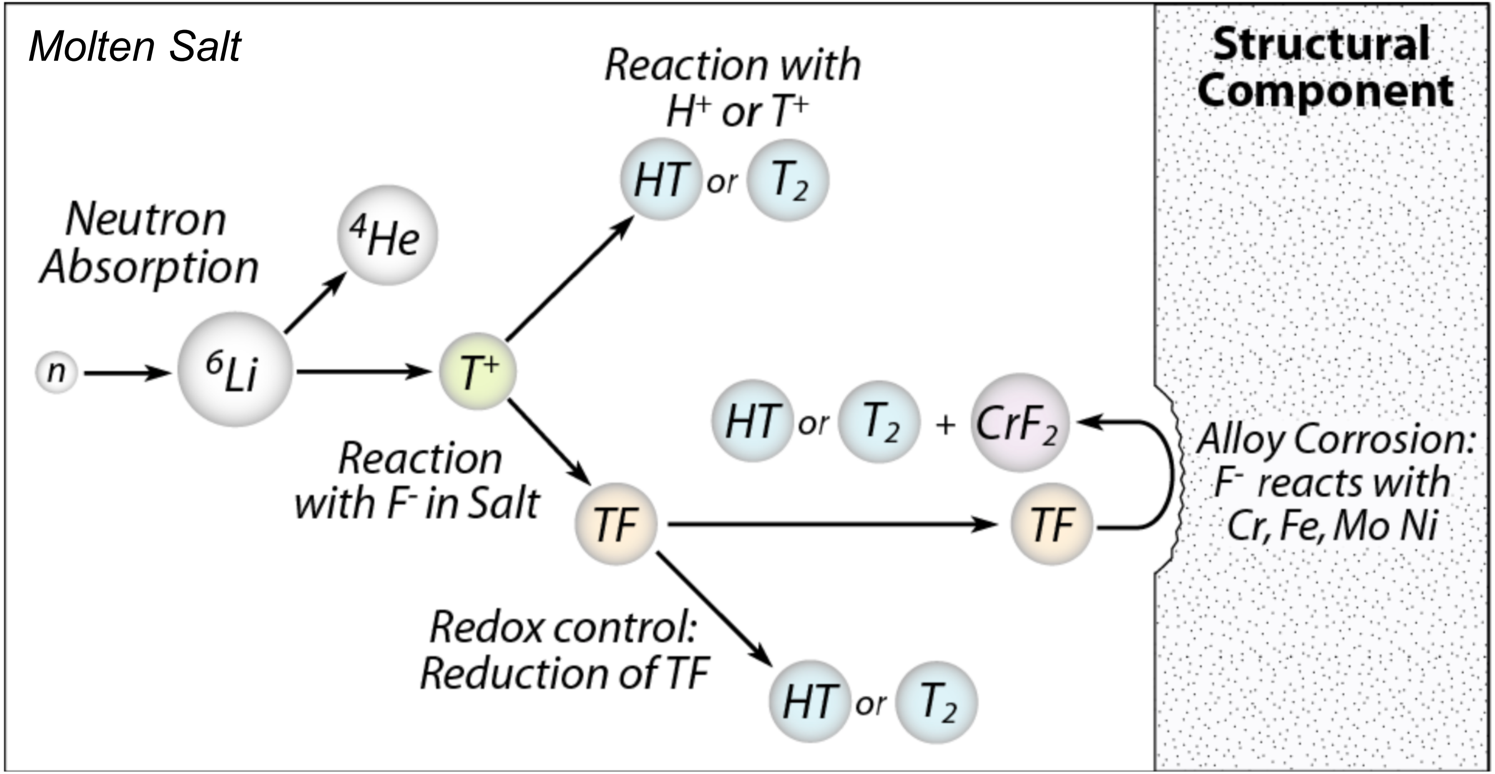
TRITIUM TRANSPORT: CHEMICAL FORM



TRITIUM TRANSPORT: CHEMICAL FORM



TRITIUM TRANSPORT: CHEMICAL FORM



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TRITIUM TRANSPORT: CHEMICAL FORM

■ Tritium Fluoride (TF)

- Likely form of tritium born from ^{6}Li reactions
- Low permeability through structural materials
- A powerful oxidizer and **principle cause of corrosion in MSRs**

■ TF Corrosion

- Unlike LWRs, corrosion products are soluble in salt, which then expose underlying metal
- Measures must be taken to reduce (in the chemical sense) TF before it interacts with structure
- Multiple techniques available for "redox control" but **all reduce TF to molecular HT/T_2**

■ Molecular HT/T_2

- Highly permeable through structural materials at the operating temperatures of MSRs, **increasing likelihood of tritium escaping the reactor system**

TRITIUM TRANSPORT: CHEMICAL FORM

Key Point

The production of tritium within the molten salt is inextricably tied to corrosion concerns due to the formation of TF, a powerful oxidizer. Corrosion control strategies will likely result in the reduction of TF to a molecular hydrogen form (HT/T_2), which are highly permeable in structural materials at the operating temperatures of most MSR designs

TRITIUM TRANSPORT: BARRIERS

Similarities and Differences

- Salt-fueled and salt-cooled MSRs share some of the same tritium barriers and transport phenomena

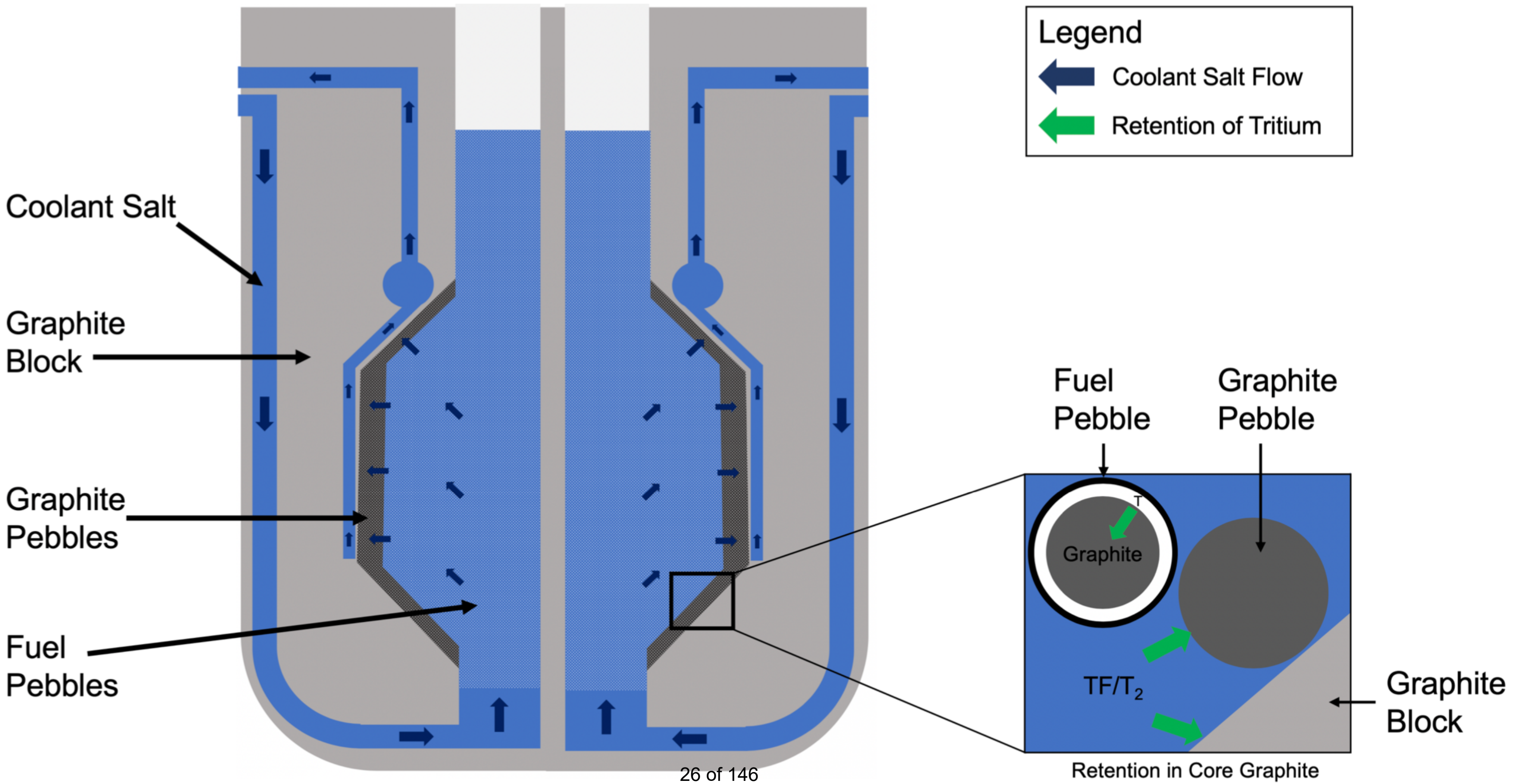
Notational Diagrams

- Following diagrams outline high-level transport and retention pathways

Importance of Graphite

- Experience with MSRE demonstrated high tritium retention within core graphite
- High specific surface area of graphite offers many bonding sites for tritium
- Tritium can be liberated from graphite at high temperatures (above normal operating temperatures)
- Many factors influence graphite retention capabilities, such as form and irradiation history
- In general, nuclear grade graphite has lower retention than activated forms of carbon due to the annealing process, which is necessary for irradiation stability in core

TRITIUM TRANSPORT: BARRIERS (PB-FHR)



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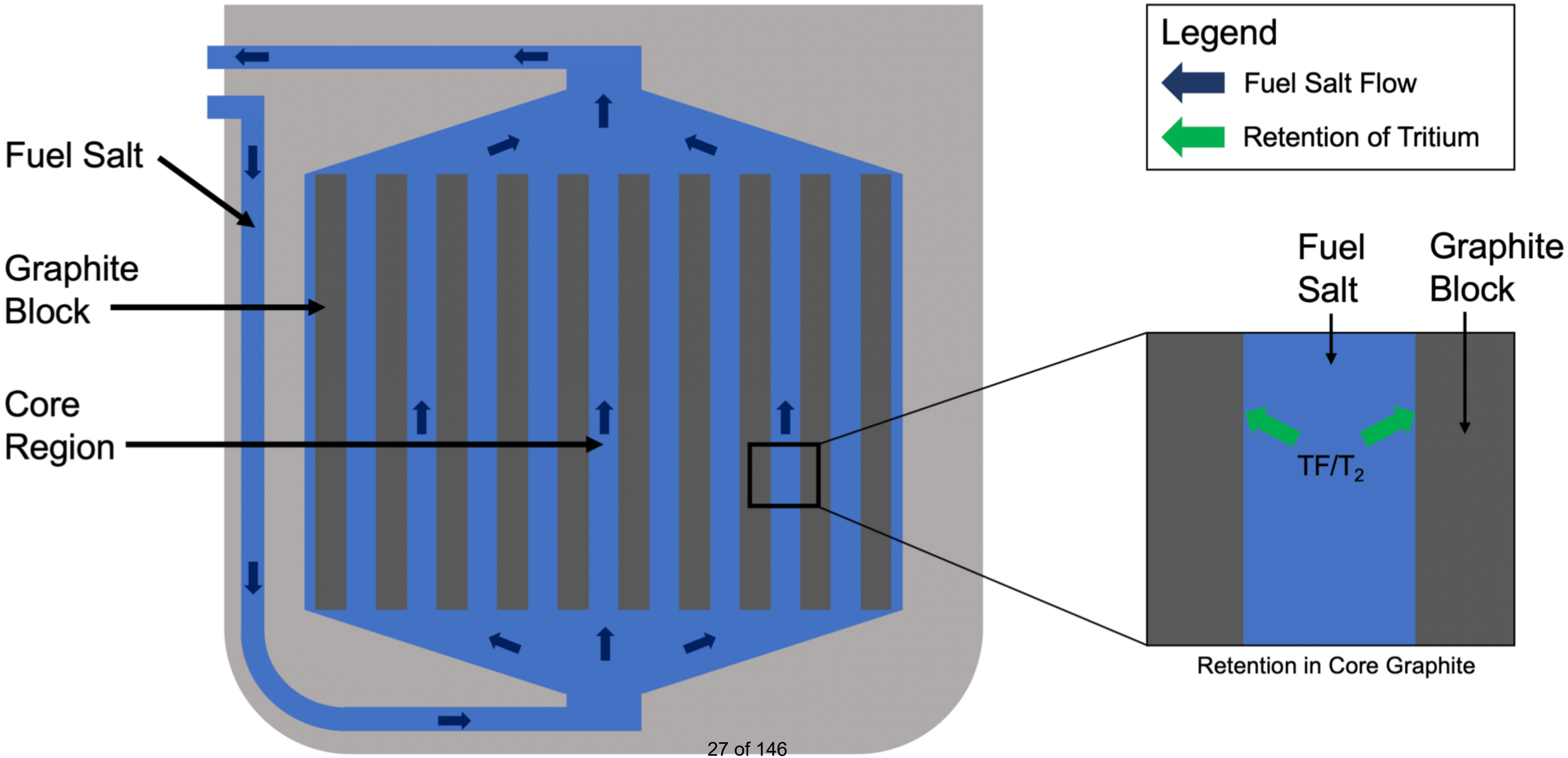
TRITIUM TRANSPORT: BARRIERS (FUEL-SALT MSR)



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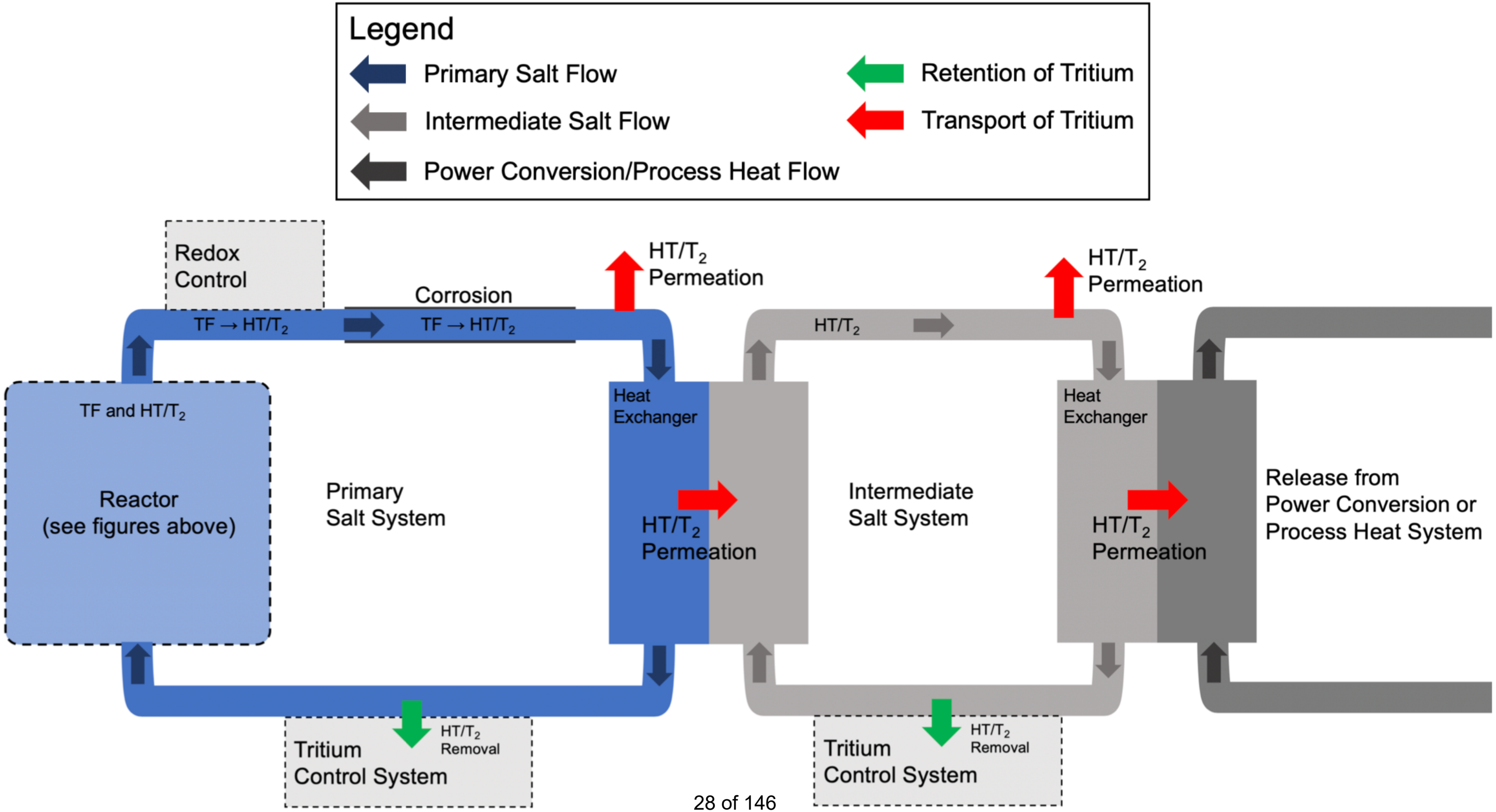
TRITIUM TRANSPORT: BARRIERS



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TRITIUM TRANSPORT: CHEMICAL FORM

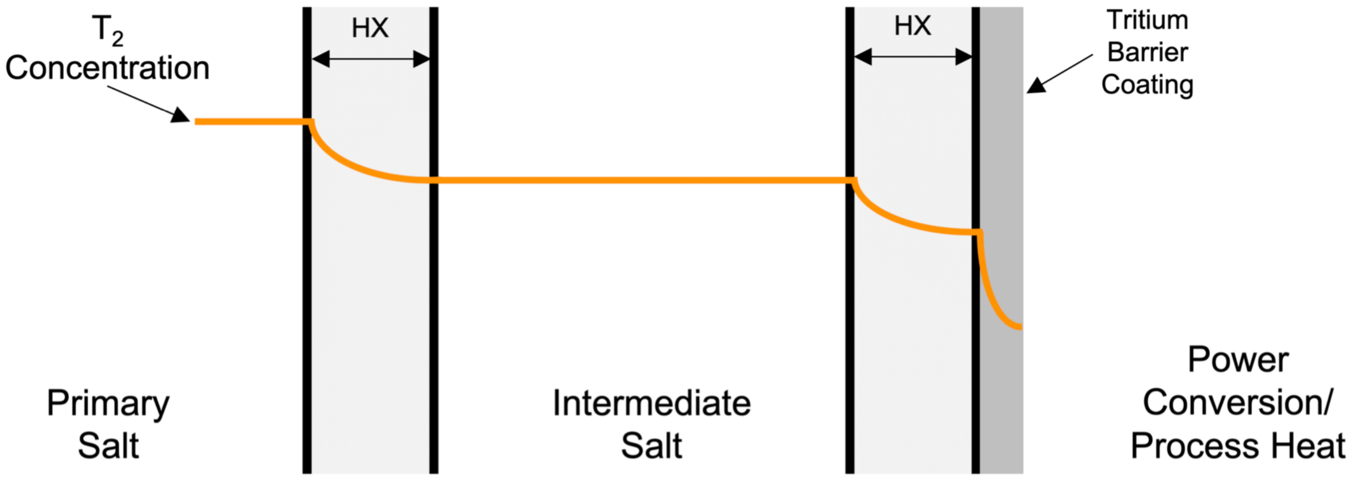
Key Points

- Due to the large quantity of tritium in the system and the mobile chemical form, tritium control and removal strategies are necessary to prevent the relocation of tritium to areas outside of the reactor system and potentially to the environment
- Graphite within thermal MSR systems likely offers an initial retention mechanism

TRITIUM TRANSPORT: CONTROL

■ Coatings

- Use of coatings or barriers that have low hydrogen/tritium permeability
- Most historical tritium coatings are not compatible with molten salts (oxides, aluminum)
- Coatings may need to be placed on surfaces not in contact with molten salt

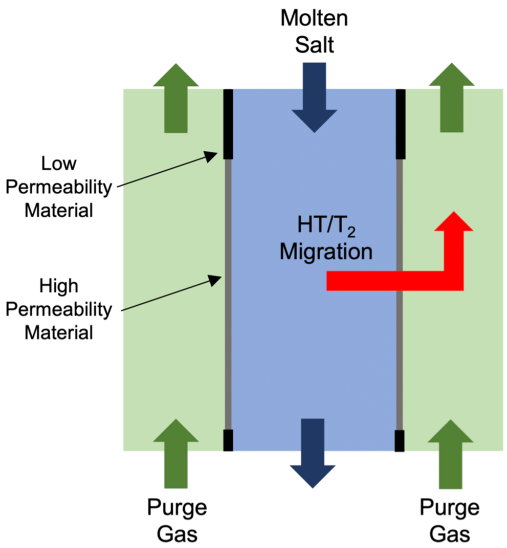


TRITIUM TRANSPORT: CONTROL

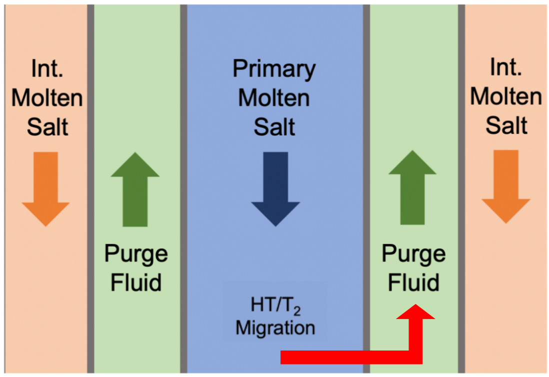
■ Permeators

- Use a combination of high and low permeability materials to direct tritium transport
- Use of low-pressure purge gas or vacuum can encourage tritium removal in certain areas of system
- Can be integrated into a double-wall heat exchanger

