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Protecting People and the Environment

Presentations for March 28, 2019 Public Meeting
Regulatory Improvements for Advanced Reactors

In order of discussion, the meeting included the following topics and presentations

1) NRC Slides

2) Advanced Reactor Siting Policy Considerations
Oak Ridge National Laboratory (R. Belles, et al)

3) Non-Light Water Reactor Mechanistic Source Term Study
Sandia National Laboratories (A. Clark, et al)

Advanced Reactor Program

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Public Meeting on Possible Regulatory Process Improvements for Advanced Reactor Designs

March 28, 2019

Telephone Bridge
(888) 793-9929
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Public Meeting

- Telephone Bridge

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  Passcode: 5149958

- Opportunities for public comments and questions at designated times

Outline

- Introductions
- Nuclear Energy Innovation and Modernization Act
- DOE/NE Update
- Siting Criteria Related to Populations
  - ORNL
  - NRC
  - NEI, General Discussion
- Mechanistic Source Terms
  - SNL
  - NRC
  - TWGs, General Discussion
- Status Update, Future Meetings

Nuclear Energy Innovation and Modernization Act

- NEIMA – Public Law No: 115-439; Jan. 13, 2019

- Sec. 101. NRC user fees and annual charges (2020)
- Sec. 102. NRC user fees and annual charges (2021 and beyond)
- ✓ **Sec. 103. Advanced nuclear reactor program.**
- Sec. 104. Baffle-former bolt guidance.
- Sec. 105. Evacuation report.
- Sec. 106. Encouraging research and test reactors.
- Sec. 107. Report on accident tolerant fuel.
- Sec. 108. Report on local community advisory boards.
- Sec. 109. Report on chilling effects study recommendations.

NEIMA

(1) ADVANCED NUCLEAR REACTOR.—The term ‘‘advanced nuclear reactor’’ means a nuclear fission or fusion reactor, including a prototype plant (as defined in sections 50.2 and 52.1 of title 10, Code of Federal Regulations (as in effect on the date of enactment of this Act)), with significant improvements compared to commercial nuclear reactors under construction as of the date of enactment of this Act, including improvements such as—

(A) additional inherent safety features;

(B) significantly lower levelized cost of electricity;

(C) lower waste yields;

(D) greater fuel utilization;

(E) enhanced reliability;

(F) increased proliferation resistance;

(G) increased thermal efficiency; or

(H) ability to integrate into electric and nonelectric applications.

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Sec. 103. Advanced nuclear reactor program.

a) Licensing
    1) Staged Licensing
    2) Risk Informed Licensing
    3) Research and Test Reactor Licensing
    4) Technology-Inclusive Regulatory Framework
    5) Training and Expertise
    6) Authorization of Appropriations
b) Report to Establish Stages in Licensing Process
c) Report to Increase RIPB Techniques
d) Report to Prepare RTR Licensing Process
e) Report to Complete Rulemaking

Advanced Reactor Program

Sec. 103(b). Report on Staged Licensing

(1) Report required within 180 days
- Stages in licensing process allowing implementation within 2 years

(2) Coordination and Stakeholder Input

(3) Cost and Schedule Estimates

(4) Required Evaluations

Sec. 103(b). Report on Staged Licensing

Item			Response
A	i	Unique aspects of adv reactor licensing	Technology-inclusive approaches (IAPs)
	ii	Strategies for fuel qualification	Activities underway (Strategy 5)
	iii	Policy issues	Routinely evaluated, prioritized (Strategy 5)
B	i	Licensing Project Plans	Roadmap, REPs, flexibility (Strategy 3)
	ii	Topical Reports	Roadmap, existing process (Strategy 3)
	iii iv	Standards setting organizations, Consensus codes and standards	ASME, ANS (Strategy 4)
	v	Conceptual design assessments	Roadmap, preapplications, SDA (Strategy 3)
	vi	Policies and guidance for staff	IAPs, team approach (Strategies 1, 3)
C		Efficiency, timeliness, and cost	REP, project plans, team approach (Strategy 3)
D		Improving predictability	REP, project plans, team approach (Strategy 3)
E		Commission action or modification of policy	None Identified

Sec. 103(c). Report on RIPB Techniques.

(1) Report required within 180 days
- Increasing, where appropriate, the use of risk-informed and performance-based evaluation techniques and regulatory guidance

(2) Coordination and Stakeholder Input

(3) Cost and Schedule Estimates

(4) Required Evaluations

Sec. 103(c). RIPB Techniques and Guidance

Item				Response
A	i	I	Licensing basis events	NEI 18-04 & DG-1353; related SECY
		II	Mechanistic source term	Future guidance
		III	Containment performance	SECY-18-0096
		IV	Emergency preparedness	SECY-18-0103
		V	Qualification of reactor fuel	ongoing
	ii	ii	Other policy issues	Routinely evaluated, prioritized
B			Commission action or modification of policy	None Identified

Sec. 103(d). Report on RTR Licensing

(1) Report required within 1 year
- Preparing the licensing process for research and test reactors

(2) Coordination and Stakeholder Input
- Seeking perspectives on potential modifications and enhancements to the NRC's RTR licensing process

(3) Cost and Schedule Estimates

(4) Required Evaluations
- (A) Unique aspects of RTR licensing and any associated legal, regulatory, and policy issues the Commission should address to prepare the licensing process
- (B) Feasibility of developing guidelines for advanced reactor demonstrations and prototypes to support the review process for advanced reactors designs
- (C) Extent to which Commission action or modification of policy is needed

Sec. 103(e). Report on Rulemaking

(1) Report required within 30 months
- Completing a rulemaking to estalblish technology-inclusive regulatory framework for optional use by advanced reactor applicants
- NRC capabilities to support evaluations, including qualification of advanced nuclear reactor fuel

(2) Coordination and Stakeholder Input

(3) Cost and Schedule Estimates

(4) Required Evaluations

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Break

Meeting/Webinar will begin shortly

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Advanced Reactor Program

Siting Discussions

- Background
- Longstanding Policy Issue
- Discussed in previous stakeholder meetings
  - December 14, 2017 (ML17354B219)
    - White Paper (ML17333B158)
  - May 3, 2018 (ML18130A688)
- Desire to move to resolution