Permeator Tritium Removal System



Double-wall Heat Exchanger Tritium Removal System



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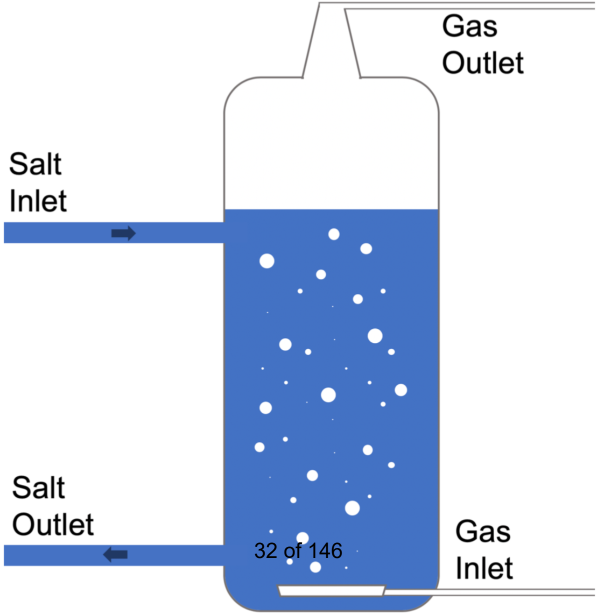
TRITIUM TRANSPORT: CONTROL

Gas Sparging/Stripping

- Bubbling an inert gas, such as helium, through the molten salt encourages the movement of tritium from the salt to the sparge gas
- Technique dependent on contact surface area between gas and salt
- Also can invert the process by spraying salt through a gas volume or vacuum



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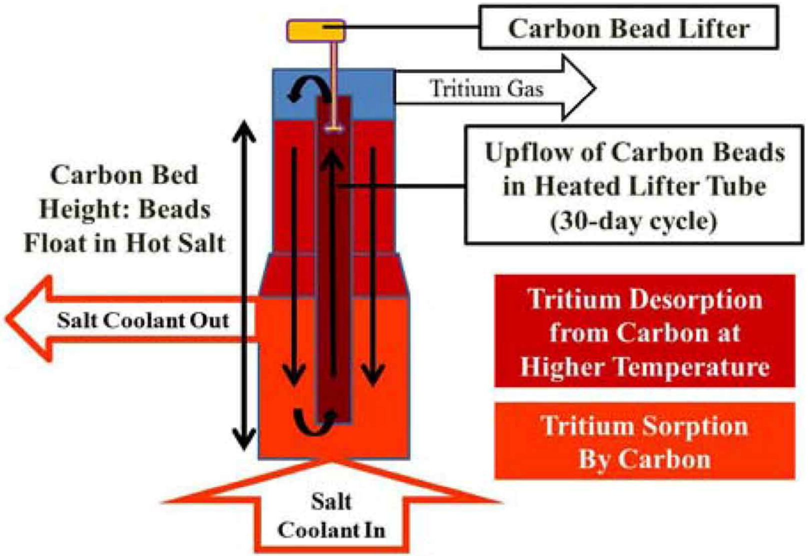
TRITIUM TRANSPORT: CONTROL

■ Adsorber Bed

- Utilizes a bed (such as spheres) made of material with high tritium retention
- Could use activated carbon rather than nuclear grade graphite since placed away from the core
- Once saturated, spheres could be removed and stored or heated to liberate tritium



Source: C. Forsberg et al., "Tritium Control and Capture in Salt-Cooled Fission and Fusion Reactors: Status, Challenges, and Path Forward," Nuclear Technology, vol. 197, no. 119-139, 2017

TRITIUM TRANSPORT: MONITORING AND STORAGE

■ Monitoring

- Infeasible to directly measure tritium concentration in molten salt, due to self-shielding of low energy beta emission from the salt
- Instead, tritium concentrations likely derived from the tritium removal system, such as the off-gas stream
- Flow-through detectors are needed due to low energy beta, although alternative approaches are being explored (optical spectroscopy)

■ Storage

- For CANDUs, removed tritium is stored as a metal hydride (tritide)
- Metals, such as titanium, form metal hydrides when exposed to hydrogen/tritium and can retain incredible amounts of hydrogen (densities greater than that of liquid hydrogen)
- Metal hydrides are stable at room temperature and pressure, but the process is reversible and tritium can be liberated if heated above 500ºC
- Other storage avenues possible, such as within low-water cement

TRITIUM TRANSPORT: CONTROL

Key Point

Numerous tritium control and removal concepts exist, with varying levels of technology readiness. An MSR tritium control strategy will likely include multiple components or systems to both retain tritium within the salt and remove it at designated locations.

MSR Tritium Phenomena

- Production
- Transport
- Control

MSRE Experience

- Design
- Operation
- Tritium

Modeling and Simulation

- Requirements
- Current Capabilities
- Assessment

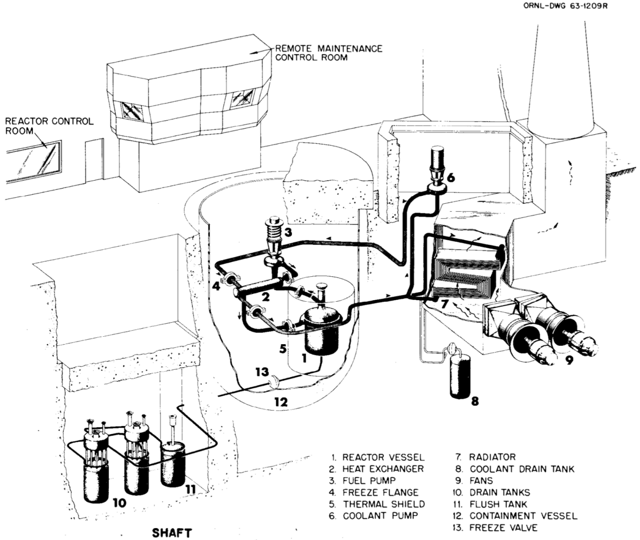
Regulatory Considerations

- Quantitative Limits
- Regulation
- Assessment

MOLTEN SALT REACTOR EXPERIMENT (MSRE)

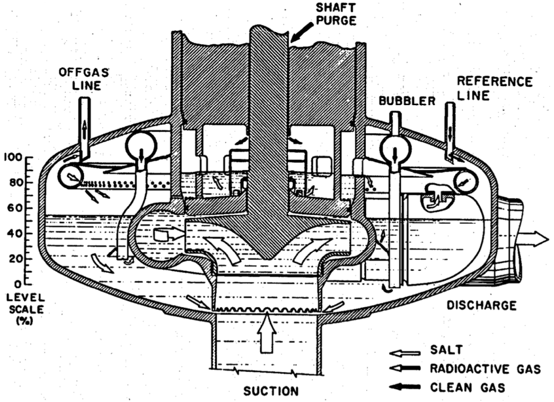
■ Design

- Built at Oak Ridge National Laboratory
- Salt-Fueled: LiF-BeF_2-ZrF_4-UF_4 (99.9923% ^{7}Li)
- 7.34 MWth, no power conversion
- Graphite moderated
- FLiBe secondary system



■ Operation

- Operated 1965-1969 (~17,000 critical hours)
- Used both ^{235}U and ^{233}U at different stages
- Gas space of primary pump used for off-gas system



MSRE: TRITIUM EXPERIENCE

Tritium Balance

- During final MSRE runs, a study performed to examine tritium transport
- Through measurements of reactor systems, the study attempted to determine where the produced tritium was going
- Tritium production was estimated through neutronic calculation and compared to measured quantities

Item	Ci/d	%
Production (estimate)	54	100
Known Disposition		
Discharged from fuel offgas system	26	48
Discharged from coolant offgas system	1	1
Discharged in coolant radiator air	4	7
Appearing in cell atmosphere	5	9
Retained in graphite	8	14
Total	43	80
Difference (unaccounted[1])	11	20

[1]Assumed to be present within fuel offgas system as part of oil residue.

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MODELING AND SIMULATION

■ Functional Requirements

- The final report outlines functional requirements for the modeling and simulation of tritium in MSRs based on production and transport phenomena
- The functional requirements aid in the identification of necessary code capabilities

■ Code Survey

- The current code landscape was examined
- Multiple MSR tritium analysis stand-alone codes or packages currently under development
- Development of data for code validation is a need recognized by the MSR industry

■ Tritium Production Assessment

- To gauge current capabilities, a trial calculation was performed of tritium production in MSRE
- Utilized MCNP 6.2, ORIGEN-S/COUPLE

Calculation Method	Tritium Production Rate: ^{235}U Fuel (Ci/d)
Single Flow Passage Model	27.1
Whole Core Model	29.0
ORNL Estimate	31.7

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REGULATION: QUANTITATIVE CONSTRAINTS

Multiple Regulatory Sources

- Almost all constraints are dose or dose-derived
- Only exception is tritium release to sewers, not shown in table (limit of 5 Ci per year)
- Some constraints are cumulative across all releases or all beta releases

Type	Regulation	Annual Dose (mrem)	Effluent Concentration (µCi/ml)	
			Air	Water
Limit	10 CFR 20.1301(a)1	100	-	-
	10 CFR 20 Appendix B Table 2	50	1E-7	1E-3
Standard	10 CFR 20.1301(e)/40 CFR Part 190.10(a)	25	*5E-8*[a]	*5E-4*[a]
ALARA	10 CFR 50 Appendix I Sec II. B.1	20 (β, air)[b]	*4E-8*[a]	-
	10 CFR 50 Appendix I Sec II. B.2(b)	15 (β, air)[b]	*3E-8*[a]	-
	10 CFR 50 Appendix I Sec II. A	3[c]	-	*6E-5*[a]
Drinking Water	EPA Standard	4	-	2E-5

See report (ANL/NSE-20-15) for table footnotes

REGULATION: EFFLUENTS

- **10 CFR 50.34a and 50.36a**
  - Applicants must identify design objectives to keep effluent releases to unrestricted areas ALARA
  - ALARA, in this context, allows for the consideration of the state of technology and economics in relation to public health and safety and public benefits of atomic energy
  - Appendix I limits meet these objectives

- **Applicant must describe:**
  - Equipment utilized to achieve ALARA requirements
  - Estimate of annual liquid and gases effluent releases
  - Description of packaging, storage, and shipment of waste from treating effluents

- **Expectations**
  - Licensee shall be guided by past experience, which indicates the typical releases are only a small percentage of 10 CFR 20.1301 limits

REGULATION: MINIMIZATION OF CONTAMINATION

- **10 CFR 20.1406**
  - Applicants shall describe how facility design and procedures will minimize, to the extent practical, generation of waste and contamination of the facility and the environment

- ***To the extent practical…***
  - RG 4.21 provides guidance, "other competing concerns, such as the implication to safety systems and the overall cost should be considered. Thus the minimization of facility contamination must be considered in the context of overall facility safety."

- **RG 4.21 Guidance**
  - Utilizes a risk-informed, performance-based approach
  - Minimizing facility contamination through use of SSC and operational procedures
  - Minimizing environmental contamination through understanding of radionuclide transport and use of a conceptual site model
  - Facilitation of decommissioning considered in the design process
  - Minimizing generation of waste, however NRC recognizes the constraints and competing factors to waste minimization

REGULATION: OTHER FACTORS

- **10 CFR 20.1701: Restricting Internal Exposures**
  - Licensee shall use, to the extent practical, process and engineering controls to control radioactive material in air

- **10 CFR 50 – Appendix A: GDCs**
  - RG 1.232 found no need to modify effluent GDCs for non-LWRs

- **Others…**
  - Assessment of tritium in PRA as part of Licensing Modernization Project (LMP) process
  - Storage of removed tritium, DC/COL-ISG-013/014
  - Monitoring effluents: RG 1.21, 4.1, 4.15, 1.109

REGULATION: ASSESSMENT FINDINGS

Key Points

- Limits on tritium release to the environment are primarily dose- or concentration-based, rather than centered on cumulative activity released. This is essentially a performance-based system, which is not LWR-specific and could allow MSR vendors the necessary flexibility to develop tritium control strategies.

- Current regulation requires a description of the systems and procedures in place to limit radioactive releases, including an estimate of predicted effluents during operation. This would encompass tritium control strategies and systems.

- Regulation and guidance on the release of radioactive effluents to the environment permits the use of a risk-informed performance-based evaluation to minimize releases to the extent practical. Although there may be subjectivity in the determination of “practicality”, the diversity in MSR designs and tritium control strategies likely makes generic guidance on this issue difficult.

SUMMARY

■ Tritium in MSRs

- For MSRs that contain lithium or beryllium in the salt, the production of tritium must be considered
- Due to corrosion concerns, tritium will be converted to a mobile molecular form
- There are many options available for the control and removal of tritium
- Development of modeling tools and validation data is an ongoing project

■ Regulation

- Current regulatory environment appears adequate to address tritium concerns in MSRs
- Generally performance-based dose limits on tritium release
- Existing requirements for license applicants to minimize releases and describe the strategies and systems utilized to control releases
- Flexibility to consider plant operation and economics when developing control strategies

QUESTIONS?

Discussions on Mechanistic Source Term Methodologies and Associated Information

INTRODUCTION

NRC'S Vision and Strategy and the development of mechanistic source terms for non-LWRs

- Development of sufficient computer codes and tools

Staff interactions with ACRS

- Related to mechanistic source term (MST) methodologies
- Expanding guidance for developing MSTs
- Expectations for the technical adequacy in using MST
- Tools for staff confirmatory analysis

NEIMA requirement

- Evaluation on developing and implement guidance for the resolution of issues relating to the use of MST

INTRODUCTION (Cont'd)

Development of final reports

- SAND2020-0402, Simplified Approach for Scoping Assessment of Non-LWR Source Terms
- INL/EXT-20-58717, Technology-Inclusive Determination of Mechanistic Source Terms for Offsite Dose-Related Assessments for Advanced Nuclear Reactor Facilities

Path forward

- Use INL and SNL reports as additional aid in resolving MST issues, and for developing design-specific MST methodologies
- Methods, results, and conclusions of the staff's pilot studies and use of MELCOR will be publicly shared

August 20, 2020
Kurt Vedros
Andrea Alfonsi
Paul Humrickhouse
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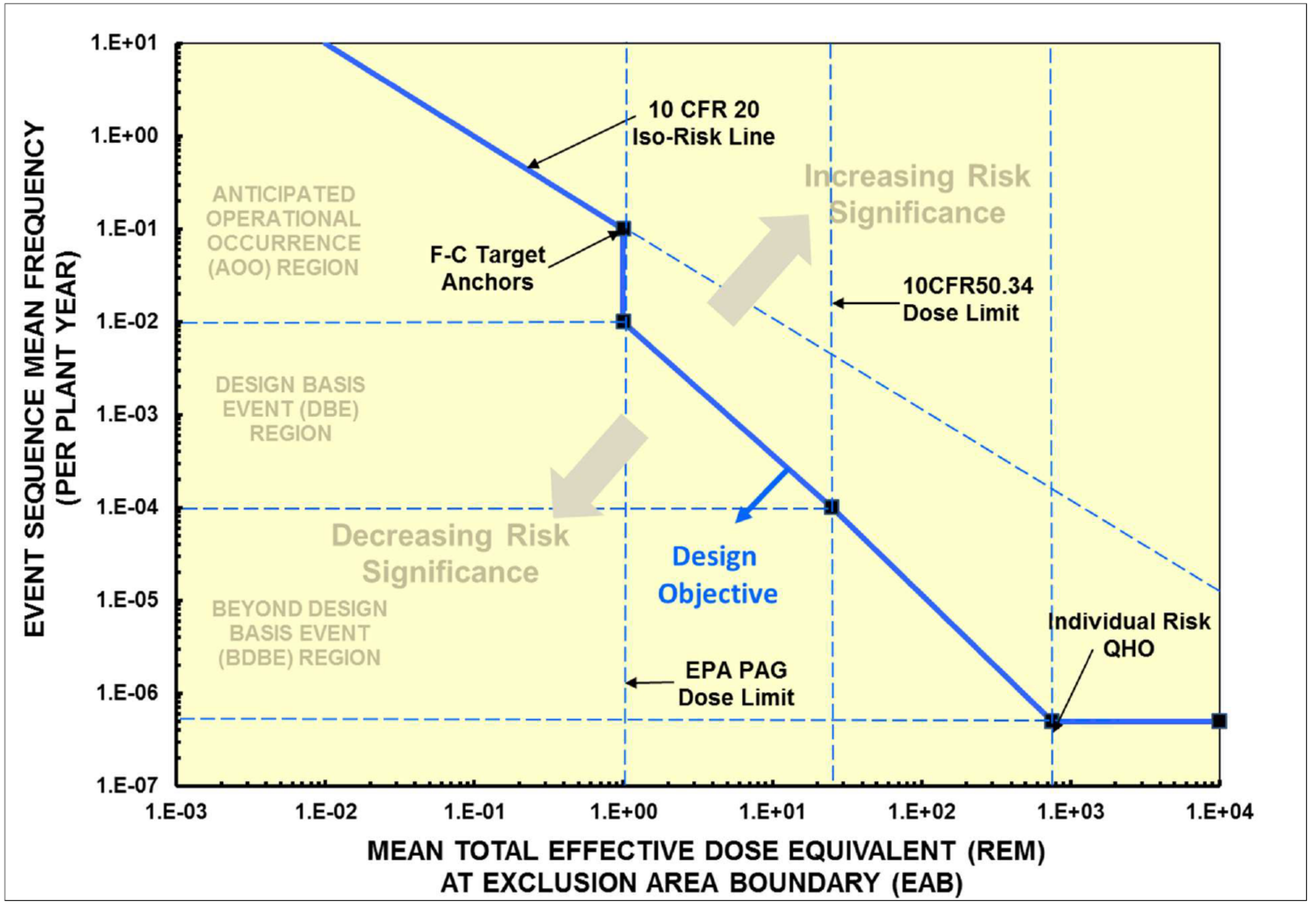
Technology-Inclusive Determination of Mechanistic Source Terms for Offsite Dose-Related Assessments for Advanced Nuclear Reactor Facilities

Idaho National Laboratory

Objective

- Document written as a project for NRC team: INL/EXT-20-58717
- Develop a risk-informed, performance-based, and technology-inclusive approach to determine source terms for dose-related assessments at advanced nuclear facilities to
    1) support the NRC's Non-LWR Vision and Strategy Near-Term Implementation Action Plans (ADAMS Accession No. ML16334A495) and,
    2) the NRC's response to the Nuclear Energy Innovation and Modernization Act (NEIMA) Public Law No: 115-439, of January 2019

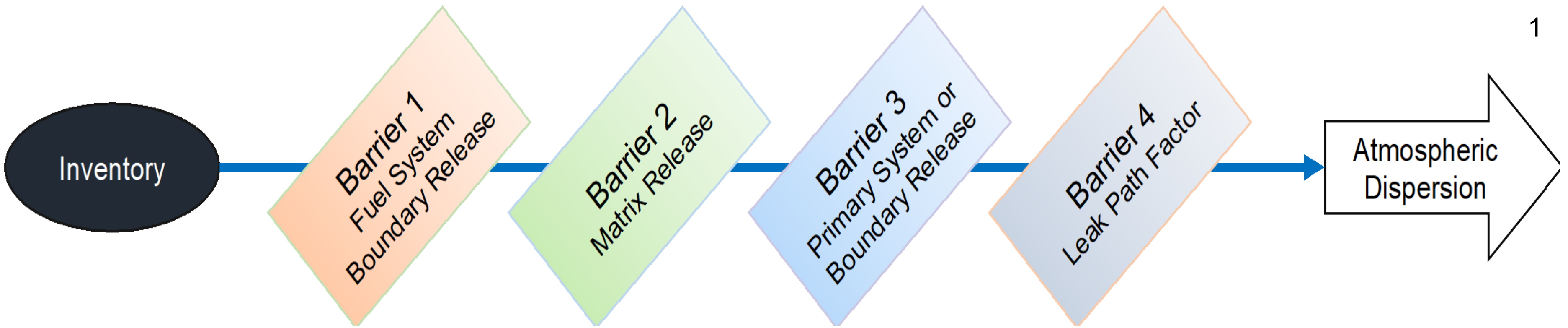
F-C target curve (NEI 18-04).



Definitions

- **Source Terms for Advanced Reactors**: the quantities, timing and other characteristics of radionuclides released from the facility to the environment.
- **Non-Mechanistic Source Terms Methodology**: adopt conservative approaches and assumptions based on known physical and chemical principles.
- **Mechanistic Source Terms Methodology**: consider design-specific scenarios and use best-estimate models with uncertainty quantification for a range of licensing basis events to be used for the design and licensing of advanced nuclear technologies.

Illustration of radionuclides retention and removal process for one non-LWR concept



Mechanistic source terms can be correlated using[1]:

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

[1]SAND2020-0402

Illustration of radionuclides retention and removal process for one non-LWR concept

where:

$ST(S_i, RN_j, t)$ is the total release to the environment of radionuclide RN_j over the entire release duration time (t)

$I(RN_J)$ is the initial fission product inventory at the time of the reactor accident for radionuclide RN_j

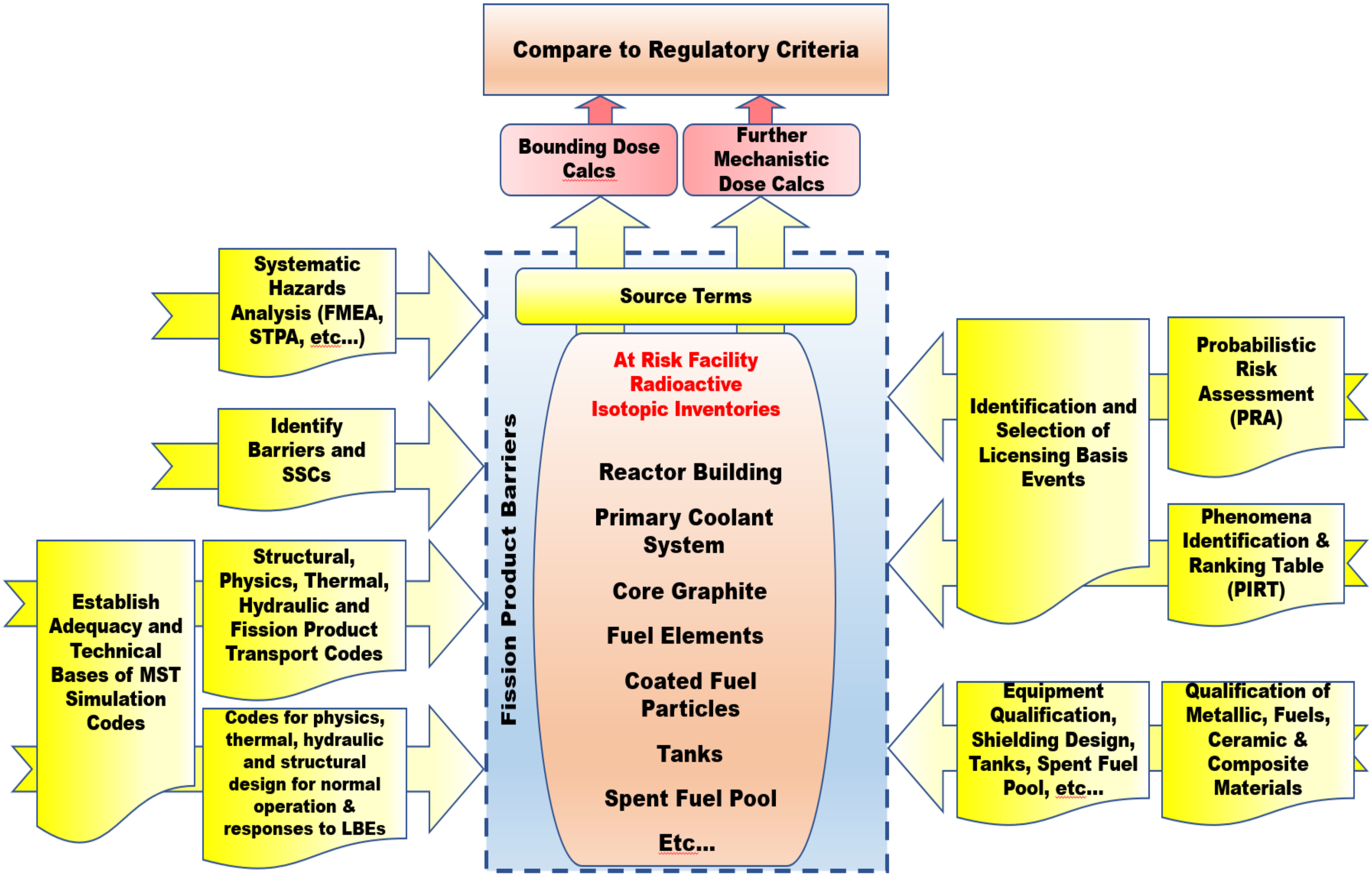
$F(S_i, RN_j, t)$ is the fraction of release of radionuclide RN_j from fuel system boundaries to the fuel matrix

$MR(S_i, RN_j, t)$ is the fraction of release of radionuclide RN_j from fuel matrix to primary system

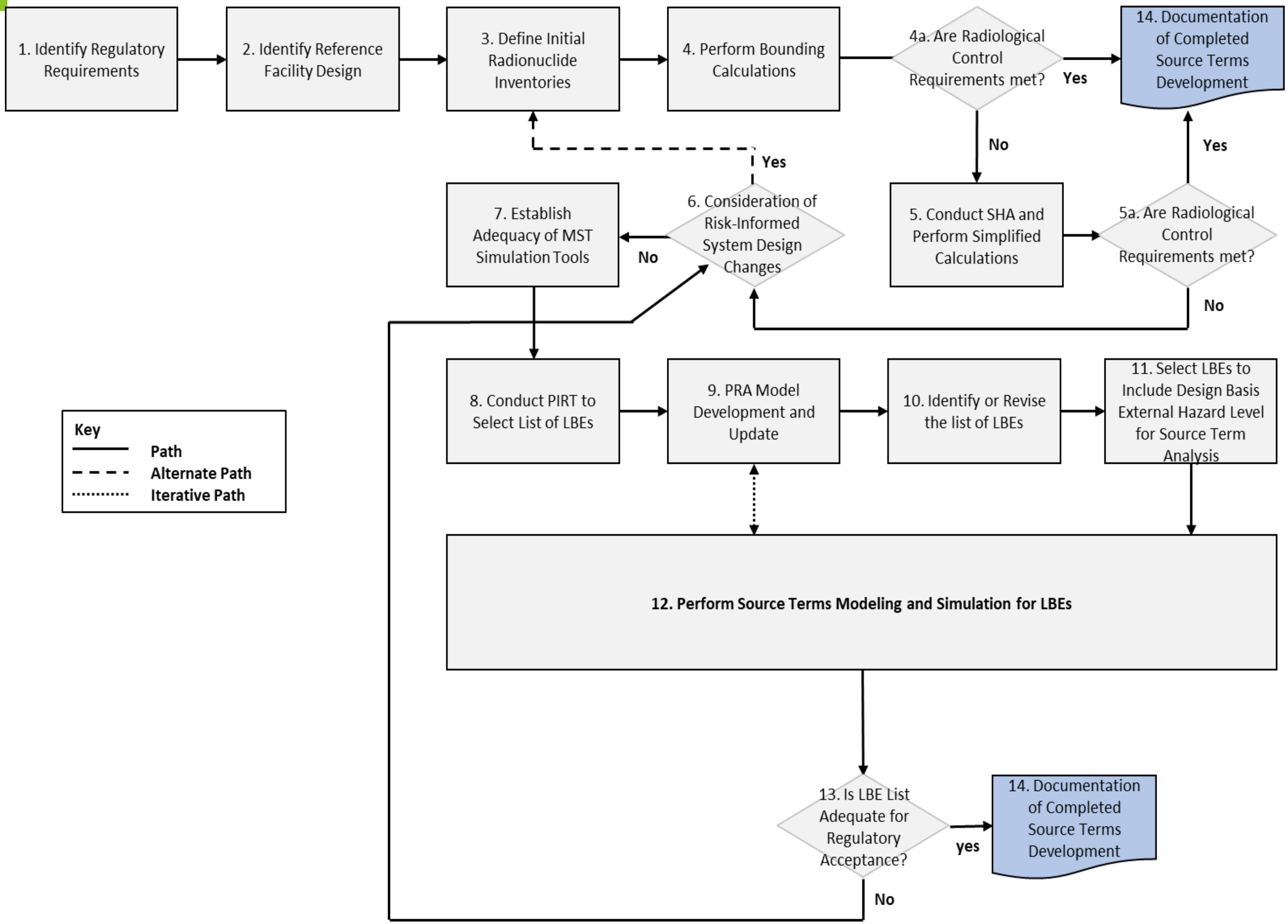
$PSR(S_i, RN_j, t)$ is the fraction of release of radionuclide RN_j from primary system to leak path

$LPF(S_i, RN_j, t)$ is the fraction of release of radionuclide RN_j from leak path to the environment

Technology-inclusive source terms determination methodology components



Technology-inclusive source terms determination methodology



Step 1: Identify Regulatory Requirements

Top-Level Regulatory Requirements		Comment
1	10 CFR 30, Schedule C	Emergency plan
2	10 CFR 50.34(a)(1)(ii)(D) TEDE ≤ 25 rem at EAB over worst two-hour dose period TEDE ≤ 25 rem at outer edge of low population zone (LPZ) for the duration of the passage of the plume	Facility siting Offsite dose criteria
3	10 CFR 50, Appendix I, LWR Design Objectives for Radionuclides in Plant Effluents, dose to individual in unrestricted area: Whole Body Dose ≤ 5 mrem/yr Dose to any organ ≤ 15 mrem/yr	Plant effluents
4	10 CFR 20 Subpart C Occupational Dose Limits: Total effective dose equivalent (TEDE) <5 rem/yr Organ Dose ≤ 50 rem(/yr)	Standards for occupational protection
5	10 CFR 20 Subpart D Public Dose Limits: Annual TEDE ≤ 0.1 rem Hourly External Dose ≤ 0.002 rem	Standards for public protection
6	40 CFR 190 Subpart B Environmental Standards for the Uranium Fuel Cycle, (LWRs), normal operations, annual dose equivalent: Whole Body ≤ 25 mrem Thyroid Dose ≤ 75 mrem Organ Dose ≤ 25 mrem	Standards for fuel cycle
7	10 CFR 52.47 Offsite Dose Criteria for LBEs, standard design certification: TEDE ≤ 25 rem for 2 hours at the EAB TEDE ≤ 25 rem for duration of passage of plume at the LPZ boundary	Offsite dose criteria*
8	EPA PAGs for Radioactive Release for Public Sheltering & Evacuation (EPA 2017): TEDE over four days ≤ 1 rem Thyroid Dose ≤ 5 rem	Public shelter & evacuation
9	NRC Safety Goal Policy Statement (NRC 1986)	Safety goal

Step 2: Identify Reference Facility Design

- The developer defines the reference facility design
- Identifies:
  - All foreseeable facility system operating modes
  - Barriers
  - Engineered safety features within barriers
    - SSCs of these systems, or needed for these systems

Step 3: Define Initial Radionuclide Inventories

- Determine equilibrium radionuclide inventories (or appropriate values if equilibrium conditions are not achieved for a particular plant design) in all plant systems (e.g., fuel, barrier 1, barrier 2, etc.) during normal steady-state operation.
  - Description is provided of initial inventories
    - e.g., equilibrium nominal end of life

Step 4. Perform Bounding Calculations

- These bounding calculations are performed to determine the dose consequences of the releasing radionuclide inventories identified by the previous step for the “maximum credible accident (MCA)”
  - The MCA is postulated as a nuclear accident that would result in a potential hazard that would not be exceeded by any other accident considered credible during the lifetime of the facility.
- Demonstrate compliance with the established regulatory criteria. If criteria met, proceeds to documentation.

Step 5. Conduct SHA and Perform Simplified Calculations

- Conduct a SHA (FMEA, STPA, or equivalent) to identify potential SSC failure modes that lead to radionuclide releases, as well as to identify a spectrum of postulated LBEs.
  - Consider the behavior of the barriers after SHA and determine dose consequence by using simplified methods.
    - Simplified methods are still bounding calculations based on proven physical properties.
    - Inventory release to environment is modified from MCA by behavior of design barriers identified in SHA.
- If criteria met, proceeds to documentation.

Step 6. Consider Risk-informed System Design Changes

- Consider a system redesign to include additional barriers or SSCs as identified by hazard analysis, which will either return to Step 3 or proceed to Step 7.

Step 7. Establish Adequacy of MST Simulation Tools

- Identify any gaps from MST simulation tools criteria[1]:
  - The performance of the reactor and fuel under normal and off-normal conditions is sufficiently well understood to permit a mechanistic analysis.
  - 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.
- Develop and complete analytic and testing programs to fill identified gaps in available MST simulation tools.

[1] SECY-93-092

Step 8. Select Initial List of LBEs and Conduct PIRT

- Develop initial list of LBEs which may not be complete but are necessary to develop the basic elements of the safety design
- Conduct Phenomena Identification and Ranking (PIRT) exercise to identify safety-significant phenomena for the LBEs
- Assess importance, knowledge level, and status of modeling for each phenomenon:

#	Phenomenon	Importance	Rationale	Knowledge Level	Rationale	Model Status
1	Transport phenomenon A	High	Primary barrier for radionuclide transport	Low	Lack of, or uncertain, experimental data	Major need
2	Transport phenomenon B	Medium	Minor barrier for radionuclide transport	Medium	Some experimental data available	Minor need
3	Transport phenomenon C	Low	No credit taken for barrier C in source term analysis	High	Well characterized experimentally	Adequate

Step 9. Develop and Update PRA Model

- PRA is used to model LBEs in a probabilistic manner.
- Utilize the PRA group of analyses that inform the logic model which informs consequence modeling.
- Static PRA is used for design and regulatory decisions.
- Dynamic PRA can be used to validate the outcome of sequence end states.
- Adhere to the most current Non-LWR PRA Standard (ASME/ANS RA-S-1.4-2013) when any conflicts are encountered between standards.

Step 10: Identification or revision the list of LBEs

- The identification process:
  - needs to be considered as an integral part of the overall design process and,
  - should be “re-iterated” since its selection informs the design requirements of safety-related and non-safety-related SSCs

Identification Process

Selection of Initial Event set	LBEs updated if the design changes	LBEs are reviewed at the end of the design phase

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

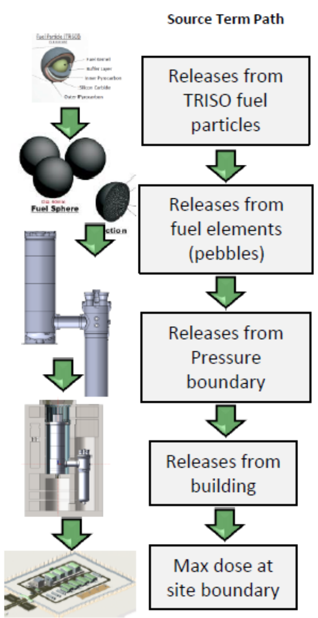
- A set of design basis external hazard levels (DBEHLs) will be selected to form an important part of the design and licensing basis:

- As supported by methods, data, design, site information, and supporting guides and standards, these DBEHLs:
  - will be informed by a probabilistic external hazards analysis and
  - will be included in the PRA using design features that are incorporated to withstand these hazards

- Other external hazards not supported by a probabilistic hazard analysis will be covered by DBEHLs that are determined using traditional deterministic methods.

Step 12: Perform LBEs Source Term Modeling and Simulation

- The source term assessment needs to characterize the generation, release, transport, and retention of fission product and activation radionuclides
- The process for the development of modeling and simulation tools for non-LWR applications is like LWR applications.



Element / Isotope	Form / State	Mechanism	Physical Phenomena	Methods / Software Codes
Iodine Silver Strontium Cesium	Gaseous FPs Metallic FPs	- Release from TRISO particles into matrix graphite - Activation of impurities	Temperature, irradiation time, fast fluence, burnup, particle defects, contamination	VSOP-A, VSOP-99, MGT SCALE, PARCS, OREGIN FLOWNEX, STAR-CCM+ XS-Term
Iodine, Silver Strontium Cesium	Gaseous FPs Metallic FPs Dust Particles	- Diffusion from pebble into the helium stream - Activation of impurities	Temperature, irradiation time, fast fluence, burnup, contamination	VSOP-A, VSOP-99, MGT SCALE, PARCS, OREGIN FLOWNEX, STAR-CCM+ XS-Term
Iodine, Silver Strontium, Cesium	Gaseous FPs Metallic FPs Dust Particles	- Leakage from HPB into building and structures - Activation of impurities	Instrumentation line failure, small & large pipe breaks, plate-out, liftoff	ORIGEN XS-Term
Iodine, Silver Strontium, Cesium	Gaseous FPs Metallic FPs Dust Particles	- Transport throughout building to the environment	Plate-out, liftoff	XS-Term MELCOR
Iodine, Silver Strontium, Cesium	Gaseous FPs Metallic FPs Dust Particles	- Atmospheric dispersion - Ingestion	Postulates	XS-Term STAR-CCM+

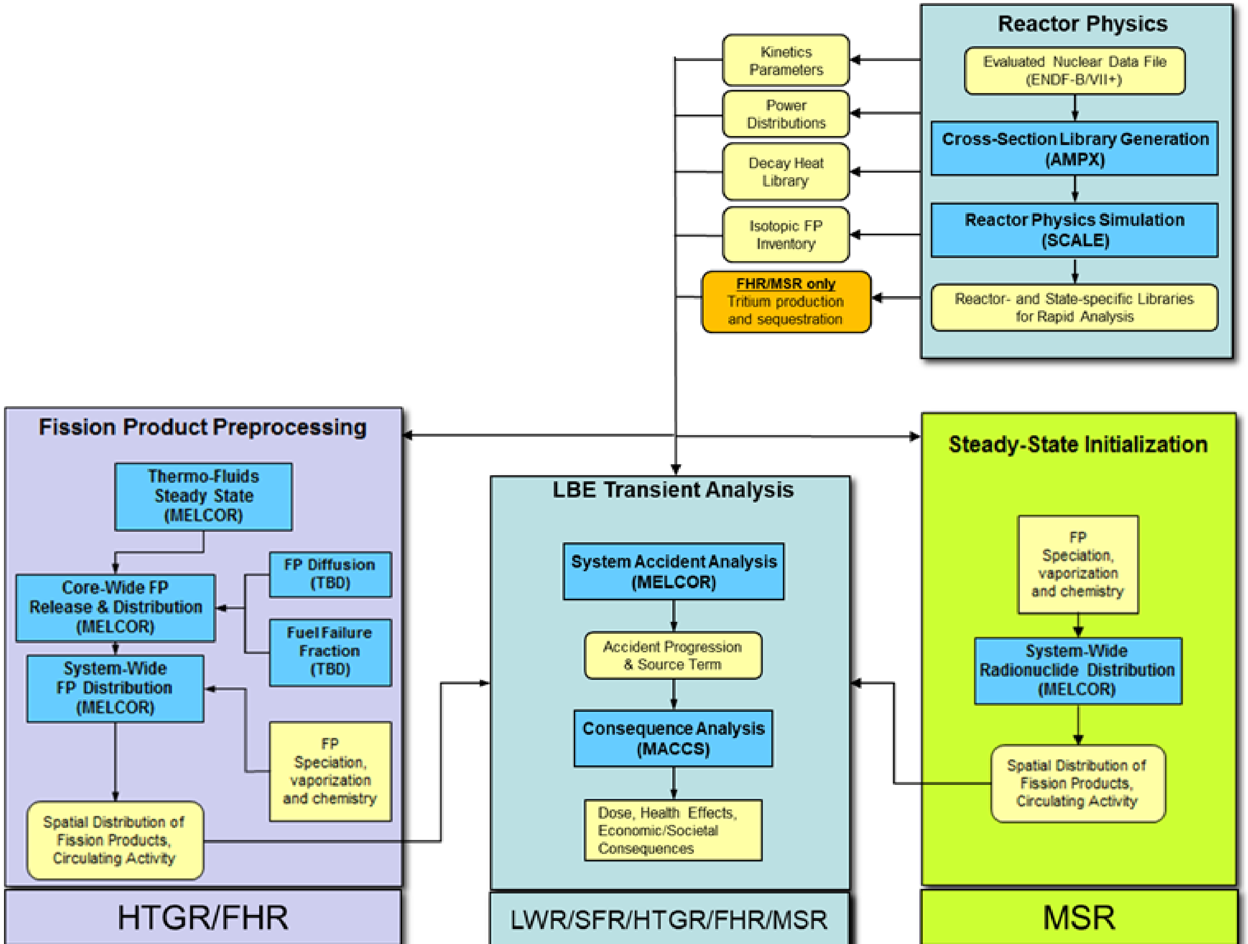
Color Legend: Legacy codes US/DOE Codes X-energy in house code Commercial NQA-1 Code

 Nuclear Energy. Reimagined. 11

X-energy plan for source term characterization

Step 12: Perform LBEs Source Term Modeling and Simulation: Source term evaluation model for non-LWRs

- Technology-inclusive because it relies on the same codes with the suite of physics models needed for the different non-LWR technologies.



Step 13. Review LBEs List for Adequacy of Regulatory Acceptance

- Develop a final list of LBEs.
  - Review current
    - PRA
    - Safety classifications of SSCs
  - Are any end results changes desired before the final list?
    - Changes to increase F-C target criteria margin
    - Reduction of uncertainties in LBE frequencies or consequences
    - Limit restrictions on siting or emergency planning
    - etc…
- If the final list is not complete, go back to Step 6.