Siting & Population

Presentation

Oak Ridge National Laboratory

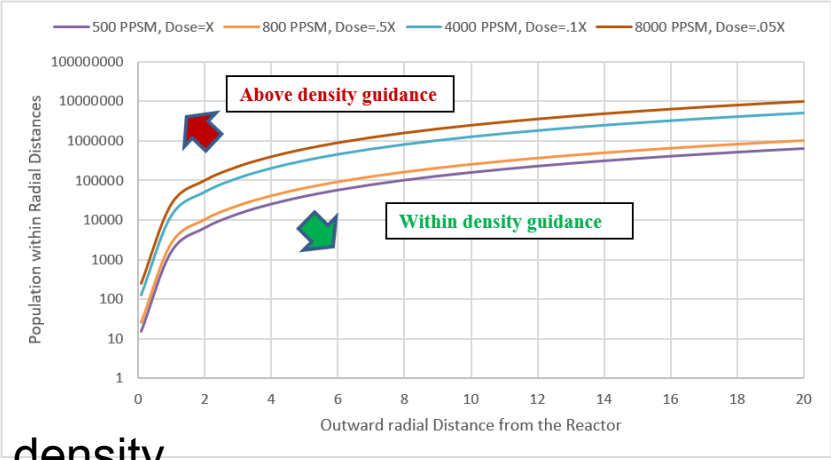
Siting Discussions

- Possible Approaches
  - Consistent with Current Regulation
  - Graded or Scaled Guidance for Population Density and/or Areas Limited by Population Density
  - Pursue larger effort (revisit regulations and past Commission policies)

Siting

Possible Approaches

- Variable Population Density
- Variable Radius for 500 ppsm
- Combination of radius, population density

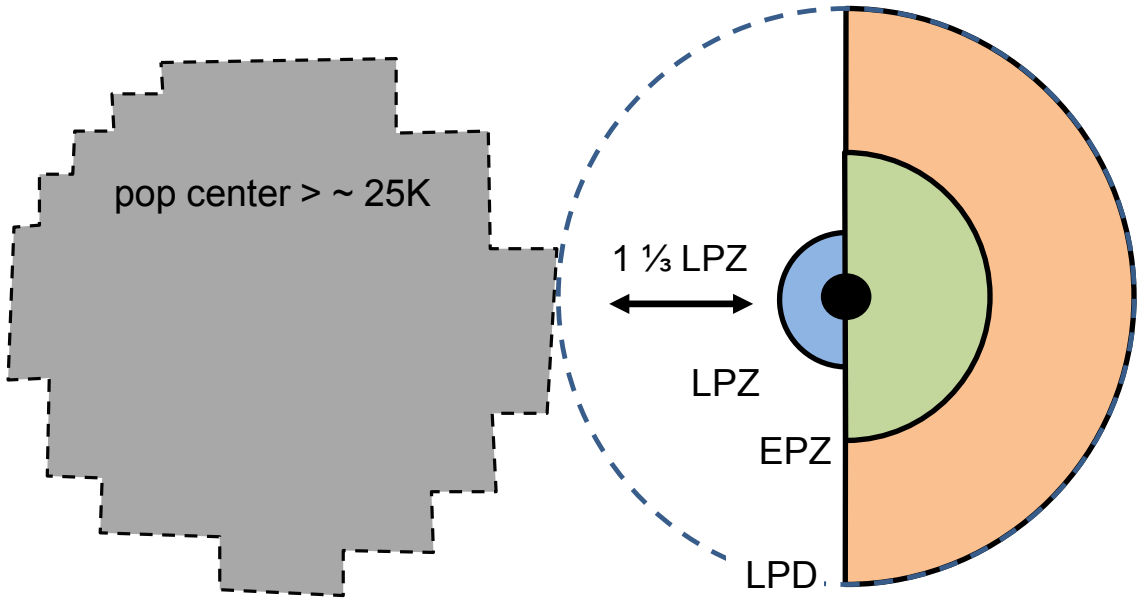


Siting Discussions

- Staff's Working Proposal
  - Consistent with Current Regulation
  - Graded or Scaled Guidance for Population Density
  - Represented by three cases (following slides)
- Path Forward
  - Stakeholder Interactions
  - Guidance Development
  - Commission Interaction

Siting Discussions: Case 1

LPZ – Event Sequences with Doses > 25R Beyond EAB
EPZ – Event Sequences with Doses > 1R Beyond EAB

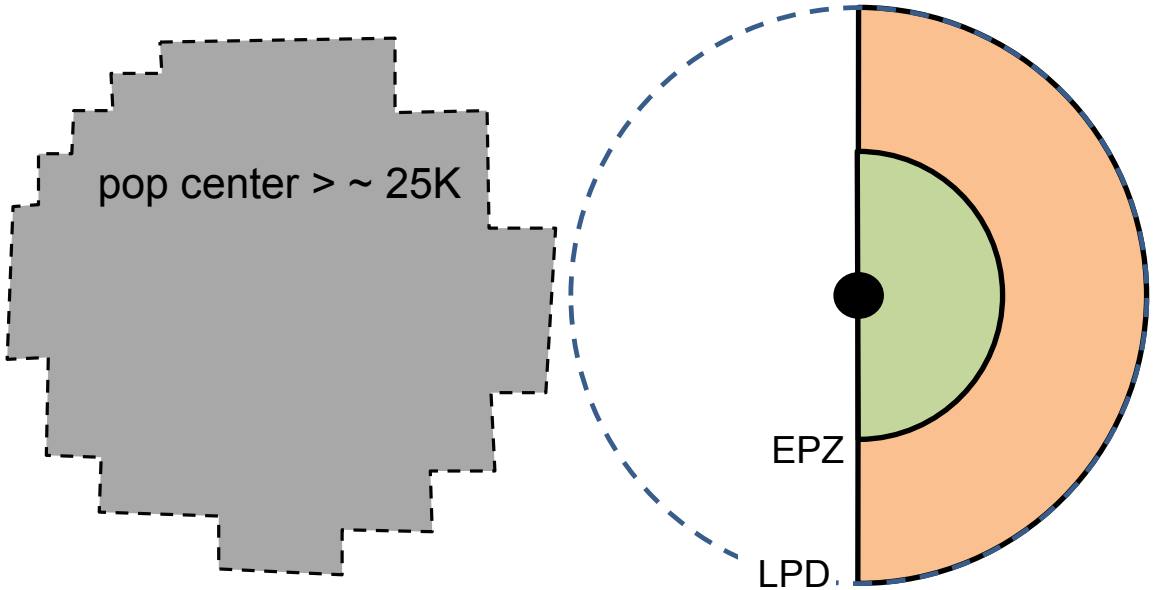


Population Density Guidance for EPZ > EAB

For plants with Event Sequence Doses > 1 R beyond the EAB, population density <500 persons per square mile over the radial distance equal to twice the radius of the EPZ