Step 14. Document Completion of Source Term Development

- Prepare a documentation covering methods used, source term calculations and results and submit to the NRC for approval.

BACK UP SLIDES

Step 12: Perform LBEs Source Term Modeling and Simulation: Source term assessment software requirements

- Reactor Physics Computer Models:
  - Calculate radionuclide inventories and power distributions in the design.
- Fuel Performance Computer Models:
  - Calculate thermal and stress histories for fuel and identify fuel failure and radionuclide release
- System Analysis Computer Models:
  - Calculate the progression of accident and radionuclide transport.
  - Requires boundary conditions from fuel performance analysis.
- Radionuclide Transport Models (linked to system analysis models):
  - Calculate radionuclide release and transport within the reactor and surrounding structures.
  - Calculate radionuclide transport from the reactor to the EAB and transport in the atmosphere (plume dispersion).
- Dosimetry Computer Models (linked to radionuclide transport models):
  - Calculate doses within and outside the site boundaries during normal operation and accident conditions. Used to determine whether the plant design meets offsite dose limits and criteria and risk goals.
- Uncertainty Assessment Computer Models:
  - Categorize the uncertainties associated with the events' source terms and select the most impactful ones to be considered.

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Protecting People and the Environment

Advanced-Reactor Source Term – Pilot Studies

Advanced Reactor Stakeholder Meeting

August 20, 2020

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

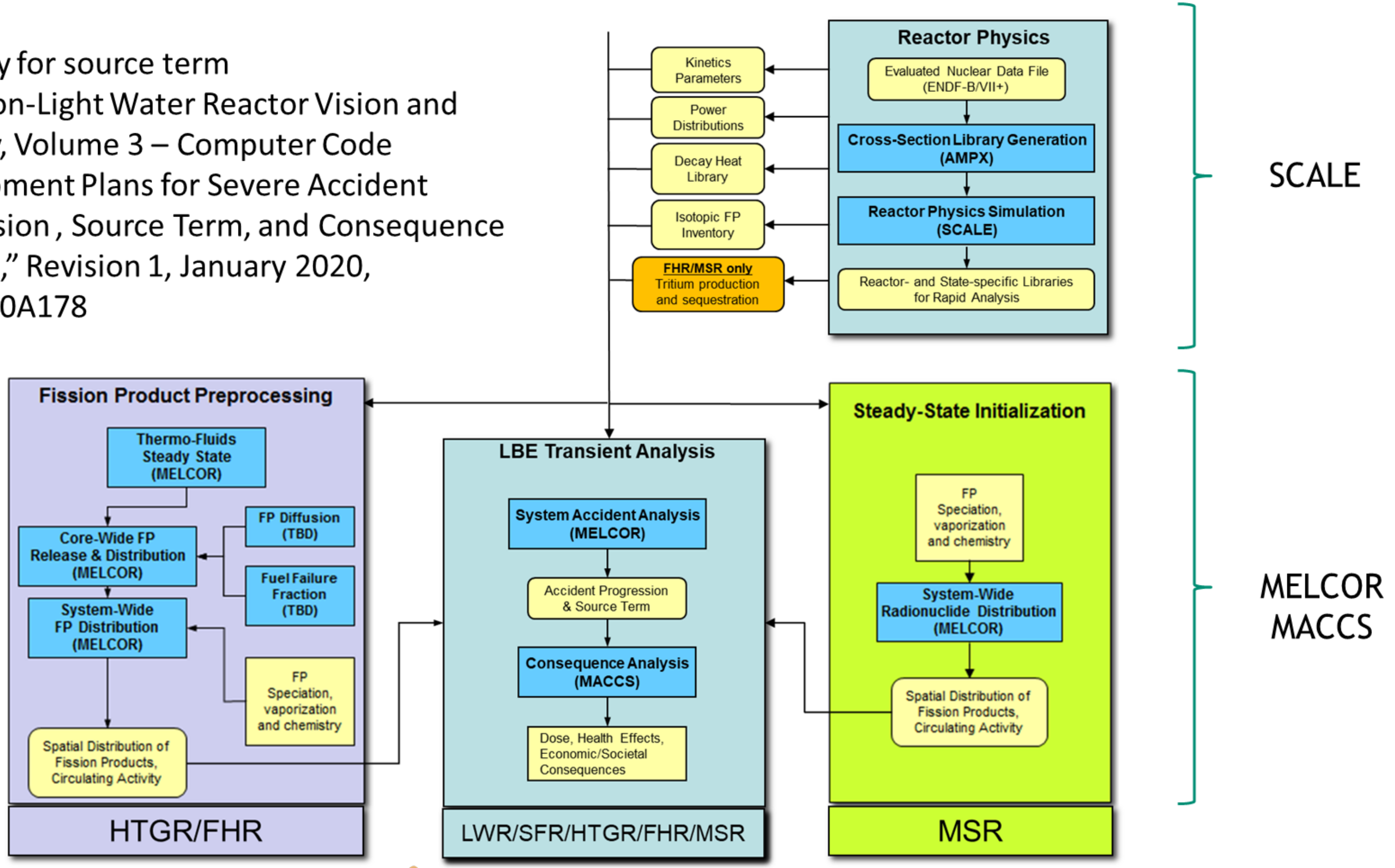
OAK RIDGE
National Laboratory

Non-LWR Evaluation Model

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



Project Objectives

Develop an understanding of non-LWR beyond-design-basis-accident behavior

- Provide insights for regulatory guidance
- Facilitate dialogue on the staff's approach to assessing source term

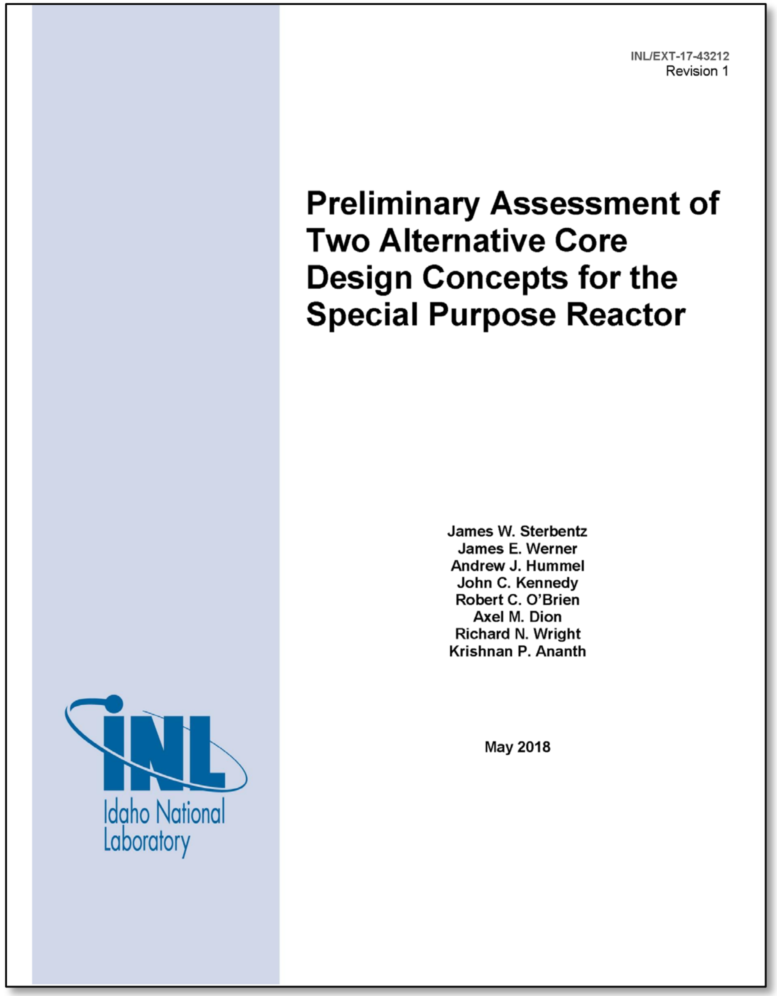
Demonstrate application of MELCOR and SCALE

- Develop publicly available input models - available upon request
- Code distribution handled separately

Project Stages

1. Select design
2. Develop input deck
3. Select scenarios
4. **Perform calculations and refine input deck**
   - Full-plant decks have been developed for heat pipe and gas-cooled reactors
   - Salt-cooled reactor input deck in preparation
   - Results shown here are preliminary to illustrate approach
5. Public workshop

INL Design A Heat Pipe Reactor

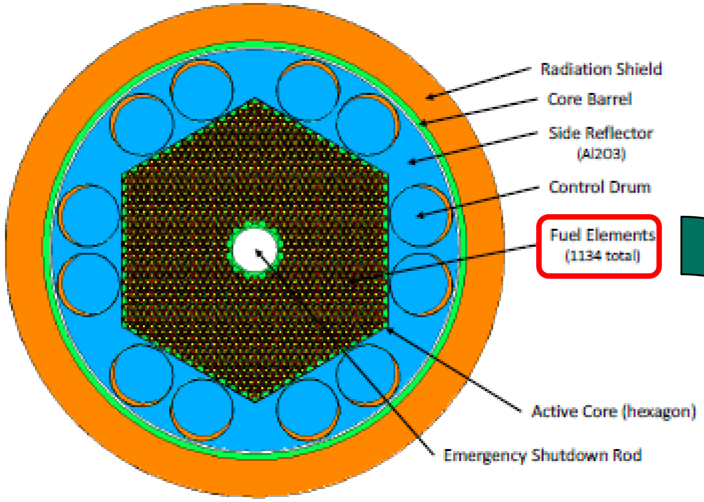

INL/EXT-17-43212
Revision 1

Preliminary Assessment of Two Alternative Core Design Concepts for the Special Purpose Reactor

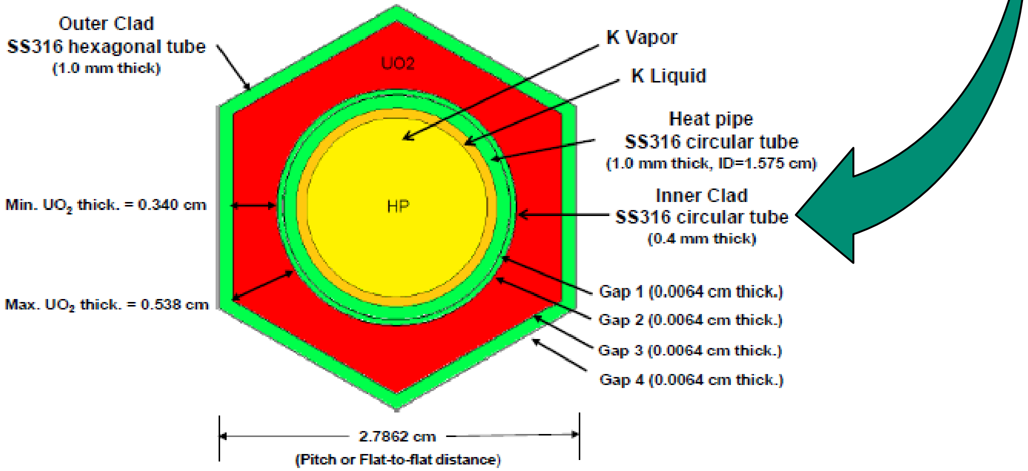
James W. Sterbentz
James E. Werner
Andrew J. Hummel
John C. Kennedy
Robert C. O'Brien
Axel M. Dion
Richard N. Wright
Krishnan P. Ananth

May 2018

Idaho National Laboratory



Cross-sectional view of the Design A core layout.



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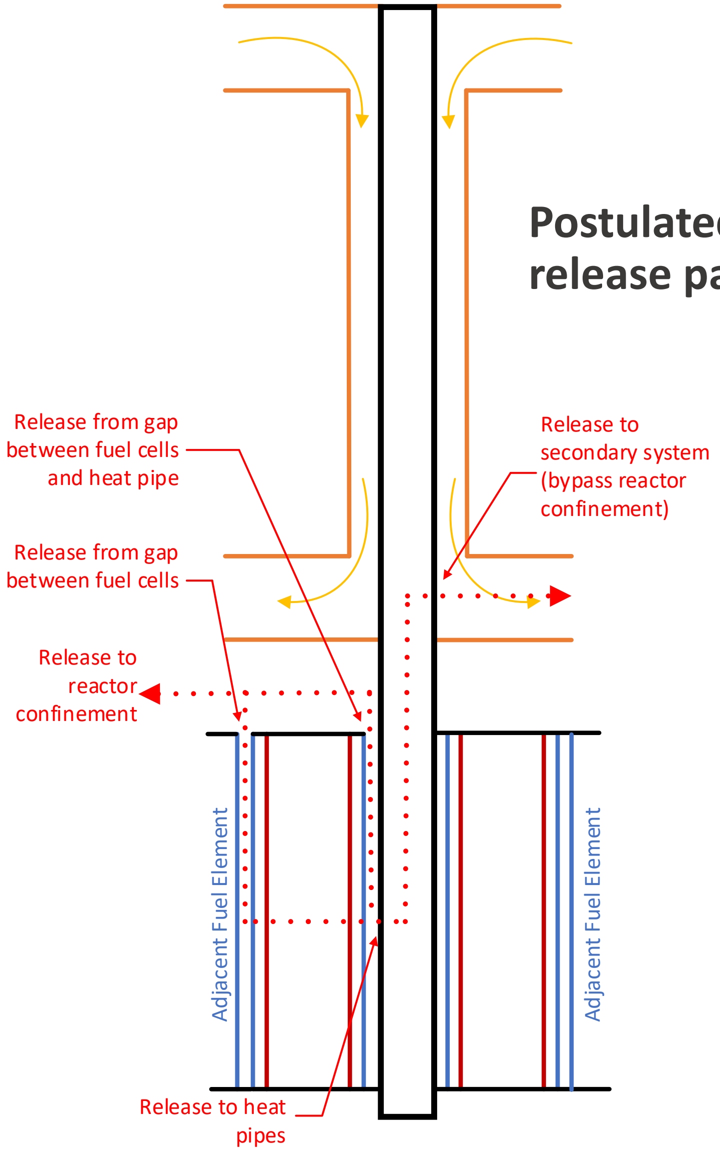
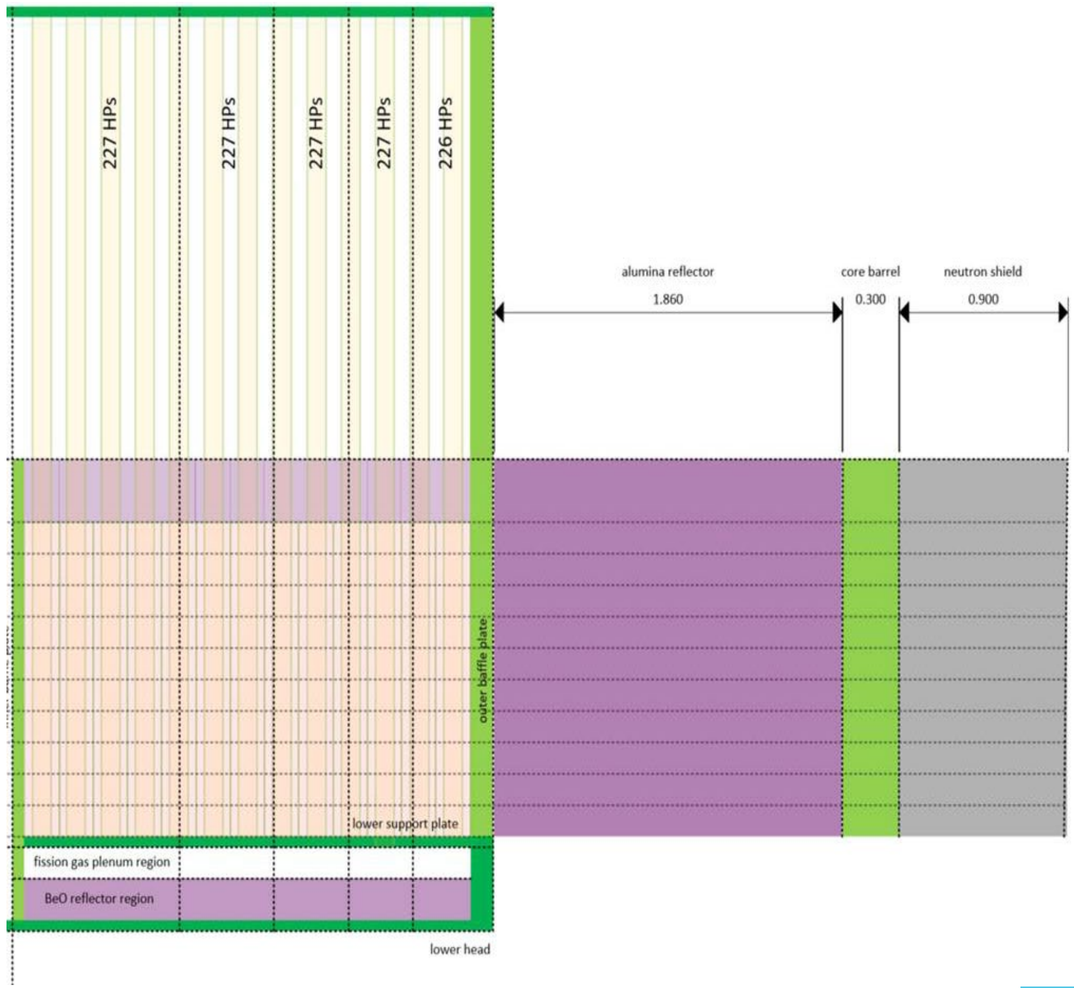
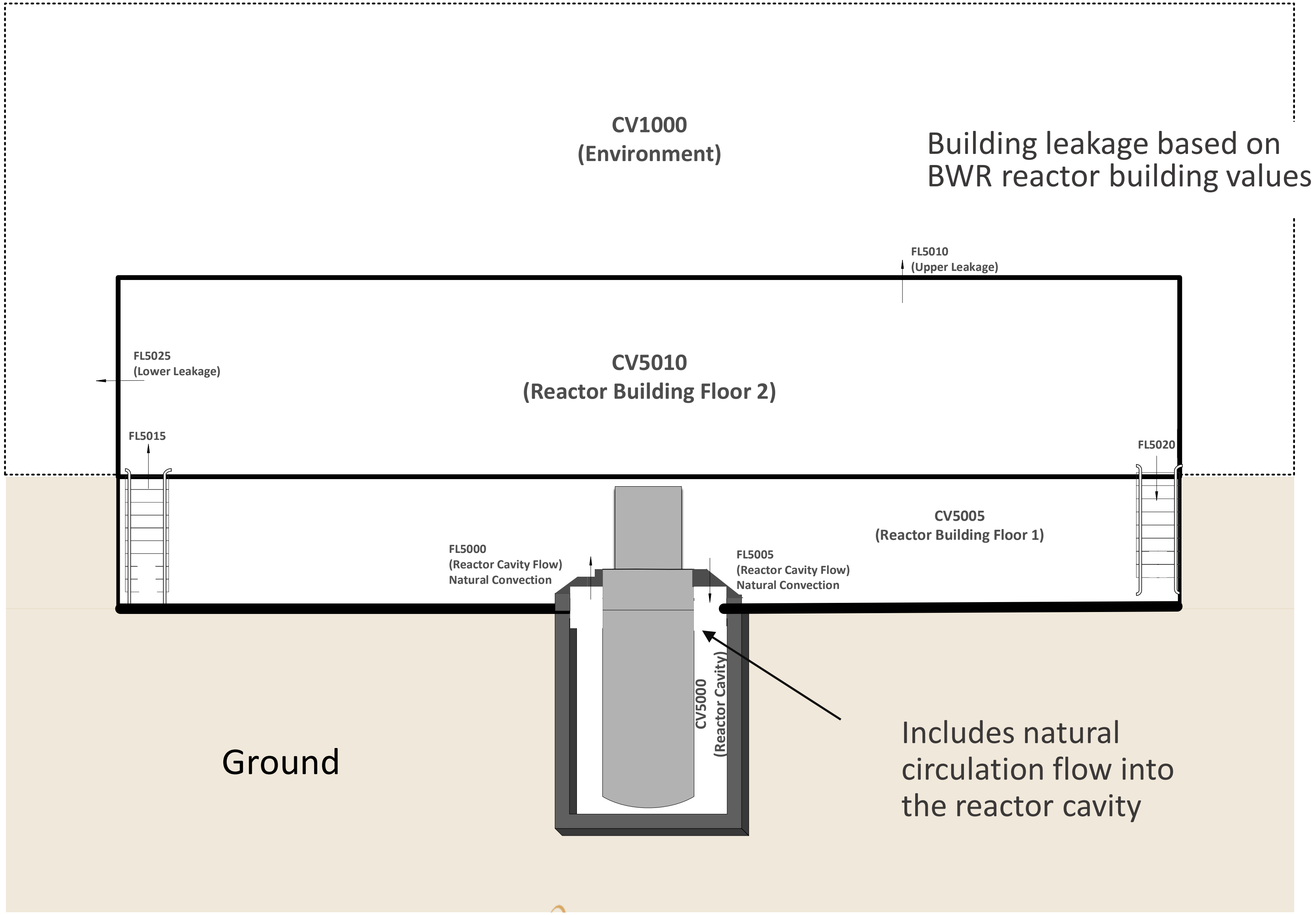
INL Design A – Reactor vessel and core nodalization



Postulated fission product release pathways

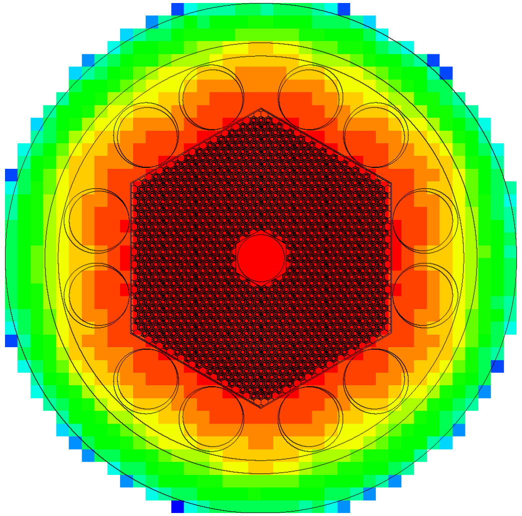


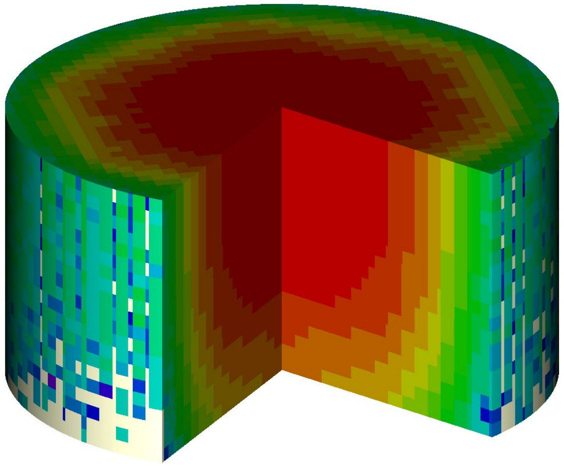
INL Design A – Reactor building nodalization

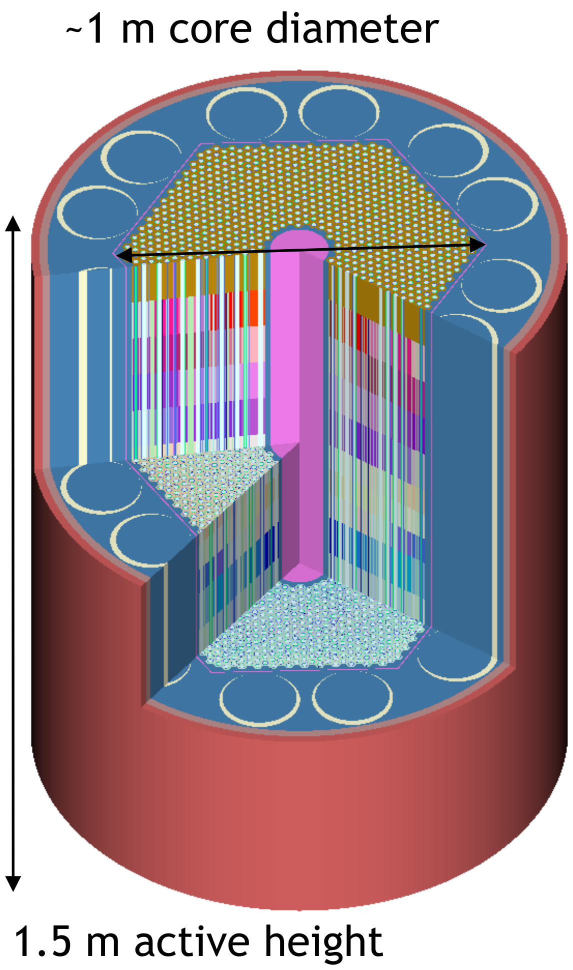


INL Design A SCALE model

- Design features
  - 1134 annular hexagonal UO_2 fuel elements (19.75% ^{235}U)
  - Fast neutron spectrum
- Modeling strategy
  - Flux was evaluated assuming a fixed control drum configuration
  - Isotopic inventory evaluated at full power over core life
- Radionuclide inventory and decay heat data provided for MELCOR model



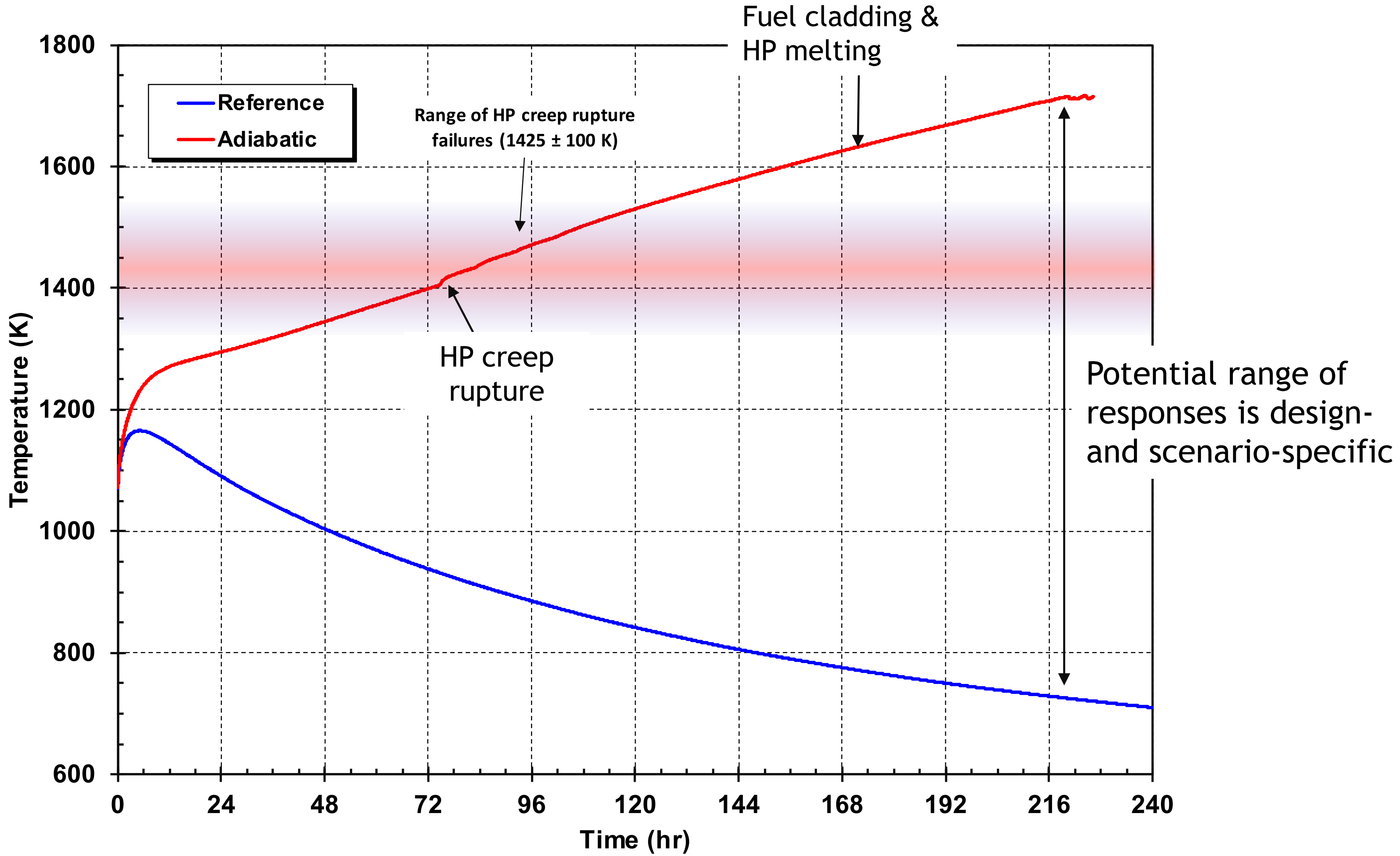




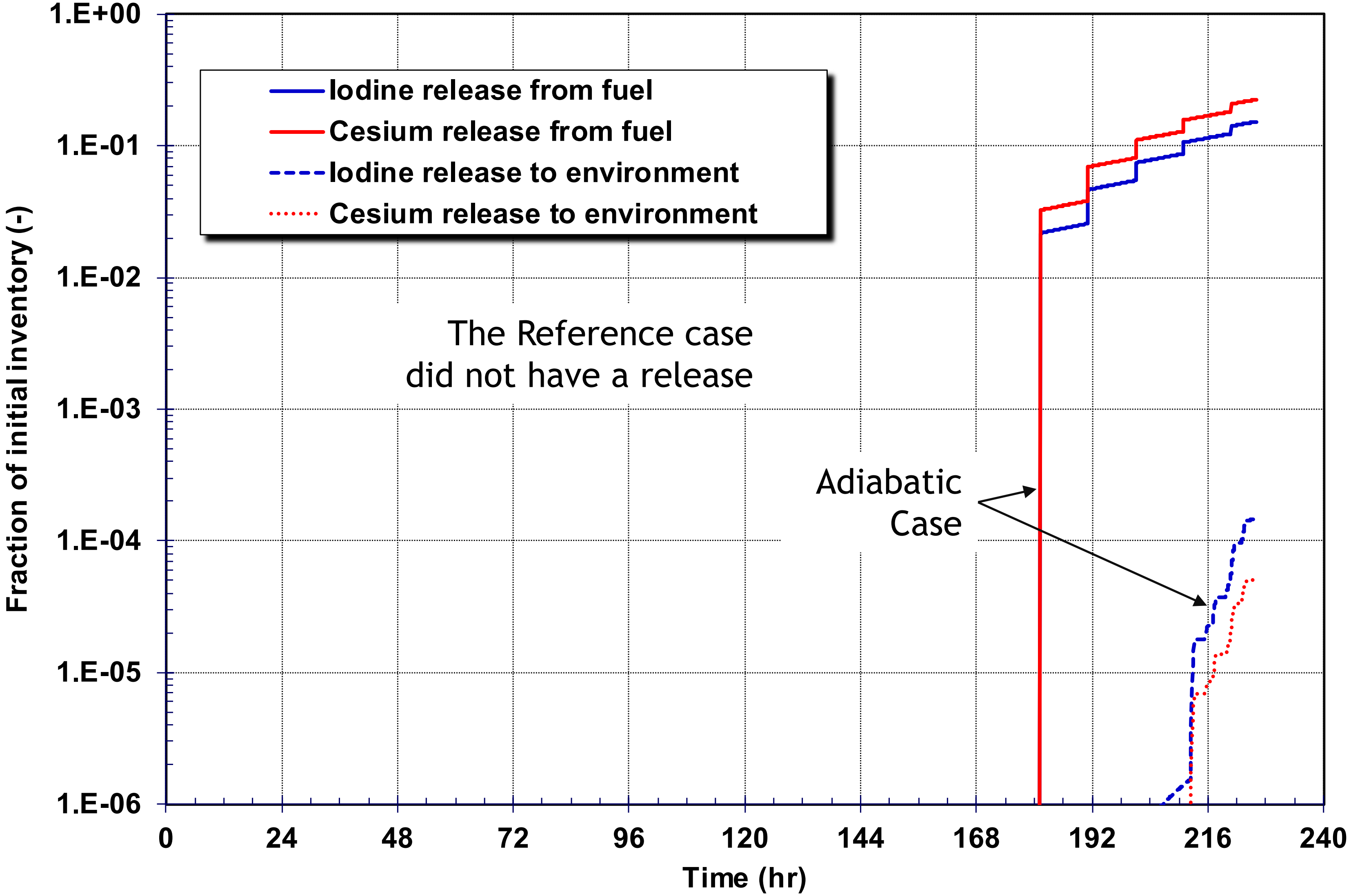
INL Design A – Demo calculations

- Reference case
  - Initiator trips secondary heat removal
  - Control rod insertion
  - Thermal radiation from the reactor vessel
  - Natural circulation flow through the reactor cavity
- Adiabatic case
  - No convective or radiative heat transfer from the vessel

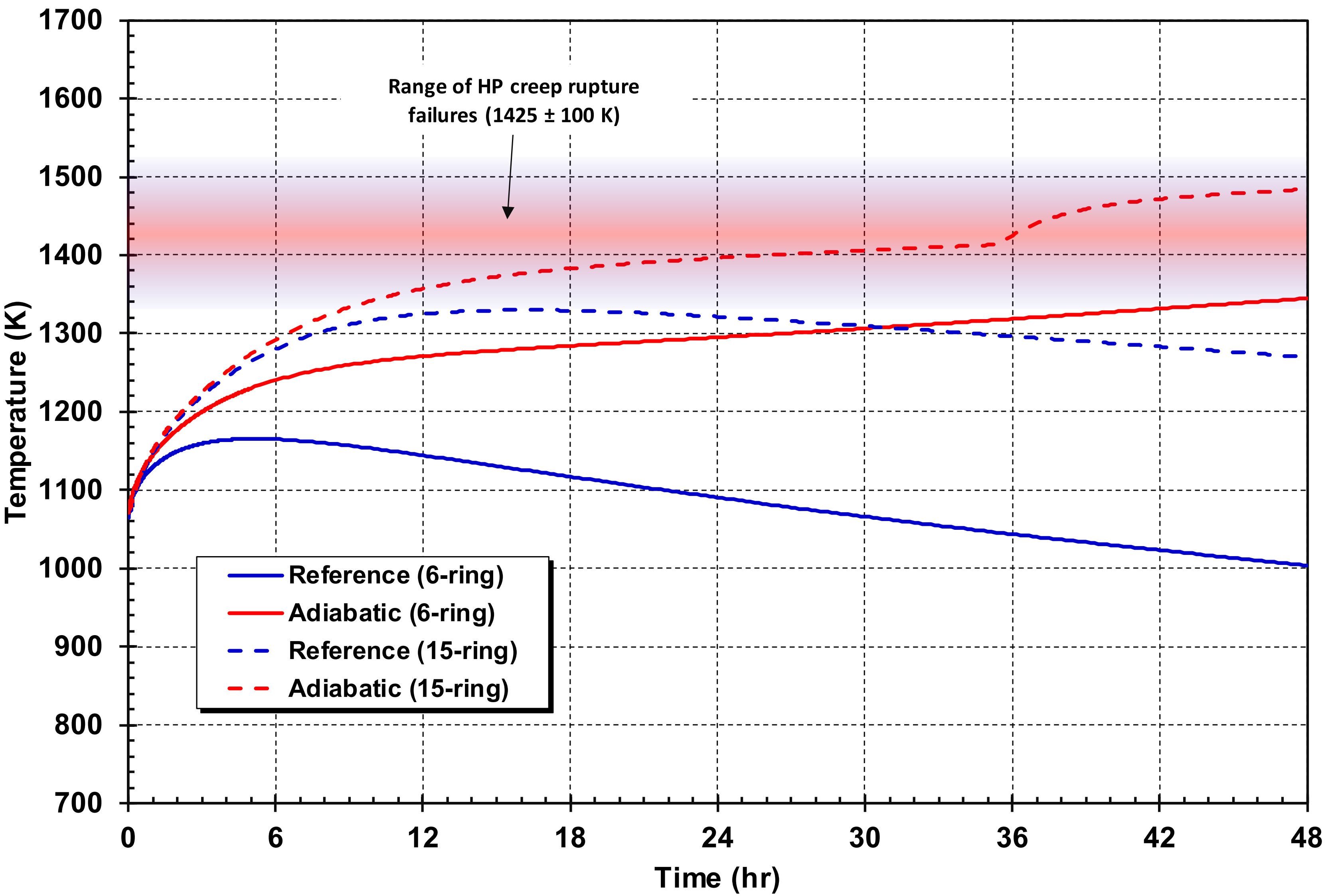
INL Design A – Peak fuel temperatures



INL Design A – Iodine and cesium release



INL Design A – Peak fuel temperatures radial nodalization sensitivity

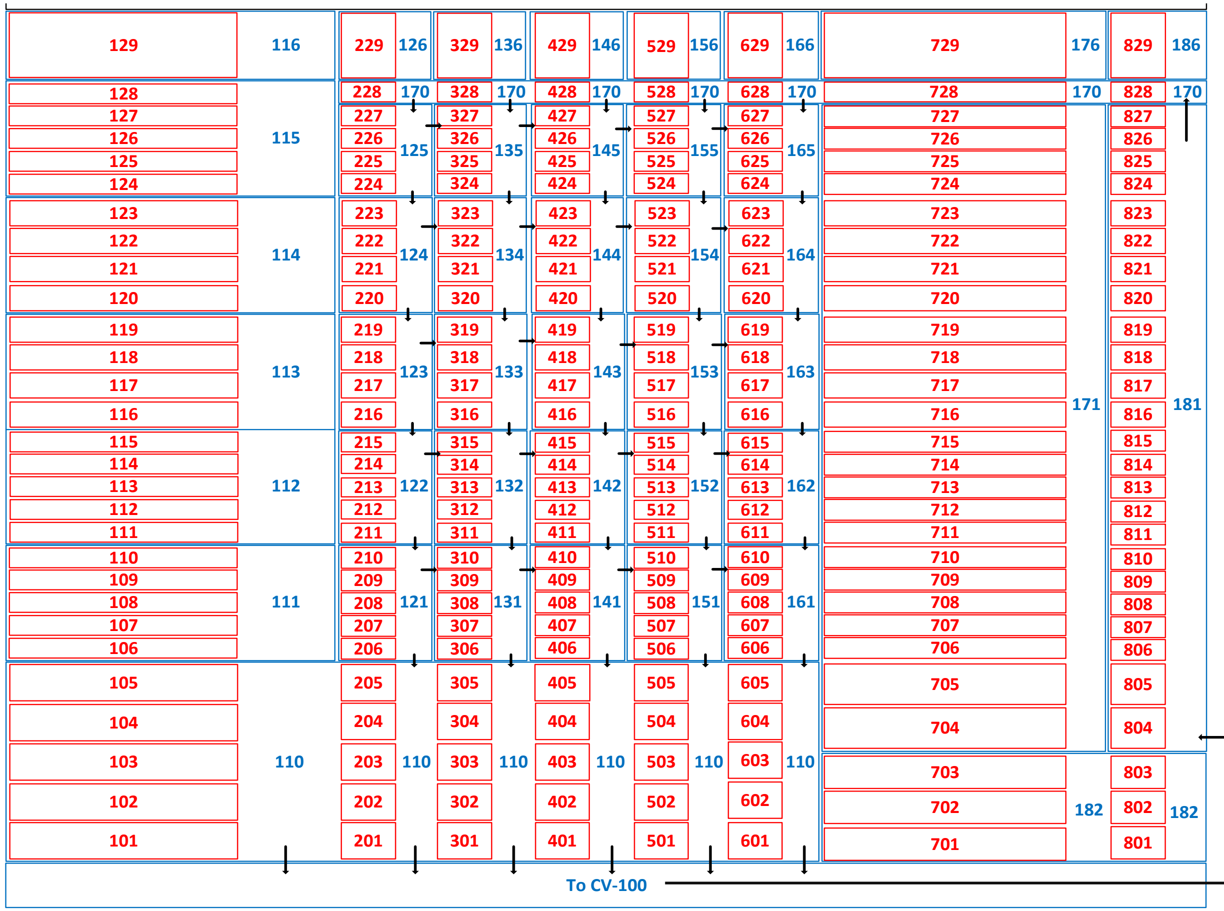


PBMR-400 reactor and core

Inlet plenum

Pebble bed

Outlet plenum



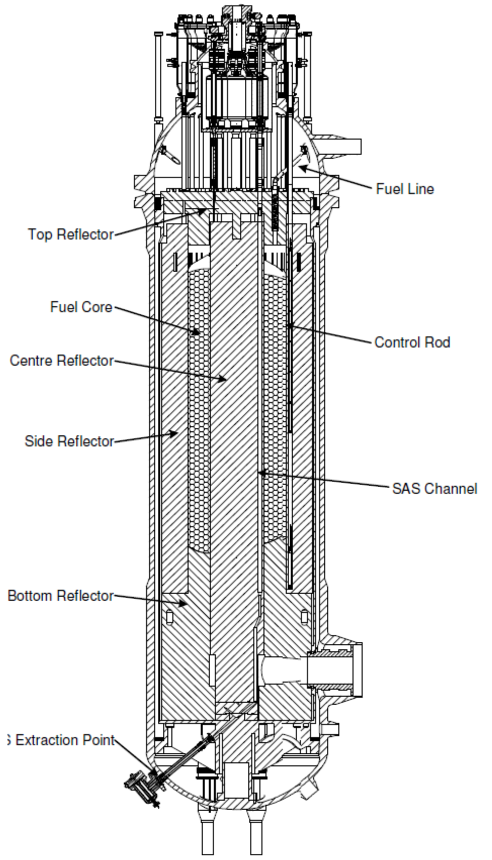
Inlet flow

Outlet flow

Ring 1 is the graphite inner reflector

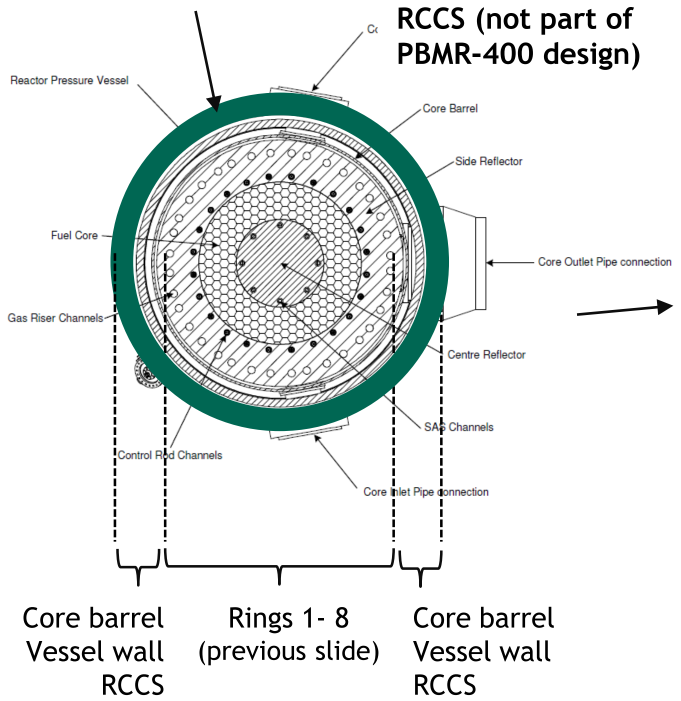
Pebble bed is in Rings 2-6

Ring 7 is the outer graphite reflector
Ring 8 is inlet flow risers & reflector

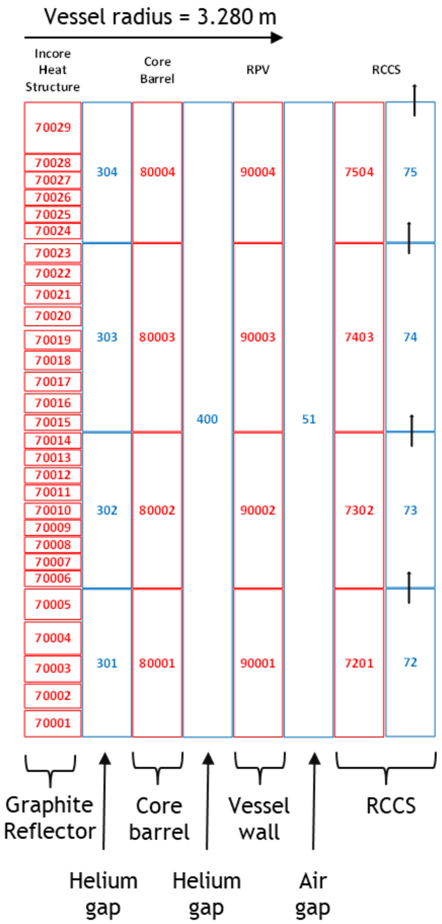


[P.J. Venter, M.N. Mitchell, F. Fortier, PBMR reactor design and development, in: Proceedings from the 18th International Conference on Structural Mechanics in Reactor Technology (SMiRT 18), Beijing, China, Aug. 2005]