Siting Discussions: Case 2

No LPZ – No Event Sequences with Doses > 25R Beyond EAB
EPZ – Event Sequences with Doses > 1R Beyond EAB

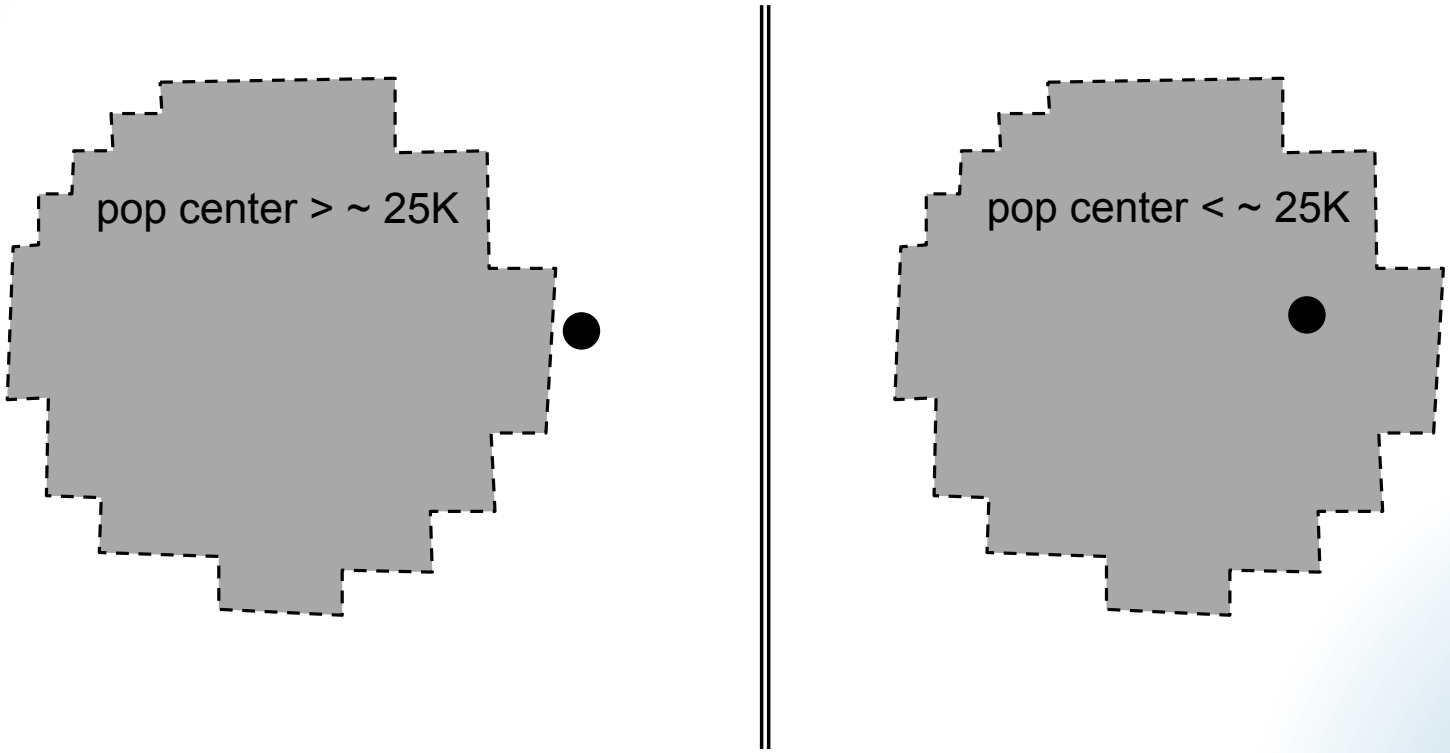


Population Density Guidance for EPZ > EAB

For plants with Event Sequence Doses > 1 R beyond the EAB, population density <500 persons per square mile over the radial distance equal to twice the radius of the EPZ

Siting Discussions: Case 3

No LPZ – No Event Sequences with Doses > 25R Beyond EAB
No EPZ – No Event Sequences with Doses > 1R Beyond EAB



Siting Discussions

DISCUSSION ?



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Lunch

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Advanced Reactor Program

Mechanistic Source Term Background

- TID-14844, Calculation of Distance Factors for Power and Test Reactors, (AEC, 1962)
- Alternate Source Term
  (10 CFR 50.67, 1999; Regulatory Guide 1.183, 2000; and NUREG-1465, 1995)