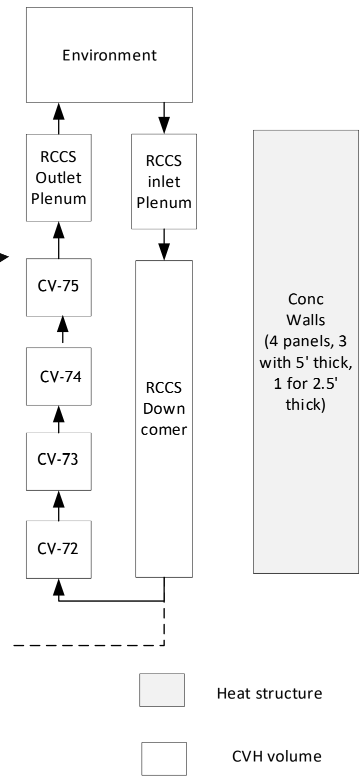
PBMR-400 vessel and reactor building



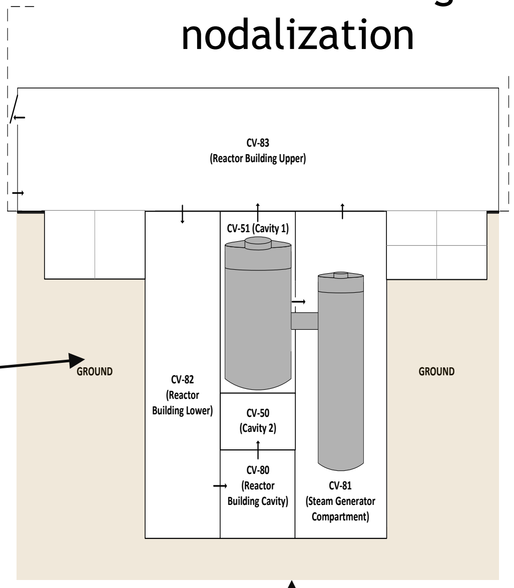
[Ducknor, Nuclear Engineering and Technology, 49, 360-372, 2017]

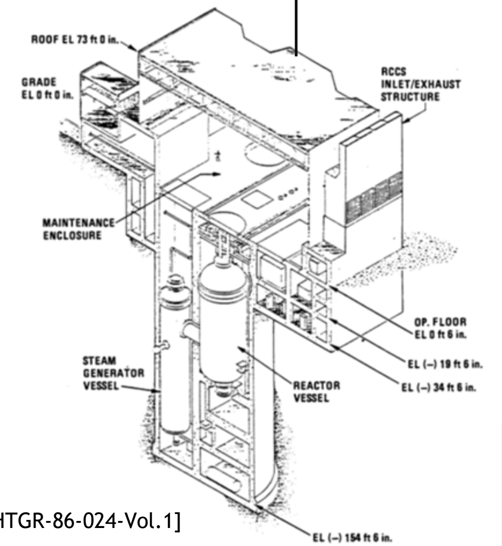


RCCS adapted from the from MHTGR



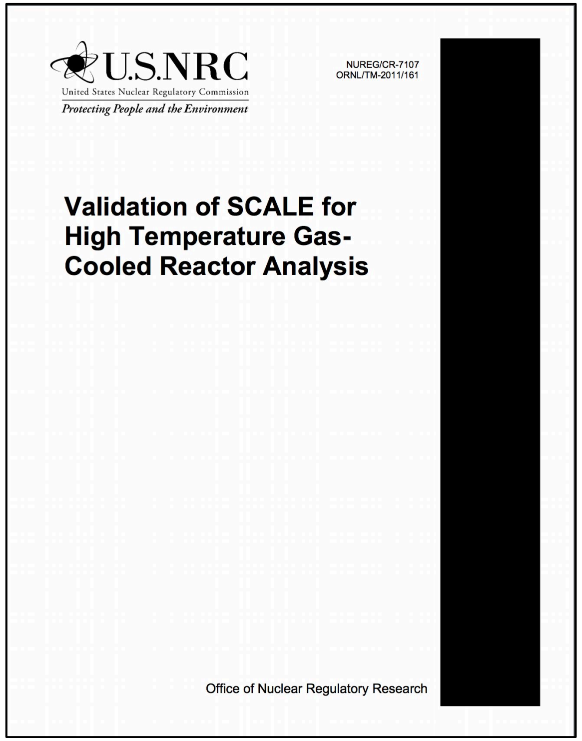
Reactor building nodalization





[DOE-HTGR-86-024-Vol.1]

PBMR-400 SCALE model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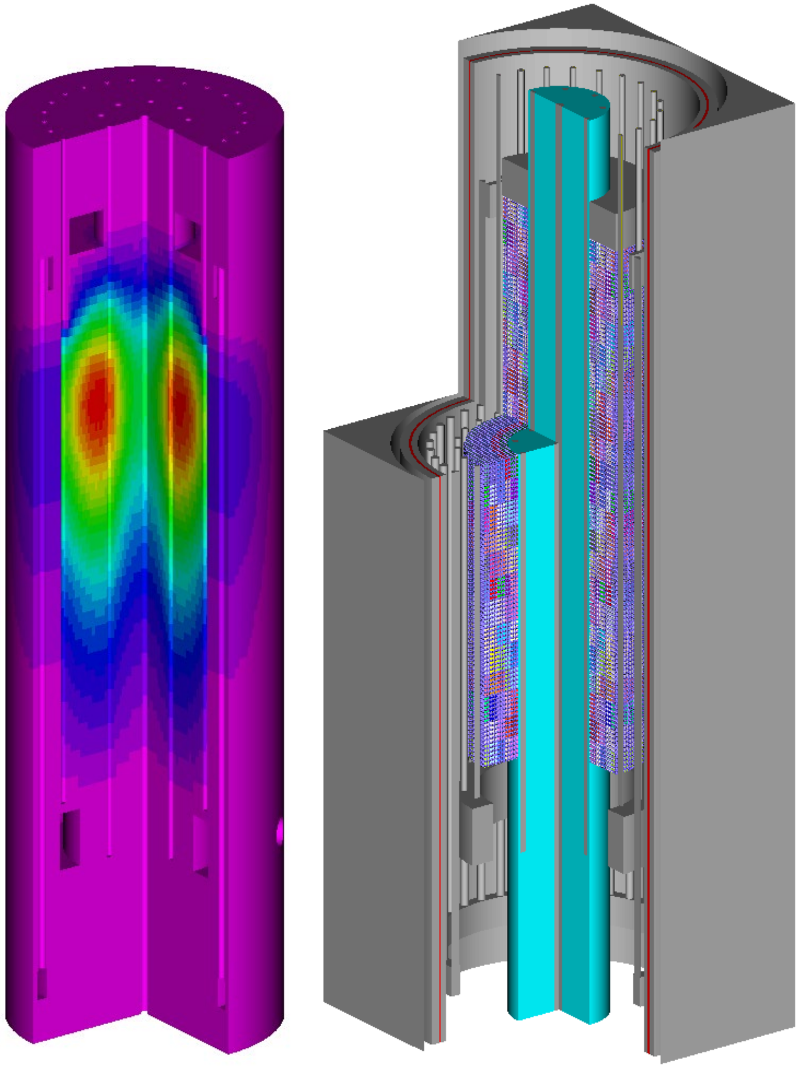
United States Nuclear Regulatory Commission
Protecting People and the Environment

NUREG/CR-7107
ORNL/TM-2011/161

Validation of SCALE for High Temperature Gas-Cooled Reactor Analysis

Office of Nuclear Regulatory Research

- Design features
  - Fueled by graphite pebbles containing UO_2-bearing TRISO fuel particles
  - Pebbles circulate multiple passes through the core to achieve a high burnup
- Modeling strategy
  - Analysis focused on understanding axial & radial power shape and the neutron spectrum for depletion calculations
  - Facilitate depletion calculations via pre-calculated Origen reactor data libraries
- Radionuclide inventory and decay heat data provided for MELCOR model



PBMR-400 SCALE geometry & neutron flux profile

PBMR-400 – Demo Calculations

Depressurized loss-of-forced circulation (DLOFC) accident

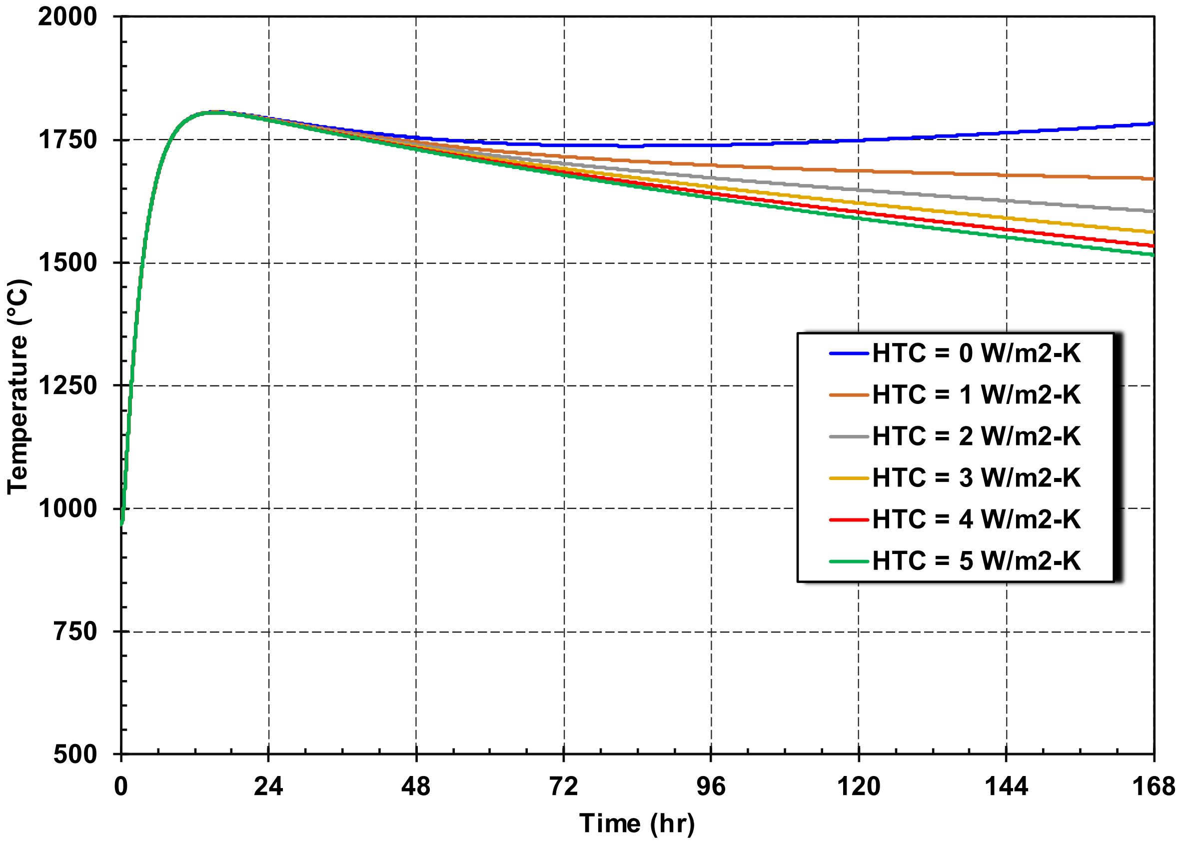
- Large recirculation pipe break
  - Reactor trip
  - Secondary system trips & isolates
  - Passive reactor cavity cooling system (RCCS) available
- Reference case includes nominal heat transfer to the RCCS
- Vessel to RCCS heat transfer sensitivity
  - Heat transfer coefficient to air in the RCCS varied from 0 to 5 $W/m^2 K$
- RCCS blockage sensitivity
  - Natural circulation air flow area into the RCCS decreased by 90%, 99%, and full blockage

TRISO fission product release model

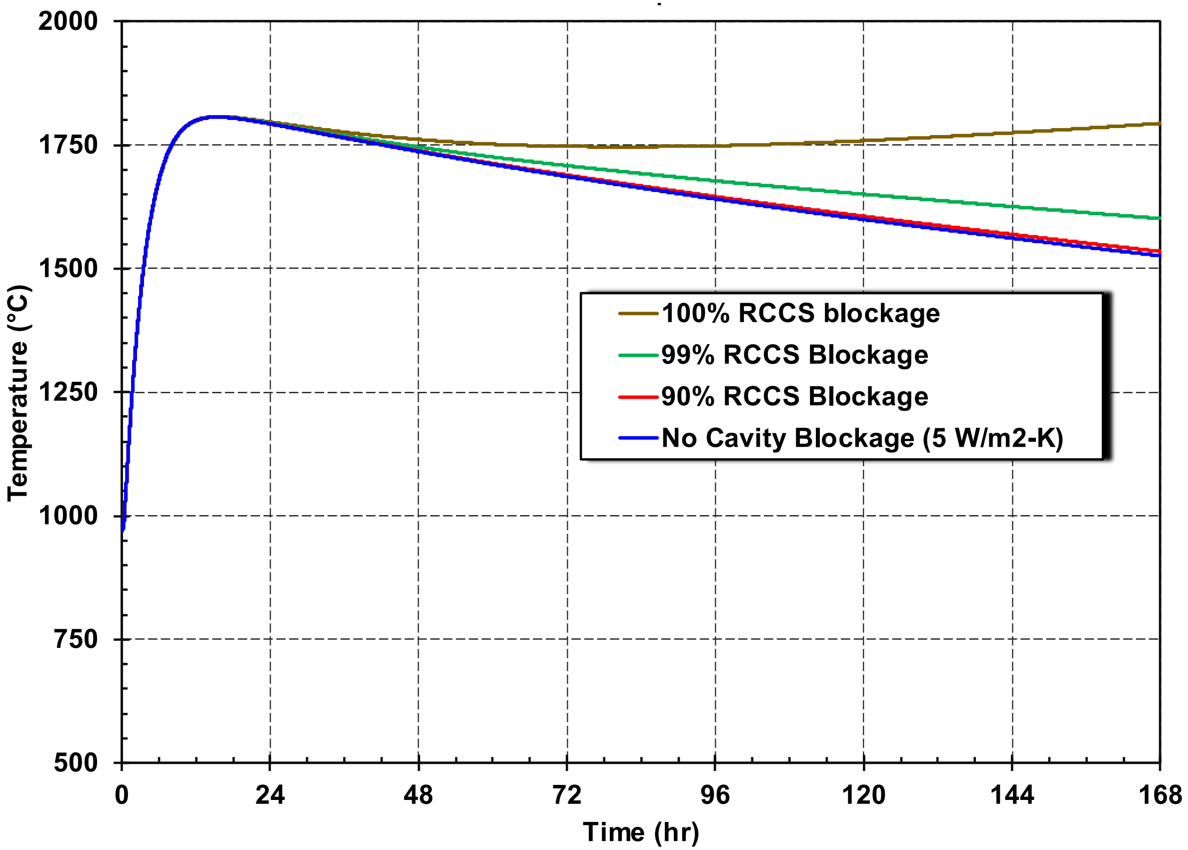
- Diffusivity data from IAEA TECDOC-978, Appendix A
- Fuel failure fraction is user-specified – temperature dependent curve

PBMR-400 – DLOFC results

Peak fuel temperature sensitivity to the RCCS heat transfer coefficient

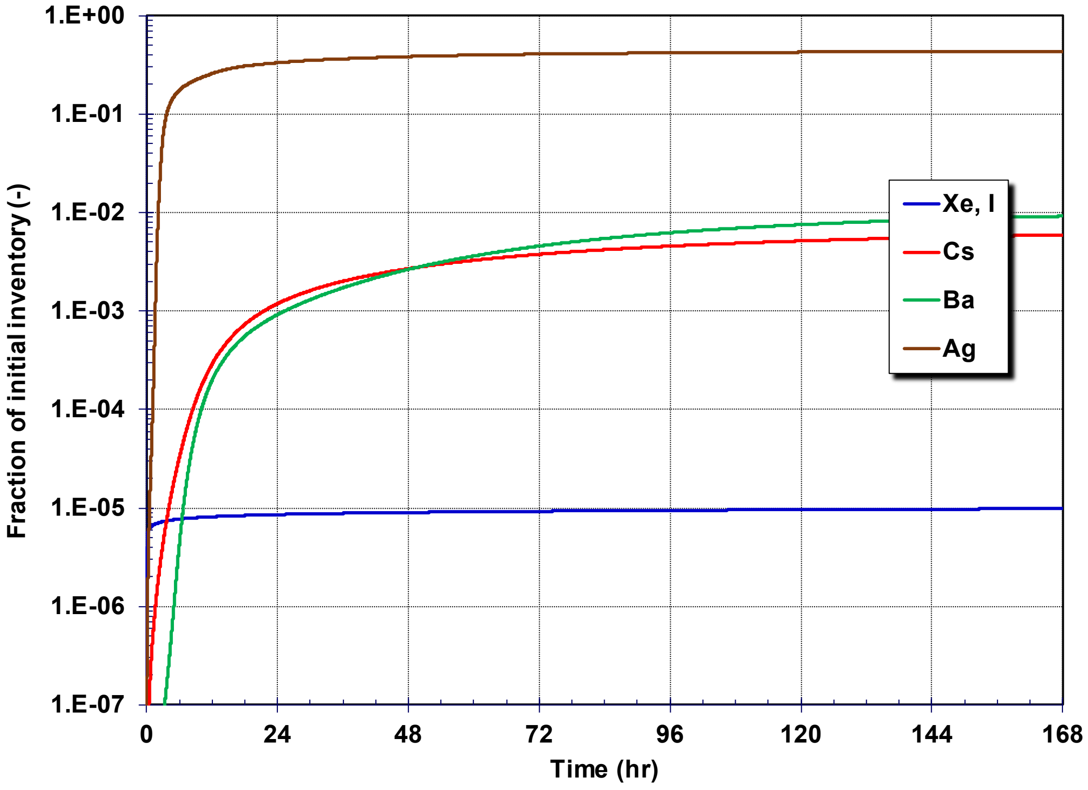


Peak fuel temperature sensitivity to RCCS blockage

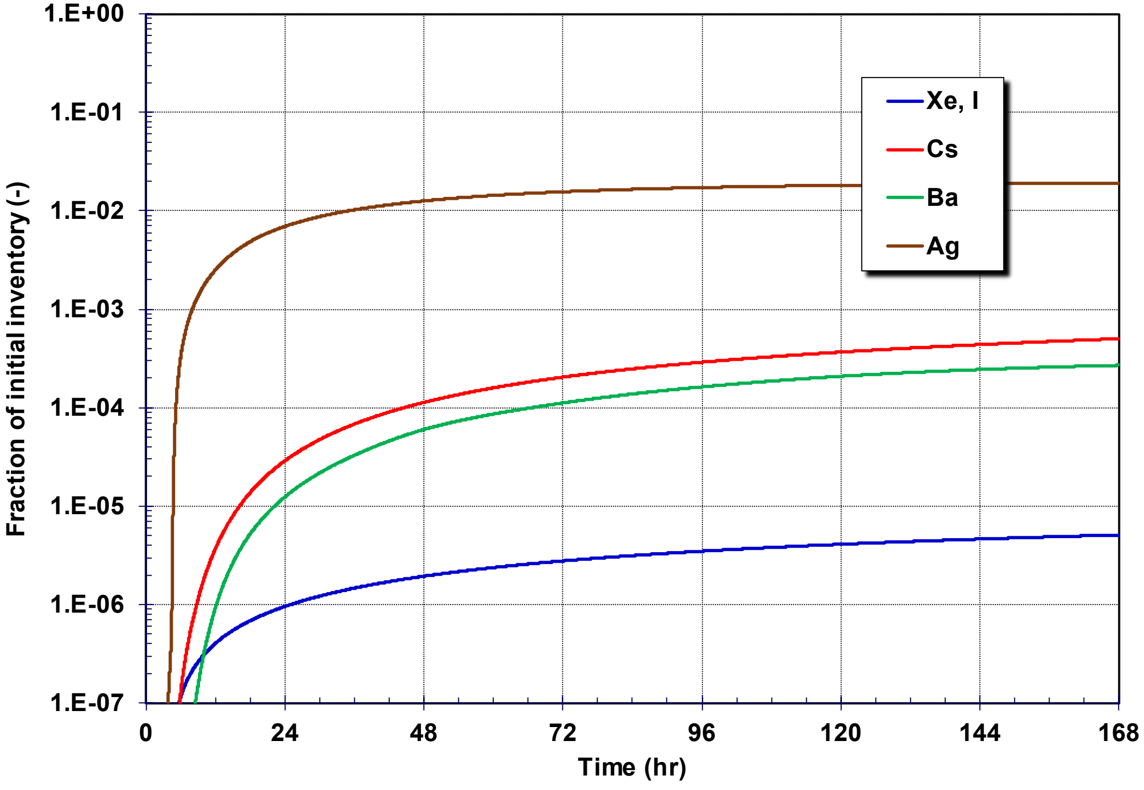


PBMR-400 – DLOFC reference case results

Release from the pebbles to the coolant



Release to the environment



Fluoride-Salt-Cooled High-Temperature Reactor (FHR)