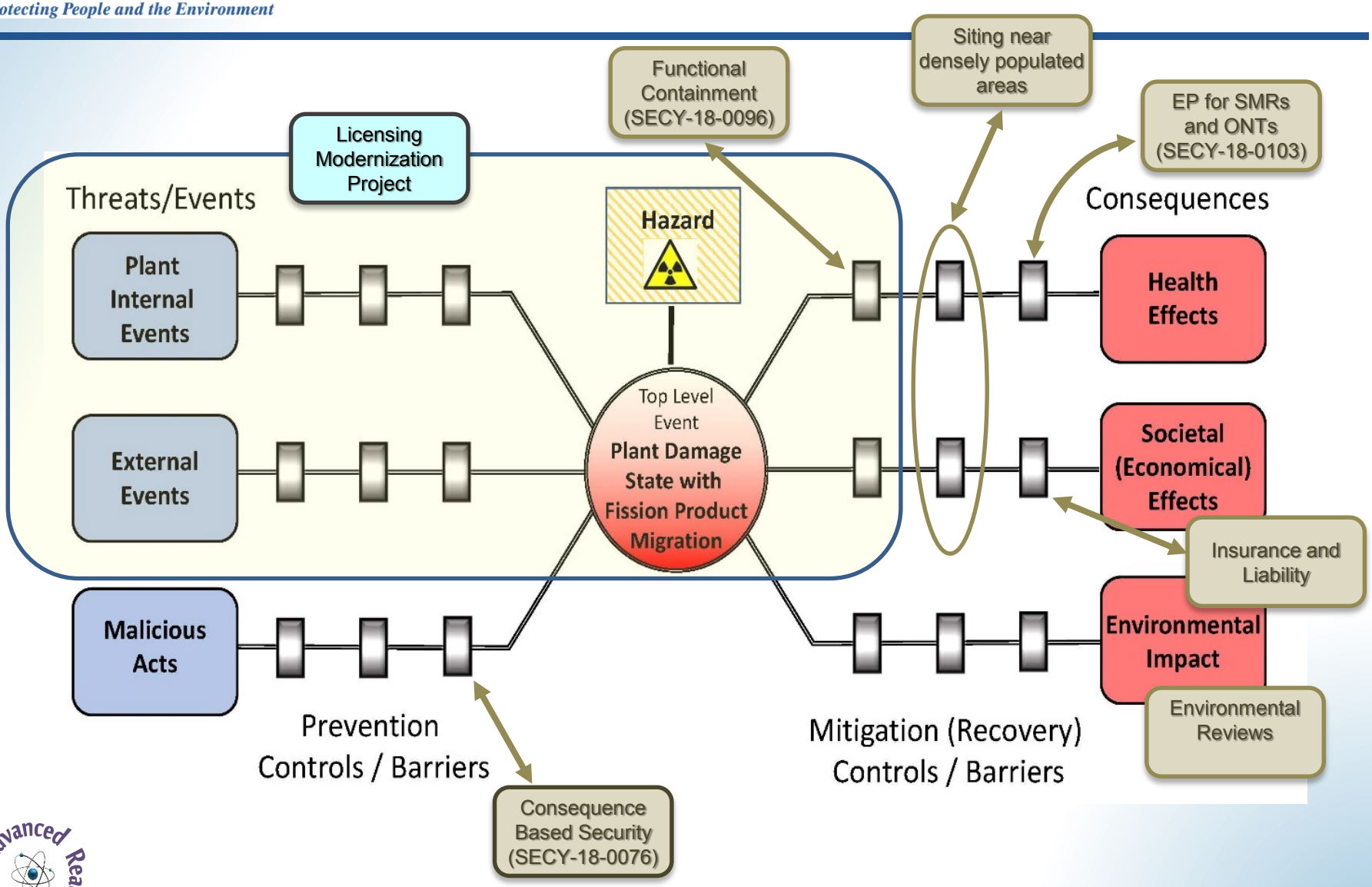
- SECY-93-092, "Issues Pertaining to Advanced Reactors…."
  - Fuel performance sufficiently well understood
  - Transport of radionuclides can be modeled
  - Consider severe accidents and uncertainties
- SECY-03-0047, "Policy Issues Related to Licensing Non-Light Water Reactor Designs"
- SECY-16-0012, "Accident Source Terms and Siting for Small Modular Reactors and Non-Light Water Reactors"

Mechanistic Source Term Background

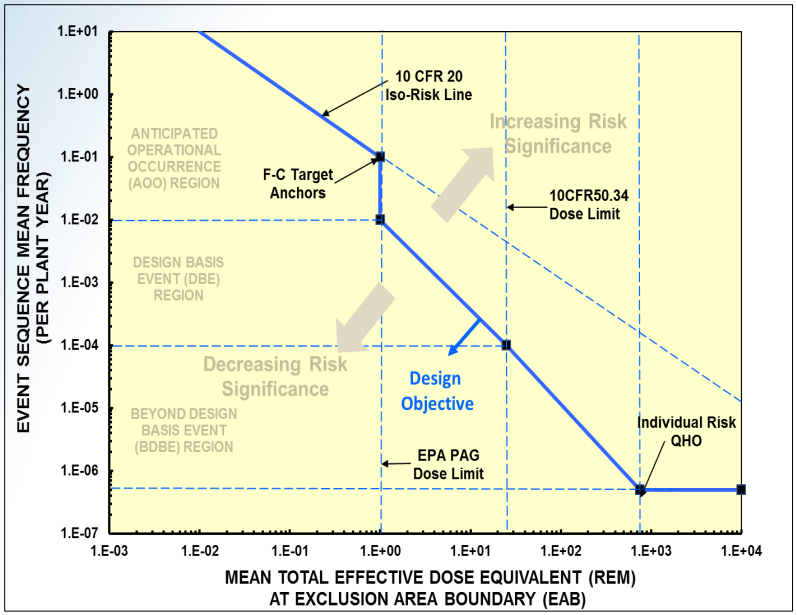
- ACRS Conclusions & Recommendations
  - Proposed Emergency Preparedness Rulemaking
    - For the rule to be applied effectively, the staff will also need to provide guidance to define their expectations for the technical adequacy of mechanistic source terms.
  - SECY Paper (DG-1353, NEI-18-04)
    - ML19078A240
    - The guidance proposed in DG-1353 is adequate to support implementation of the approach described in the SECY paper, with the exception that guidance for developing mechanistic source terms should be expanded.

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Integrated Design/Review

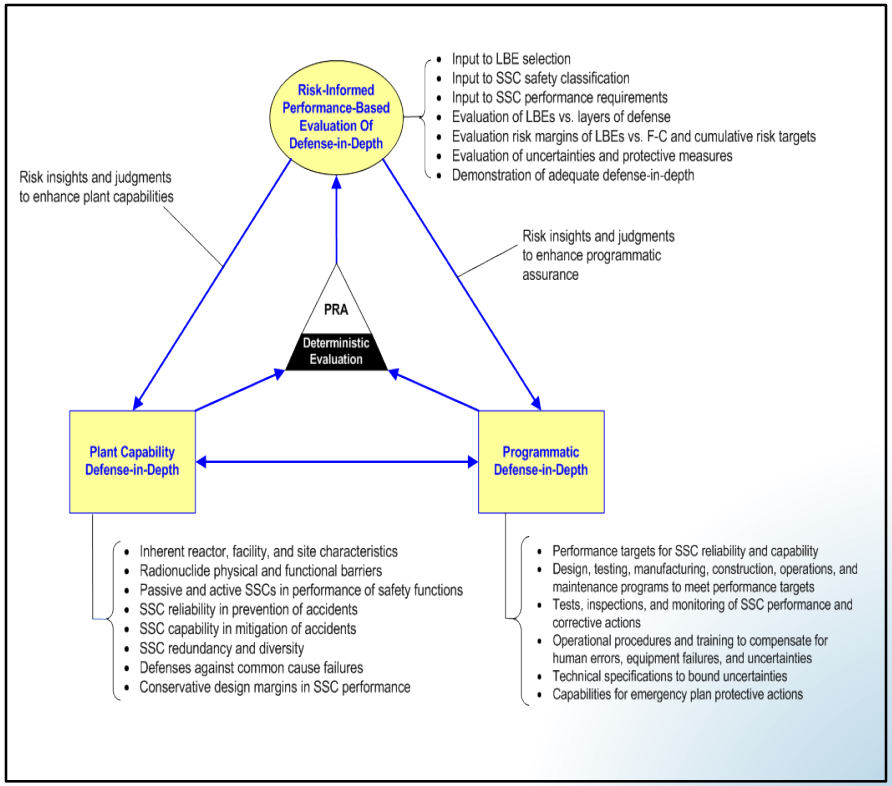


Licensing Basis Development



Underway:

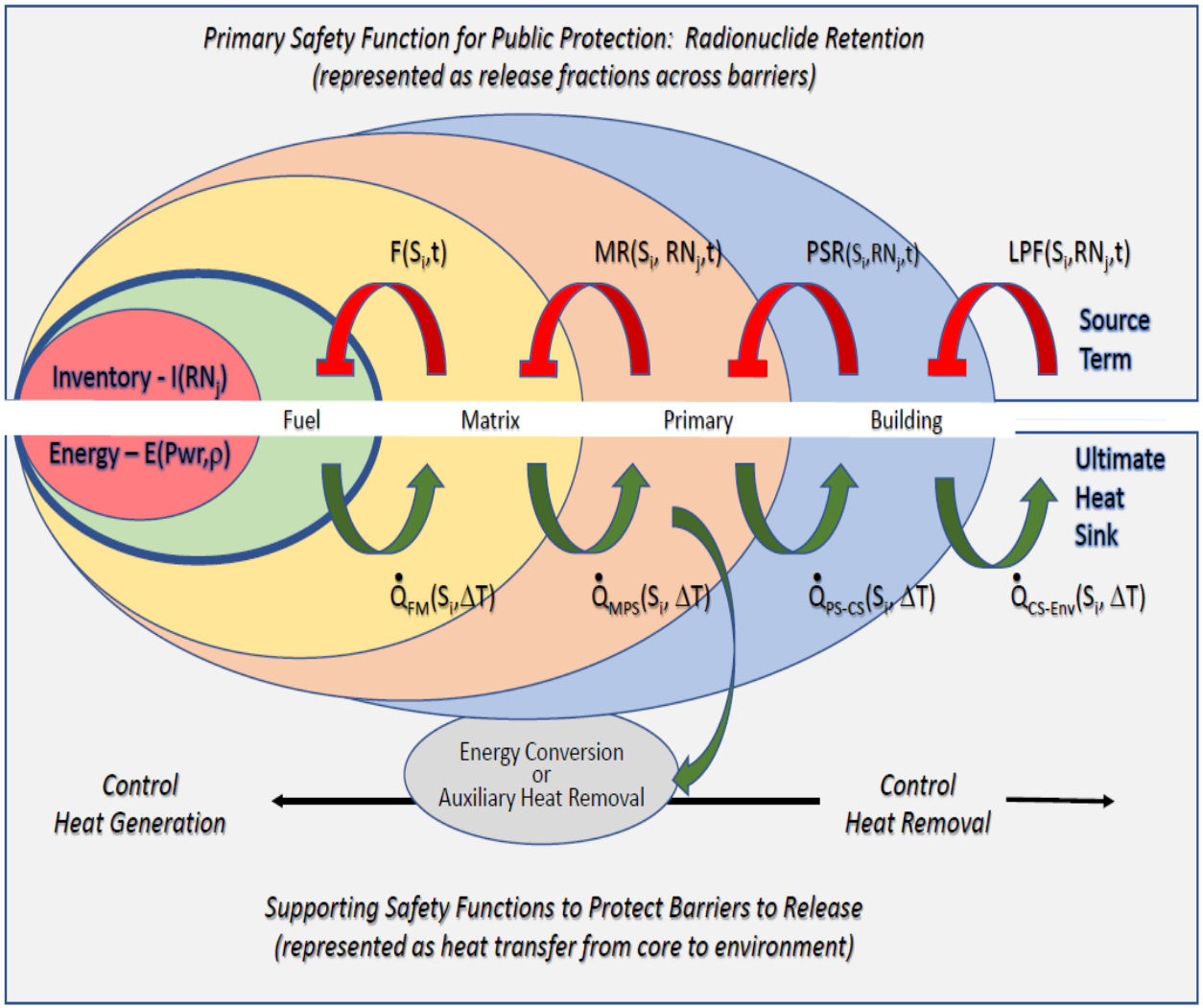
- DG-1353 (endorsing NEI 18-04)
- Related SECY Paper



Being Initiated:

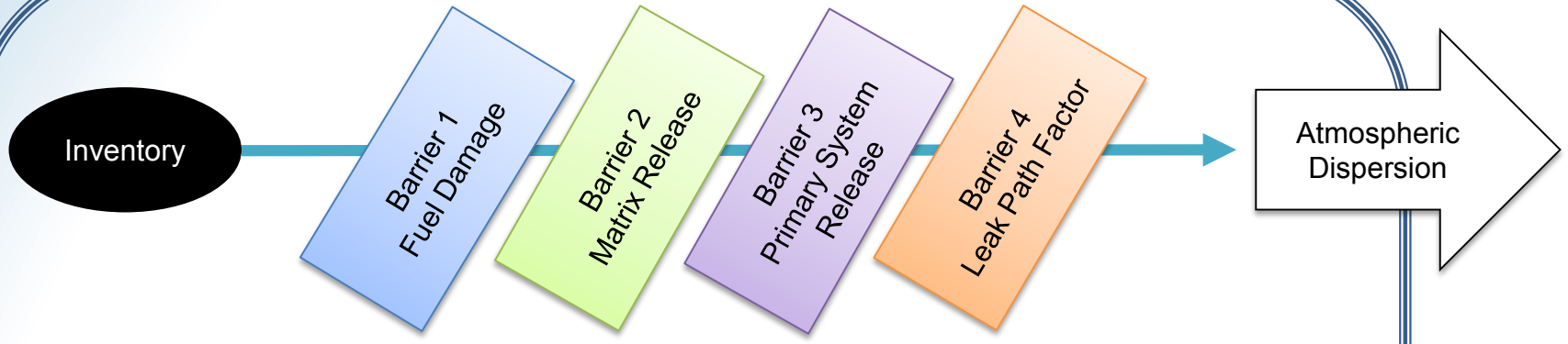
- Content of Applications
- Mechanistic Source Term
- Other ?