FHR Fluoride-Salt-Cooled, High-Temperature Reactor

Mark-1 PB-FHR Technical Description

Technical Description of the "Mark 1" Pebble-Bed Fluoride-Salt-Cooled High-Temperature Reactor (PB-FHR) Power Plant

Charalampos "Harry" Andreades
Anselmo T. Cisneros
Jae Keun Choi
Alexandre Y.K. Chong
Massimiliano Fratoni
Sea Hong
Lakshana R. Huddar
Kathryn D. Huff
David L. Krumwiede
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Nicolas Zweibaum
Ehud Greenspan
Per F. Peterson

UCBTH-14-002

September 30, 2014
Department of Nuclear Engineering
University of California, Berkeley

This research is being performed using funding received from the U.S. Department of Energy Office of Nuclear Energy's Nuclear Energy University Programs.

NEUP Nuclear Energy University Programs
U.S. Department of Energy

Technical Description of the Mark-1 PB-FHR Power Plant 1 | 153

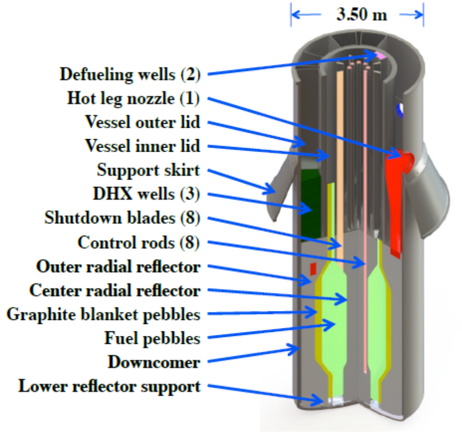


Figure 1-3. The Mk1 PB-FHR reactor vessel

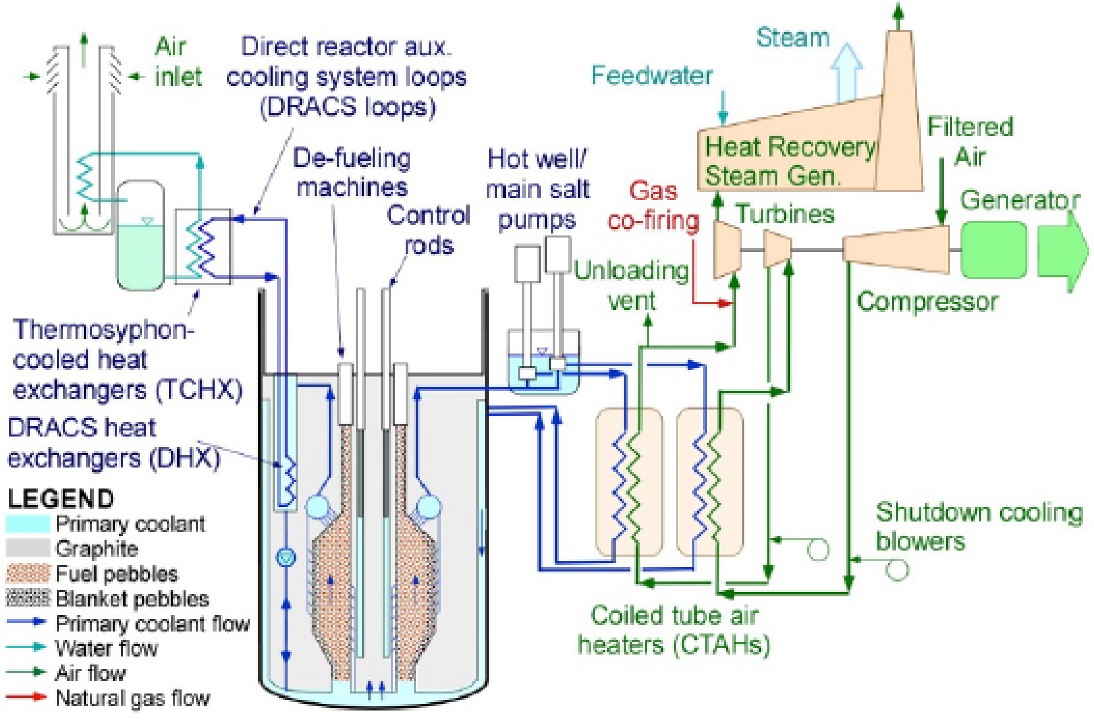
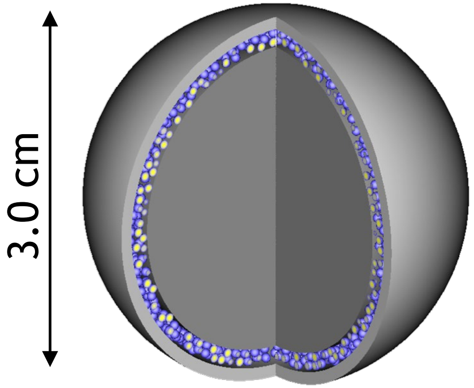


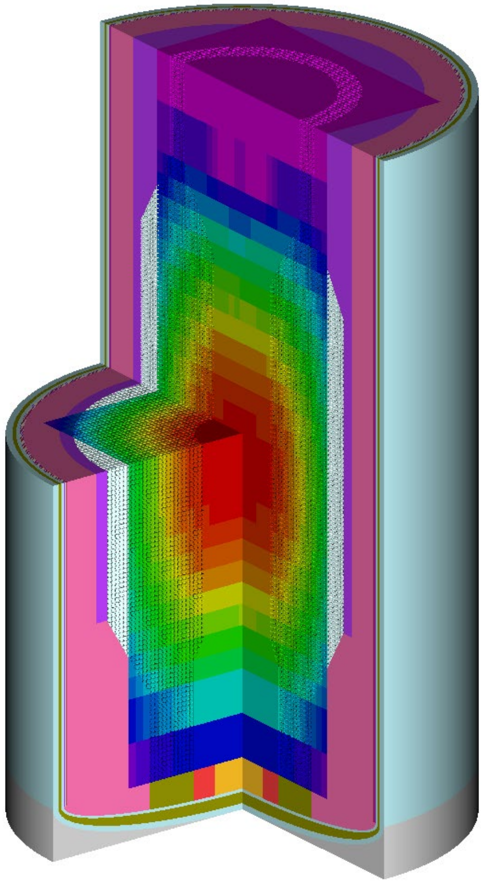
Figure 1-1. Mk1 PB-FHR flow schematic.

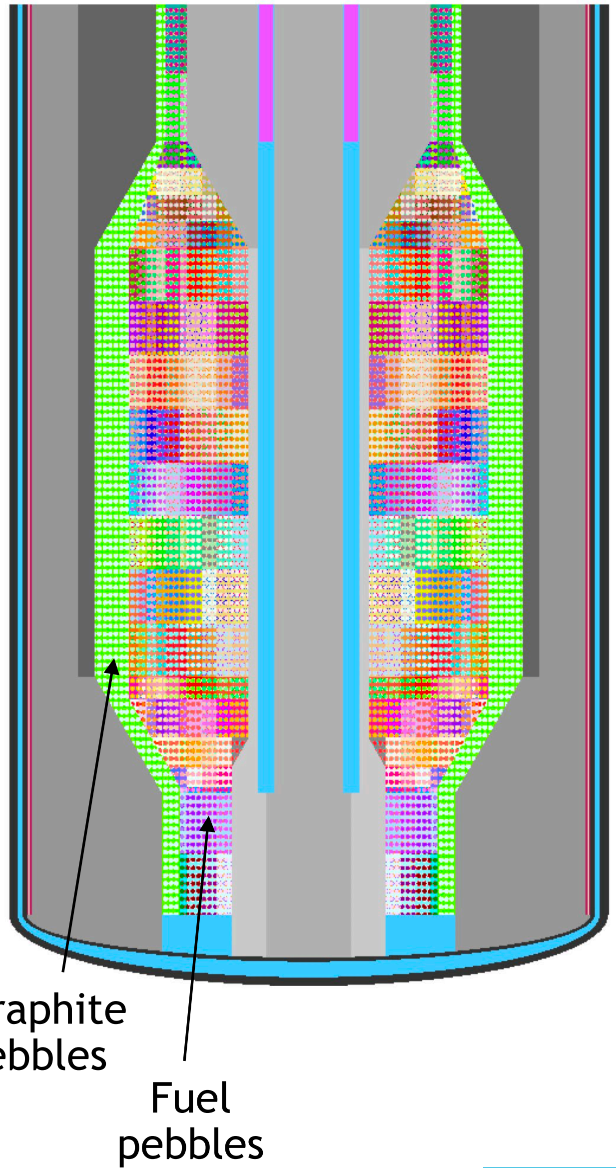
FHR SCALE model

- Design features
  - TRISO particles with UCO fuel kernel (19.9% ^{235}U) in graphite pebbles
  - 236 MW_{th} core with approx. 470,000 fuel pebbles & 218,000 graphite pebbles
  - FLiBe salt coolant
- Modeling strategy
  - Fixed pebble positions (no buoyancy effects)
- Radionuclide inventory and decay heat data provided for MELCOR model



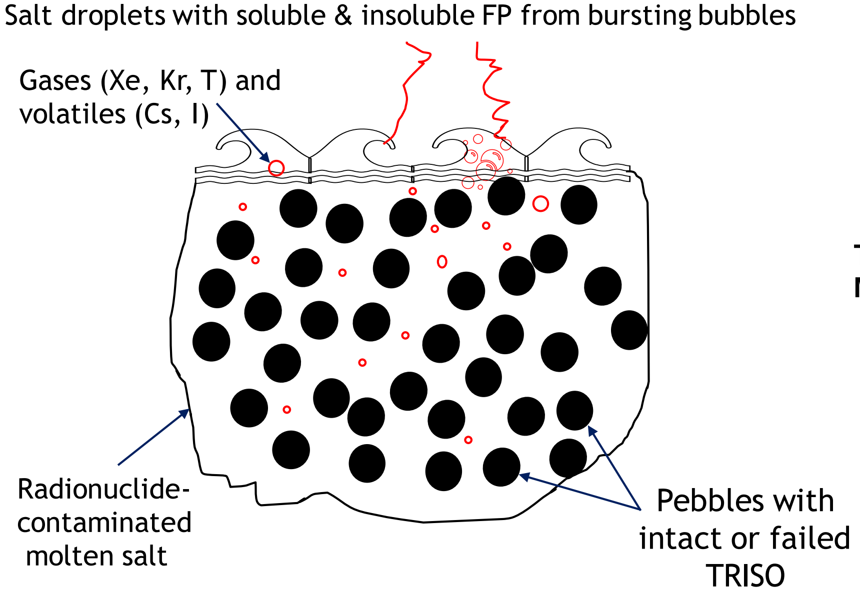
Fuel pebble



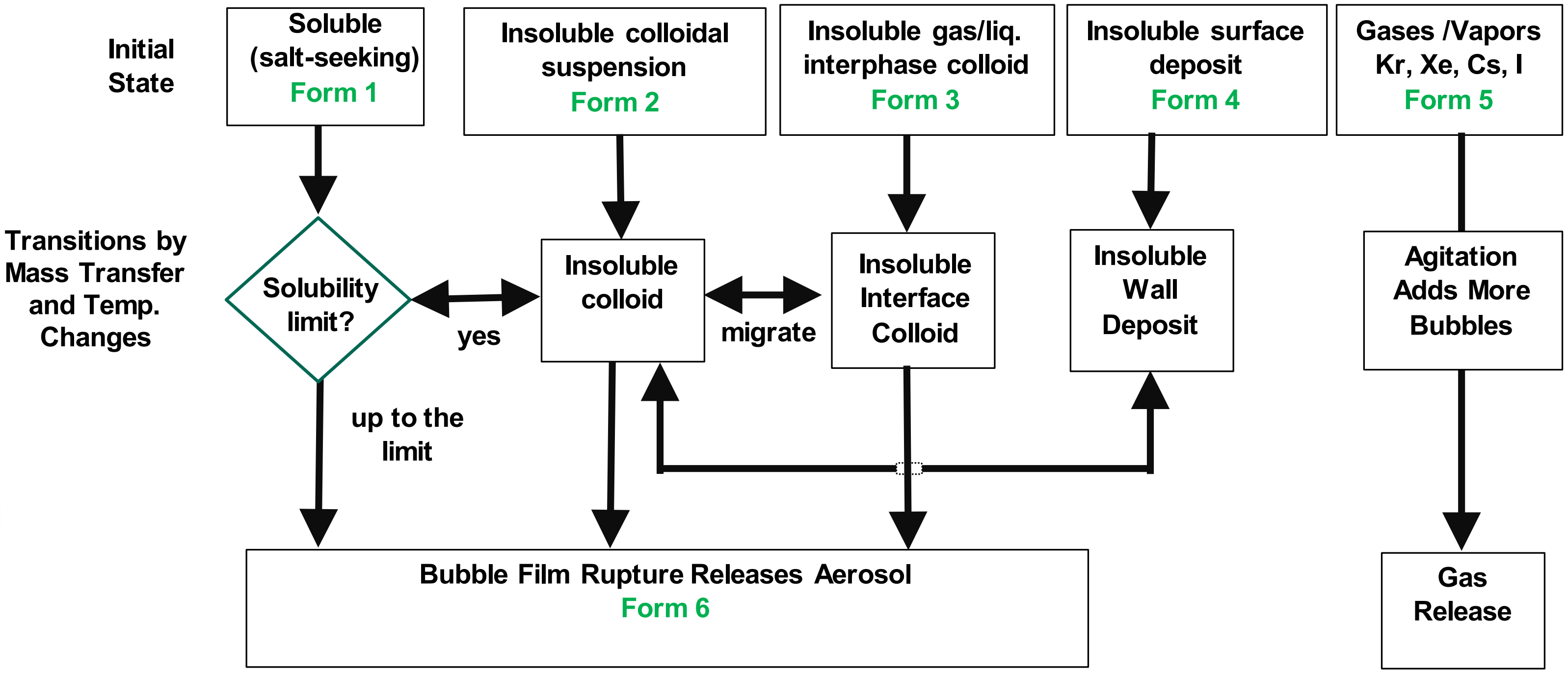


MELCOR fission product release model

Vaporization and bubble burst release (see Vol. 3)



Radionuclides grouped into 6 forms as found in the Molten Salt Reactor Experiments at ORNL



Concluding remarks and next steps

Preliminary working input models

- INL Design A - November 2020
- PBMR-400 - November 2020
- FHR model - March 2021

Followed by public workshops

New computer code versions will be released with updated phenomenological models

United States Nuclear Regulatory Commission
Protecting People and the Environment

Break

Meeting/Webinar will resume shortly

Telephone Bridgeline: (888) 810-4937

Passcode: 8854397#

Advanced Reactor Program

Annual Fee Regulations for Non-Light Water Reactors

August 20, 2020

NEI
NUCLEAR
ENERGY
INSTITUTE

Current Annual Fee Regulations

- Annual fees outlined in 10 CFR Part 171, governed by OBRA-90
  - Variable fee structure established for light-water SMRs in June 2016

- Currently, annual fees not technology-inclusive and apply only to LWRs
  - Timely consideration necessary given micro-reactor COL application docketed by NRC and more non-LWR developers in pre-application discussions with the NRC

Goals to Consider in Fee Rule Change

- Urgent need for annual fee regulations for non-LWRs; important for investment decisions
- Meet NEIMA requirements (FY 2021 and beyond)
  - Regulatory costs shared fairly and equitably among large and smaller-scale reactor facilities, as well as among various technologies
  - Reasonable relationship to cost of regulatory services
- Ensure continued protection of public health and safety

Preferred Annual Fee Rule Approach

Expand the SMR variable fee structure to include non-LWRs

- Basis for light-water SMR variable annual fee is equally applicable to non-LWRs
- Maximum, minimum and variable fees are appropriate for large & SMR non-LWRs

Address disproportionate impacts to micro-reactors

- Current minimum fee too high for micro-reactors; causes disproportionate impacts and overestimates regulatory costs
- Three options considered:
    1. Amend variable fee structure
    2. Fee cap to avoid disproportionate impact
    3. Separate fee structure for micro-reactors

Evaluation of Disproportionate Impact

Thermal Power Rating (MWt)	5	10	30	50	75	100
Current Annual Fee	$134,650	$134,650	$134,650	$134,650	$134,650	$134,650
Annual Plant Generating Cost	$554,800	$1,109,600	$3,328,800	$5,548,000	$8,322,000	$11,096,000
Annual Fee as Percent of Annual Plant Generating Cost	24.27%	12.14%	4.05%	2.43%	1.62%	1.21%

*All numbers are preliminary estimates; calculations use generating cost of $40/MWh[1] for micro-reactors, 95% capacity factor

[1]NEI Report, *Cost Competitiveness of Micro-Reactors for Remote Markets* (April 15, 2019).

Evaluation of Options to Address Disproportionate Impact

1. Amend variable fee structure

- Re-align minimum fee to micro-reactor range (100MWt)*
    - Use current variable fee rate to extend down; or
    - Set new minimum based on reduced regulatory costs

2. Fee cap to avoid disproportionate impact

- Create fee cap based on power level for those micro-reactors who would experience disproportionate impact (annual fee > 3% of annual generating cost)*
- Reactors with thermal power ratings less than 40.5 MWt pay $3,330/MWt; reactors 40.5MWt – 250MWt pay minimum fee*

*All numbers are preliminary estimates

Fee Cap to Avoid Disproportionate Impact

Thermal Power Rating (MWt)	5	10	30	40.5	75	100
New Annual Fee	$16,650	$33,300	$99,900	$134,650	$134,650	$134,650
Annual Plant Generating Cost	$554,800	$1,109,600	$3,328,800	$5,548,000	$8,322,000	$11,096,000
Percentage of Annual Cost Under SMR Structure	24.27%	12.14%	4.05%	2.43%	1.62%	1.21%
Percentage of Annual Cost Under Fee Cap	3.00%	3.00%	3.00%	2.43%	1.62%	1.21%

*All numbers are preliminary estimates; calculations use generating cost of $40/MWh[1] for micro-reactors, 95% capacity factor

[1]NEI Report, *Cost Competitiveness of Micro-Reactors for Remote Markets* (April 15, 2019).