Fundamental Safety Functions and Mechanistic Source Term



$I(RN_j)$	Inventory
RN_j	Radionuclide Groups (j)
E	Heat Energy
Pwr	Power Level
ρ	Reactivity
F	Fuel Release Fraction
MR	Matrix Release Fraction
PSR	Primary System Release Fraction
LPF	Building Leak Path Factor
S_i	Event Sequences (i)
t	Time
$\dot{Q}$	Heat Transfer
FM	Fuel to Matrix
MPS	Matrix to Primary System
PS-CS	Primary System to Cooling System
CS-Env	Cooling System to Environment
ΔT	Temperature difference

Consequence Assessment



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

Initial inventory	Factors that determine how much of the inventory is released across a given barrier and thus persists to the source term

Each factor is, in turn, a function of its initial design characteristics (e.g., materials), operating conditions (e.g., burnup, aging) and transient conditions (e.g., time, temperatures, pressures).

Consequence Assessment

- NRC Tasking – Sandia National Laboratories

  … A simplified scoping analysis providing consequence approximations and estimates of relative likelihoods, along with key assumptions and uncertainties, will allow the NRC staff to prioritize activities, refine plans to resolve policy issues, and support regulatory engagement plans for specific reactor technologies and designs. …

  … The non-reactor approaches used to support Department of Energy facilities may be a useful starting point to organize information from past studies and provide estimates of the radiological risk from different technologies in a consistent format. These approximations consider such things as the radioactive inventories or material at risk (MAR); scenarios leading to the potential release of radioactivity; the type, chemical form, timing, and amount of radionuclides released for various scenarios (which might relate to terms such as damage ratio (DR) and airborne release fraction (ARF)); the effectiveness of structures, systems and components to limit the releases (which might relate to terms such as leak path factor (LPF)); …

Mechanistic Source Term

Presentation
Sandia National Laboratories

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Break

Meeting/Webinar will begin shortly

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Advanced Reactor Program

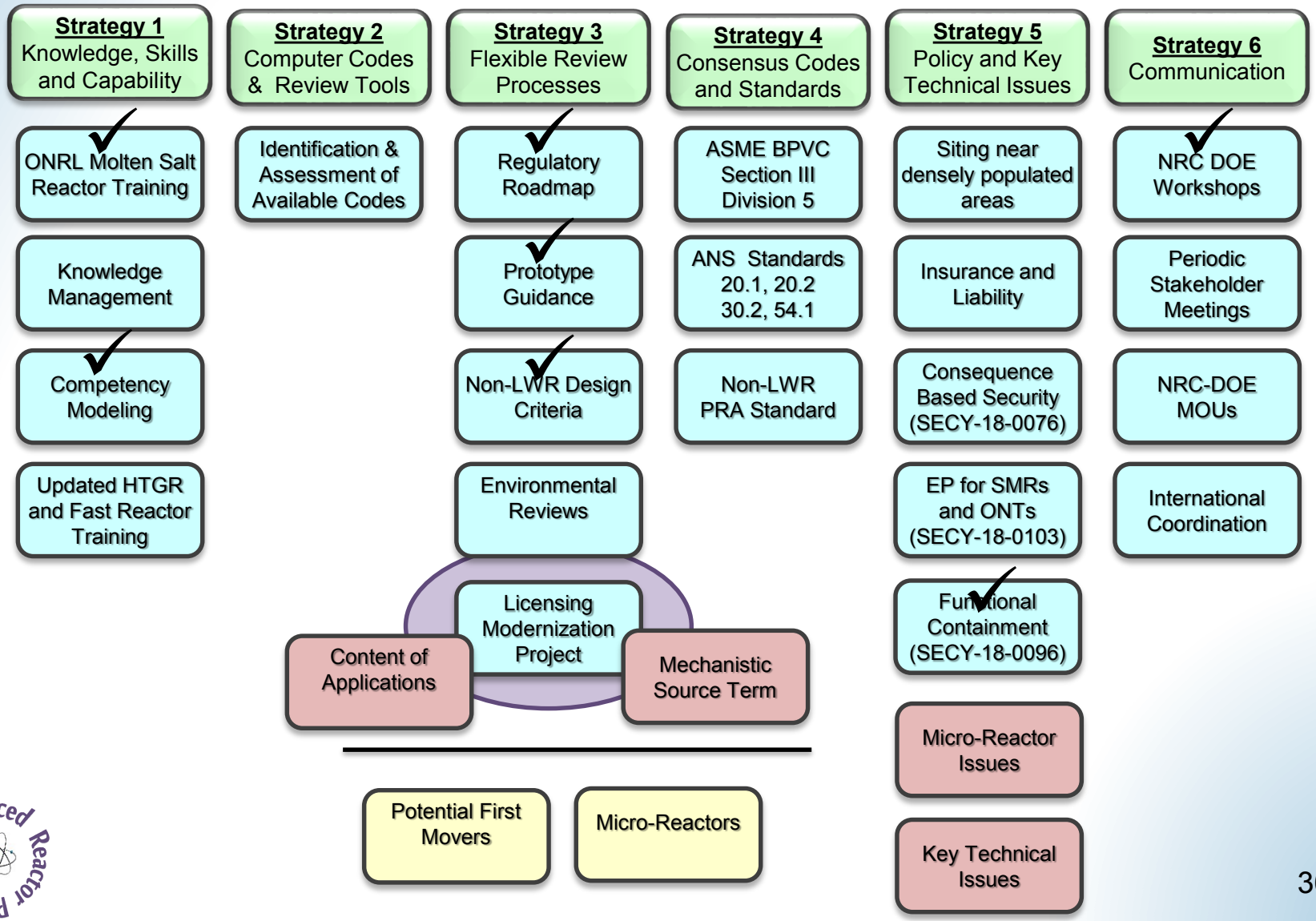
Mechanistic Source Term

- Discussion & Path Forward
  - High Level Primer/Guidance
  - Needs of Developers ?
  - Communication/Coordination Tool ?

DISCUSSIONS

- Technology Working Groups
- Other Stakeholders, Topics

Strategies & Contributing Activities



Policy Table

Ongoing Activities		
1	Prototype Guidance Staged Licensing	Roadmap *(plan to update)*
2a	Source Term	Prepare MST Guidance
	Dose Calcs	
	Siting	Prepare Siting Guidance
2b	SSC Design Issues	NEI 18-04, DG-1353
3	Offsite EP	SECY-18-103
4	Insurance/Liability	Future (2021) Report to Congress *(no change acceptable)*
5	PRA in licensing	NEI 18-04, DG-1353
6	Defense in Depth	NEI 18-04, DG-1353
7	Physical Security *(limited scope)*	SECY-18-0076 *(limited to sabotage)*

Policy Table

Ongoing Activities		
8	LBEs	NEI 18-04, DG-1353
9a	Fuel Qualification	technology specific
9b	Materials Qualification	technology specific
10a	MC&A Cat II facilities	ML18267A184
10b	Security Cat II facilities	ML18267A184
10c	Collaboration • criticality benchmark • HALEU shipping	
11	Functional Containment Performance Criteria	SECY-18-0096 & SRM
	Advanced Manufacturing	
	Environmental Reviews	
	Micro Reactors	

Policy Table

Key Technical Issues	Fuel Qualification
	Materials Qualification
	Civil/Structural

Policy Table

Open – Not Working		
1	Annual Fees	
2	Manufacturing License	
3	Process Heat	
4	Waste Issues	
5	Operator Staffing* Remote/Autonomous	
	Fusion Energy	

Policy Table

No Plans (Resolved or Need Feedback)		
1	Multi-module License	
2	Operator Staffing*	
3	Operational Programs	
4	Module Installation	
5	Decommissioning Funding	
6	Aircraft Impact Assessments	

Future Meetings

2019 Tentative Schedule; Periodic Stakeholder Meetings	
May 9	
June 27	
August 15	
October 10	
December 11	

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Advanced Reactor Siting Policy Considerations

Presentation for:

Regulatory Improvements for Advanced Reactor Designs Stakeholder Meeting

March 28, 2019

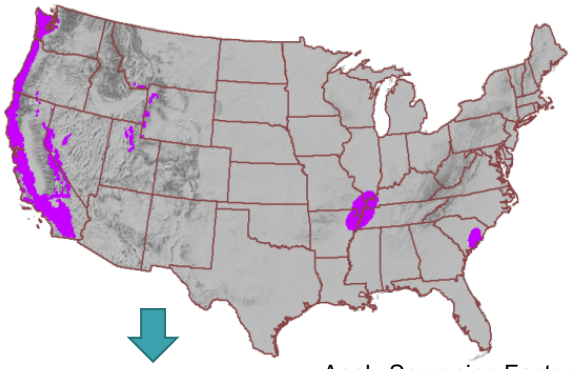
Randy Belles bellesrj@ornl.gov

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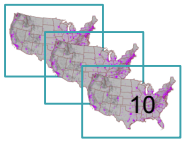
U.S. DEPARTMENT OF ENERGY

ORNL has developed a siting tool (OR-SAGE) that provides insights on the challenges and benefits of deploying SMRs

The tool uses geographical information systems (GIS) and spatial modeling techniques to visualize data

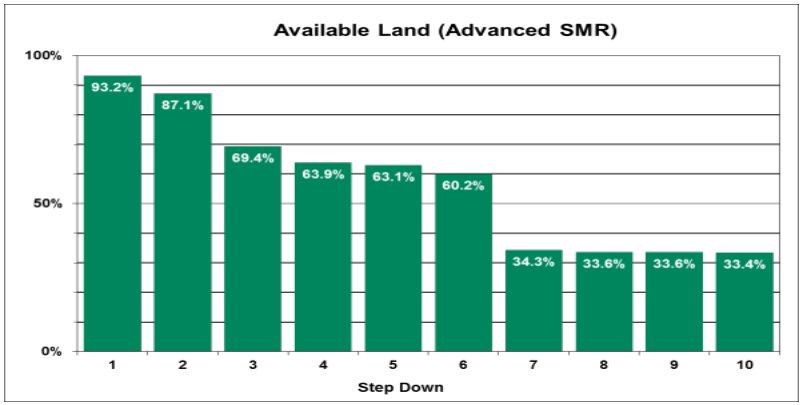


Apply Screening Factors



Land parsed into 100 x 100 m cells for evaluation

	Order of Exclusion Layers:
1	Population Density
2	Wetlands/Open Water
3	Protected Lands
4	Landslide Hazards
5	100-yr Floodplain
6	Slope
7	Cooling Water
8	Proximity to Fault Lines
9	Hazardous Operations
10	Safe Shutdown Earthquake