Evaluation of Options to Address Disproportionate Impact

3. Separate fee structure for micro-reactors

- Similar to separate SMR fee structure, create separate micro-reactor fee structure within power reactor fee class
- Annual fee proportionate to ~1.2% of estimated annual generating cost, to remain fair and equitable to current fleet (Part 171 annual fees constitute an average of ~1.2% of annual generating costs for current fleet)*
- Micro-reactors (less than 100MWt) pay $1,360 /MWt*

*All numbers are preliminary estimates

Separate Fee Structure For Micro-Reactors

Thermal Power Rating (MWt)	5	10	30	50	75	100
New Annual Fee	$6,800	$13,600	$40,800	$68,000	$102,000	$134,650
Annual Plant Generating Cost	$554,800	$1,109,600	$3,328,800	$5,548,000	$8,322,000	$11,096,000
Percentage of Annual Cost Under SMR Structure	24.27%	12.14%	4.05%	2.43%	1.62%	1.21%
Percentage of Annual Cost Under New Structure	1.23%	1.23%	1.23%	1.23%	1.23%	1.21%

*All numbers are preliminary estimates; calculations use generating cost of $40/MWh[1] for micro-reactors, 95% capacity factor

[1]NEI Report, *Cost Competitiveness of Micro-Reactors for Remote Markets* (April 15, 2019).

Path Forward

- Release of NEI position paper on non-LWR annual fees, end of August
- Develop non-LWR annual fees; costs for developing advanced reactor regulatory infrastructure fee-exempt under NEIMA[2] (until 2031)
- Use future operating experience of SMRs and non-LWRs to:
    - Verify the expectations that advanced reactors require less regulatory service due to improved safety and simplicity
    - Refine the SMR and micro-reactor annual fees as detailed information becomes available

[2] See Section 102 (b)(1)(B)(iii) of the Nuclear Energy Innovation and Modernization Act, Public Law 115-439

10 CFR Part 53
“Licensing and Regulation of Advanced Nuclear Reactors”

August 20, 2020

Background

- Advance Notice of Proposed Rulemaking, “Approaches to Risk-Informed and Performance-Based Requirements for Nuclear Power Reactors,” dated May 4, 2006 (71 FR 26267)
- NRC’s Vision and Strategy report (12/16) for non-light-water reactors and related implementation action plans identified a potential rulemaking to establish a regulatory framework
- Nuclear Energy Innovation and Modernization Act (NEIMA; Public Law 115-439) signed into law in January 2019 requires the NRC to complete a rulemaking to establish a technology-inclusive, regulatory framework for optional use for commercial advanced nuclear reactors no later than December 2027
- Periodic Stakeholder Meeting – October 10, 2019

Background - NEIMA

(1) ADVANCED NUCLEAR REACTOR—The term “advanced nuclear reactor” means a nuclear fission or fusion reactor, including a prototype plant… with significant improvements compared to commercial nuclear reactors under construction as of the date of enactment of this Act, …

(9) REGULATORY FRAMEWORK—The term “regulatory framework” means the framework for reviewing requests for certifications, permits, approvals, and licenses for nuclear reactors.

(14) TECHNOLOGY-INCLUSIVE REGULATORY FRAMEWORK—The term “technology-inclusive regulatory framework” means a regulatory framework developed using methods of evaluation that are flexible and practicable for application to a variety of reactor technologies, including, where appropriate, the use of risk-informed and performance-based techniques and other tools and methods.

SECY-20-0032, Rulemaking Plan

- SECY-20-0032, “Rulemaking Plan on “Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors,” dated April 13, 2020
- Proposing a new 10 CFR part that could address performance requirements, design features, and programmatic controls for a wide variety of advanced nuclear reactors throughout the life of a facility.
- Focus the rulemaking on risk-informed functional requirements, building on existing NRC requirements, Commission policy statements, and recent activities (e.g., SECY-19-0117)
- Expect extensive interactions with external stakeholders and the Advisory Committee on Reactor Safeguards (ACRS) on the content of the rule.
- Awaiting Commission’s Staff Requirements Memorandum; including schedule goals

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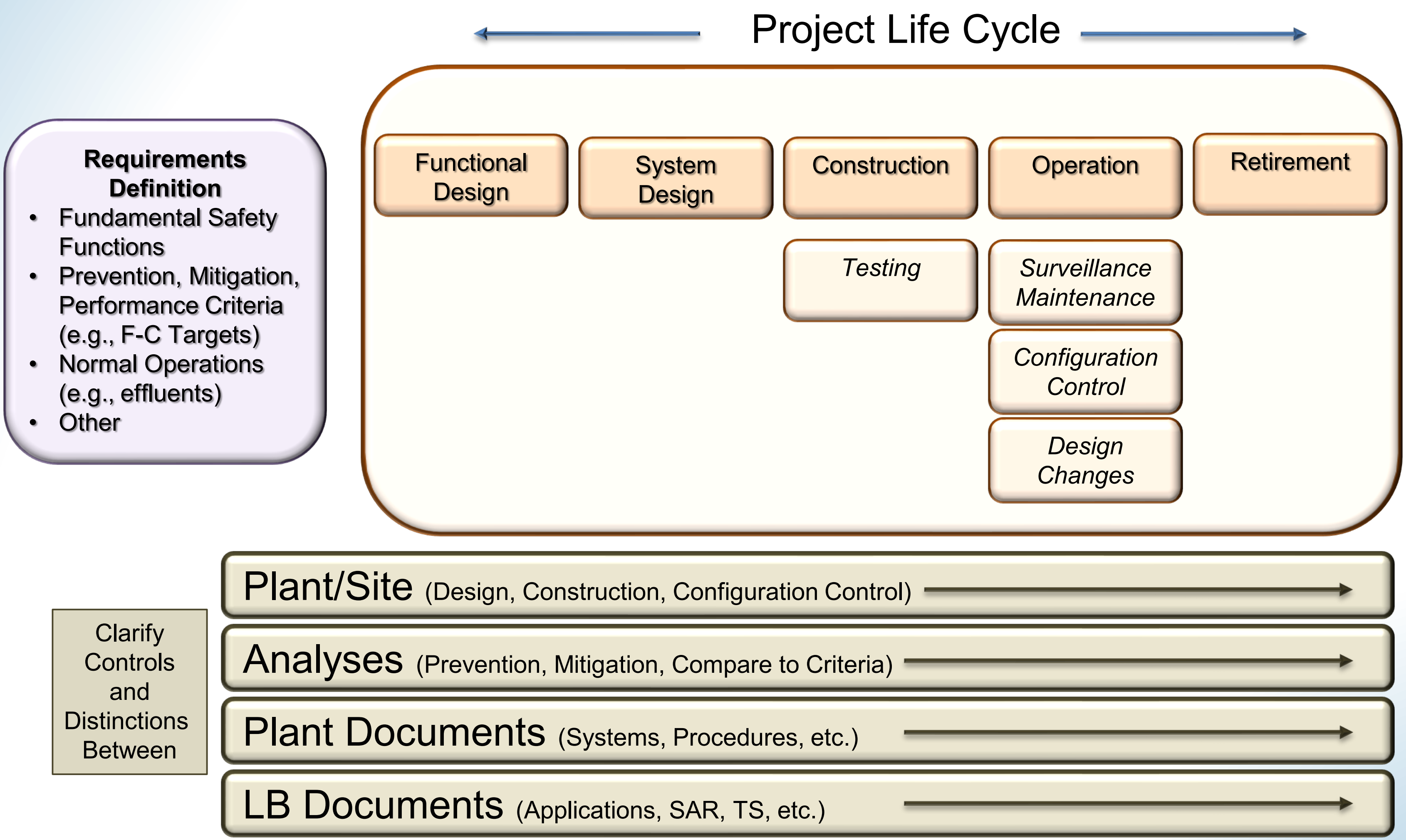
Protecting People and the Environment

Technology Inclusive Regulatory Framework



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Example – Possible Layout

- General Provisions
- Technology-Inclusive Safety Objectives
  - Regulatory limits, safety goals
- Design Requirements
- Siting
- Construction and Manufacturing Requirements
- Requirements for Operation
- Decommissioning Requirements
- Applications for Licenses, Certifications and Approvals
- Maintaining and Revising Licensing Basis Information
- Reporting and Administrative Requirements

NRC Staff White Paper

- The NRC staff developed a white paper (ADAMS ML20195A270) to support discussions with ACRS and other stakeholders

- Soliciting information that:
  1) Defines the scope of stakeholder interest in a rulemaking to develop a technology inclusive framework for advanced nuclear reactors,
  2) Identifies major issues and challenges related to technology-inclusive approaches to licensing and regulating a wide variety of advanced nuclear reactor designs,
  3) Supports prioritizing and developing plans to resolve identified issues within the rulemaking for the wide variety of advanced nuclear reactor designs, and
  4) Supports the development of the proposed rule and related guidance.

- Staff receptive to feedback on any aspect of developing a technology-inclusive regulatory framework to support the regulatory objective, whether or not in response to a question listed in this white paper or future solicitations.

United States Nuclear Regulatory Commission
Protecting People and the Environment

Part 53 Rulemaking Objectives

1) Provide reasonable assurance of adequate protection of the public health and safety and common defense and security at reactor sites at which advanced nuclear reactor designs are deployed, to at least the same degree of protection as required for current-generation light water reactors;
2) Protect health and minimize danger to life or property to at least the same degree of protection as required for current-generation light water reactors;
3) Provide greater operational flexibilities where supported by enhanced margins of safety that may be provided in advanced nuclear reactor designs;
4) Ensure that the requirements for licensing and regulating advanced nuclear reactors are clear and appropriate; and
5) Identify, define, and resolve additional areas of concern related to the licensing and regulation of advanced nuclear reactors.

Questions for Public Feedback

1. Regulatory Objectives
   - Appropriate, understandable, achievable ?
2. Scope and Types of Advanced Nuclear Reactors
   - Limit to advanced reactors as defined in NEIMA?
3. Technical Requirements versus Licensing Process
   - Limit to regulations related to technical standards?
   - Alternative licensing processes?
4. Performance Criteria
   - Technology-inclusive performance criteria?
5. Risk Metrics
   - Include risk metrics in the regulations?
6. Facility Life Cycle
   - How could new Part 53 align with facility life cycle
7. Definitions
   - Should Part 53 use existing definitions

Questions for Public Feedback

8. Performance-Based Regulation
   - How to incorporate performance-based concepts?
9. Identifying Levels of Protection
   - Differentiate requirements for adequate protection and safety improvements?
10. Integrated Approach to Rulemaking
    - How to integrate safety, security, emergency preparedness?
11. Consistency with Historical Standards
    - Use of existing standards (e.g., safety goals)?
12. Quality Standards
    - Recognize alternatives to Appendix B?
13. Stakeholder Documents, Standards, Guidance
    - Stakeholder interest in preparing guidance?
14. Other Issues?

Path Forward

- Awaiting Commission Decision on Rulemaking Plan (SECY-20-0032)
- Some stakeholders recommending accelerating schedule from rulemaking plan/NEIMA
    - See Letter dated May 14, 2020 from Senator Barrasso, Chairman Committee on Environment and Public Works (ML20136A164), and Response dated June 17, 2020 from Chairman Svinicki (ML20155K912)
- Accelerating schedule would result in need to have more active stakeholder engagement during 2021
- Public meeting dedicated to developing Part 53 tentatively scheduled for September 17th
    - White paper (ADAMS ML20195A270) provides possible topics

Part 53 Rulemaking

Marc Nichol
Senior Director New Reactors

August 20, 2020

1. Regulatory Objectives

- Establish a regulatory framework for new reactors that:
    - Provides reasonable assurance of adequate protection of the public health and safety and common defense and security
    - Is risk-informed, performance-based, and technology-inclusive
    - Is clear, flexible and efficient
    - Enables efficient foreign licensing of NRC approved designs
- Utilizes a rulemaking process that:
    - Starts with only the necessary legal requirements (e.g., AEA) as a blank-sheet approach
    - Considers all known, and unknown, reactor technologies
    - Benefits from lessons-learned through near-term licensing of new reactors

2. Scope and Types of Reactors

3. Type of Requirements

- Part 53 should be more inclusive, not less inclusive
  - All new reactor applications
  - All types of applications
  - All uses and applications
  - All power levels
- Address requirements based on needs
  - Technical requirements – complete redesign
  - Administrative requirements – improve efficiency, potential for some to be eliminated
  - Process requirements – utilize Part 50 and 52, improve/add additional flexibility and efficiency

12. Quality Assurance

- Appendix B
  - Innovative thinking when created in early 1970s
  - Only used by US nuclear industry
  - Shrinking supply chain
- ISO-9001
  - Achieves equivalent level of quality with Appendix B
  - Utilized world-wide by millions
- Benefits of using ISO-9001
  - Access to larger supply chain (higher quality)
  - Informed by broad experience (best practices)
  - Adoption of standards (more efficient)

9. Levels of Protection

7. Definitions

- Requirements should contain
  - High level standard: “reasonable assurance of adequate protection”
  - Inclusive performance objectives
  - Flexible for different licensing approaches
- Guidance could include
  - Technology specific acceptance criteria
- Need to create the Part 53 safety “paradigm” before addressing terms and definitions
  - Construct of demonstrating “reasonable assurance of adequate protection”
    - E.g., rethink: design basis, safety-related, defense-in-depth
  - Balance of deterministic and probabilistic methods

Industry's Activities

1. Evaluation of Atomic Energy Act
    - Statutory requirements relevant to Part 53
    - Statutory requirements may need to be modified
2. Envision a new Part 53 safety paradigm
    - Create a new bridge from AEA to Part 53
    - Consider scope of reactor technologies
    - Promote flexibility and efficiency
    - Evaluate international regulatory paradigms
3. Evaluate existing regulatory framework to identify what should be new for, and what could be incorporated into, Part 53
    - Scope (e.g., security, decommissioning)
    - Regulatory precedent (e.g., risk metrics, performance criteria)

QUESTIONS?

By: Third Way, GENSLER

U.S. Nuclear Industry Council Comments regarding Part 53 at NRC Stakeholders Meeting

Cyril W. Draffin, Jr.
Senior Fellow, Advanced Nuclear
U.S. Nuclear Industry Council

20 August 2020

Overall Comments

- USNIC welcomes opportunity to engage with NRC to develop Part 53
  - will actively participate in NRC Part 53 discussion in September 2020
- USNIC providing NRC with 50 comments addressing each of the 14 issues that the NRC raised in their July 2020 NRC Staff White Paper
  - 14 issues NRC identified is a good start for Part 53 planning
  - only a few of USNIC specific comments presented in these slides (due to time)
- Goal should be to craft a flexible Part 53 process that is so well defined that developers want to use it over existing Parts 50 and 52
- Part 53 should be technology inclusive

Specific Comments (on selective issues)

1. **Regulatory Objectives**

- NRC regulatory objectives for Part 53 are generally good

2. **Scope**

- Scope should be inclusive of all future applications and technologies.
- Scope should be graded approach to facilitate First-Of-A-Kind reviews but flexible enough to accelerate “nth” of a kind reviews
- Part 53 should be available to all Advanced Reactors technologies, but Advanced Reactor developers should not be compelled to use

3. **Licensing Process**

- Should address licensing, administrative, procedural, reporting and inspection matters for Advanced Reactor applications
- Goal to meet adequate protection standards, but in way that focuses on public health and safety and avoids unnecessary burden

Specific Comments (on selective issues)

9. **Levels of Protection**

- NRC should not use the development of this rule to rachet up requirements
- May be helpful to identify what prior regulations have been “justified as cost-effective safety improvements”

10. **Integrated Approach**

- Desirable to apply risk informed approaches to safety, security and emergency preparedness (as Commission did recently for Emergency Planning Zones)

12. **Quality Assurance**

- Part 53 provides opportunity for NRC to take a fresh look at Appendix B and NQA-1 Program
- Level of quality of commercially available components meets and frequently exceeds prior “nuclear standards” without the need for the overly burdensome reporting requirements
- Alternative approaches such as ISO 9000 series and commercial dedication programs should be considered

14. **Other issues**

- When available, we look forward to understanding timeline for the Commission to review and vote on Part 53 SECY paper

United States NIC Nuclear Industry Council

Closing Comments

- USNIC believes today is first step on interactive approach to developing an effective and useful Part 53
- USNIC welcomes opportunity to continue the dialog with NRC staff to achieve rule that is fully effective in meeting the Adequate Protection Standard-- but does in a way that allows Advanced Reactors to be developed, licensed, and deployed in a manner that avoid unnecessary burden -- and enables the deployment of these important contributors to avoiding carbon emissions

U.S. Nuclear Industry Council Contacts

For questions contact

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U.S. Nuclear Industry Council

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Chairman, US Nuclear Industry Council
Advanced Reactors Task Force

U.S. NRC Commissioner (1998-2007)

Jeff.Merrifield@pillsburylaw.com

Status – Spent Fuel Reprocessing Rulemaking

Jonathan Marcano, P.E.
NMSS/DFM

August 20, 2020

Background

- In 2013, the Commission directed the staff to develop a reprocessing rule focused on light water reactors (SRM-SECY-13-0093).
  - Limited scope to resolving Gap 5 (of 21) - safety and risk analysis.
  - Engage DOE to assess ongoing activities.
  - Regulatory basis for rule due 3/31/2021.
- Between 2013-2016, the NRC staff worked to develop a draft regulatory basis for Gap 5.
- In 2016, NRC suspended the work.
  - NRC budgetary constraints.
  - Apparent lack of commercial interest in constructing and operating a reprocessing facility.

Background

- On March 4, 2020, NRC held a public meeting to discuss status of the proposed rulemaking and to obtain stakeholder input.
  - Staff informed stakeholders that a limited scope rulemaking would cost approximately $2.4 million dollars.
  - Assess interest regarding continuation of rulemaking.
- On May 28, 2020, the Nuclear Energy Institute (NEI) and American Nuclear Society sent letters encouraging the NRC to assess the needs of advanced reactors prior to discontinuing efforts on the proposed rulemaking.

Current State

- NRC staff assessed the interest from the Advanced Reactor community and engaged with DOE to determine the need to continue rulemaking activities.
  - Some designers have the capability to eventually source their fuel from the spent fuel of other reactors.
  - NRC staff is not aware of any definitive vendor interest in pursuing reprocessing activities in the near future (next decade).
  - No near-term industry or DOE initiatives are currently planned or undergoing associated with reprocessing of spent light water reactor fuel or potential efforts to reprocess spent HALEU fuel for reuse in advanced reactors.
- NEI working group to assess community interest.

Next Steps

- NRC staff plans to inform the Commission of its recommendation regarding any proposed rulemaking for spent fuel reprocessing on or before 3/31/2021.
- In the future, NRC staff encourages early interactions from developers on anticipated needs or activities involving reprocessing.

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Contacts and References

- Email feedback to Jonathan.Marcano@nrc.gov and Tom.Boyce@nrc.gov.
- References
  - March 4, 2020, Public Meeting Summary (ADAMS Accession No. ML20077K144).
  - Letter from the Nuclear Energy Industry (ADAMS Accession No. ML20154K554).
  - Letter from the American Nuclear Society (ADAMS Accession No. ML20154K530).
  - SRM-SECY-13-0093, "Reprocessing Regulatory Framework – Status and Next Steps," dated November 4, 2013 (ADAMS Accession No. ML13308A403).

Overview of the Oak Ridge National Laboratory Report on Preparing and Reviewing a Molten Salt Non-Power Reactor Application

William B. Kennedy
Project Manager
Non-Power Production and Utilization Facility Licensing Branch
Division of Advanced Reactors and Non-Power Production and Utilization Facilities
U.S. Nuclear Regulatory Commission

Background

- In response to the Nuclear Energy Innovation and Modernization Act of 2019, the NRC staff identified an opportunity to enhance its readiness to license non-power reactors that will use molten salt reactor (MSR) technology
- Under contract with NRC, Oak Ridge National Laboratory developed a report titled, "Proposed Guidance for Preparing and Reviewing a Molten Salt Non-Power Reactor Application"

Overview of the Report

- An information resource for stakeholders interested in licensing of non-power MSRs
- Based on NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors"
- Focuses on the technical information needed to apply NUREG-1537 to a non-power MSR licensing application

Overview of the Report

- Covers topics including:
  - Siting
  - Design of structures, systems, and components
  - Reactor description
  - Reactor cooling systems
  - Engineered safety features
  - Instrumentation and control systems
  - Auxiliary systems
  - Radiation protection and waste management
  - Accident analysis
  - Technical specifications

Future Plans

- The NRC staff intends to endorse the report for use by potential non-power MSR applicants by January 2021
- Subsequently, the report will be incorporated in durable guidance (likely the next revision of NUREG-1537)
- An update will be provided at the next advanced reactors planning meeting and any feedback on the report will be welcome

How to Get the Report

- Available on the NRC's Agencywide Documents Access and Management System (ADAMS) at Accession No. ML20219A771
- Posted on the NRC's public website on the advanced reactors page at https://www.nrc.gov/reactors/new-reactors/advanced.html under the heading, "Advanced Reactor Reference Materials"
- Contact me at william.kennedy@nrc.gov

Future Meeting Planning and Open Discussion

2020 Tentative Schedule for Periodic Stakeholder Meetings
August 25 (GEIS for Advanced Reactors)
August 27 (TICAP, ARCAP, and Construction Permit)
September 17 (10 CFR Part 53)
September 24 (TICAP and ARCAP)
October 1
November 5