OR-SAGE - Oak Ridge Siting Analysis for power Generation Expansion

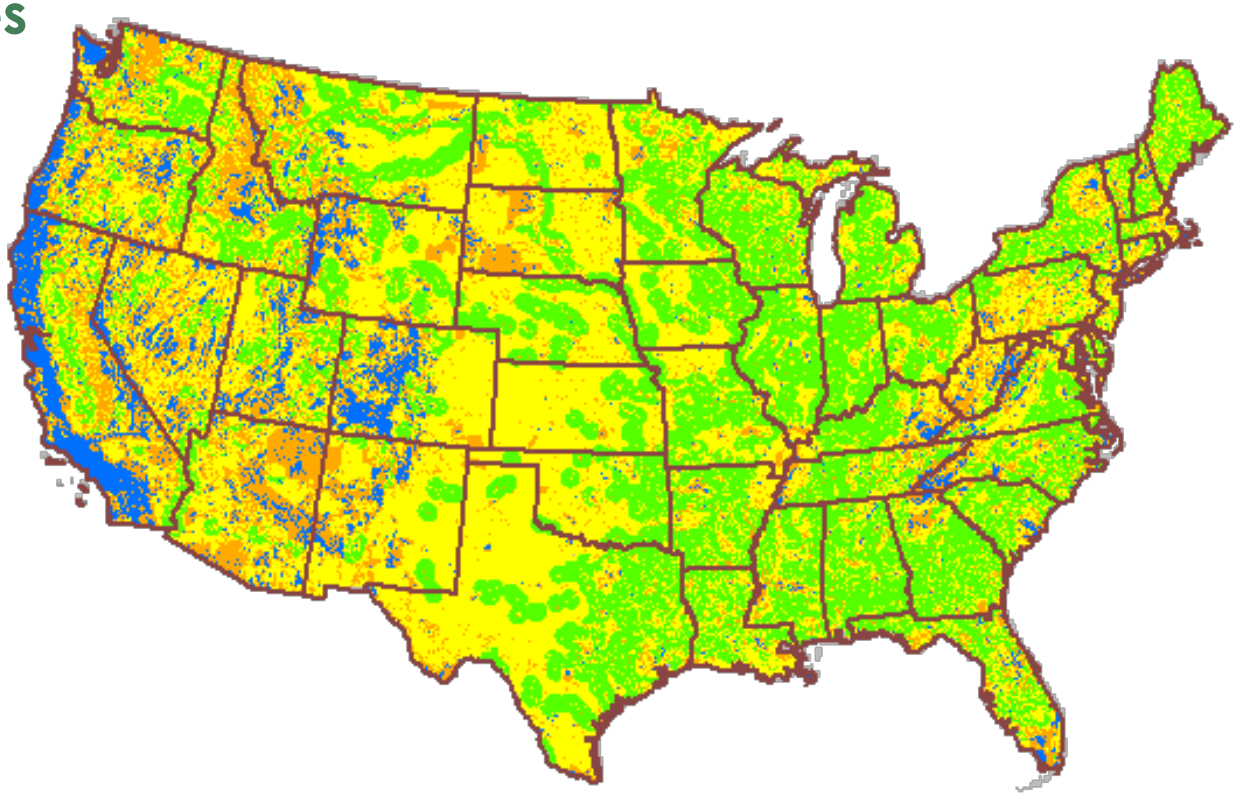
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Composite screening results provides additional insight into siting possibilities

Composite Map Result

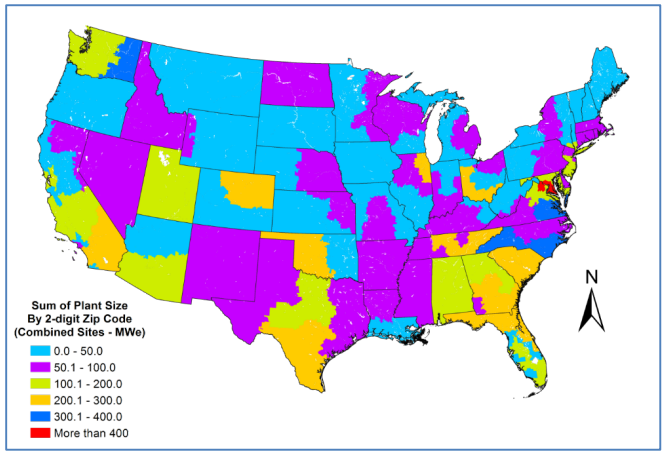
Green – Meets all Criteria
Yellow – Single issue
Orange – Two issues
Blue – 3+ issues

Yellow + Green = 74.7% potentially meets criteria

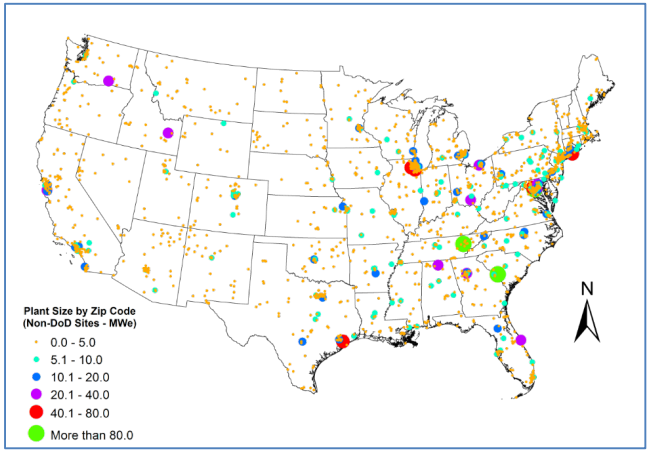


OR-SAGE provides capability to interrogate any 100 M x 100 M cell to evaluate status.

Previous projects have focused on population density

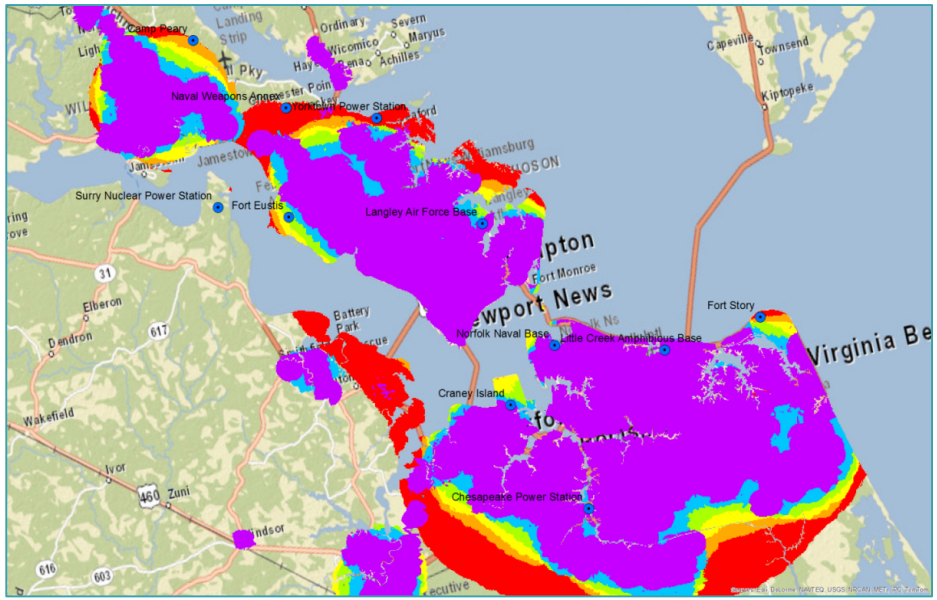


Federal power consumption by zip code and by facility



ORNL/TM-2013/578

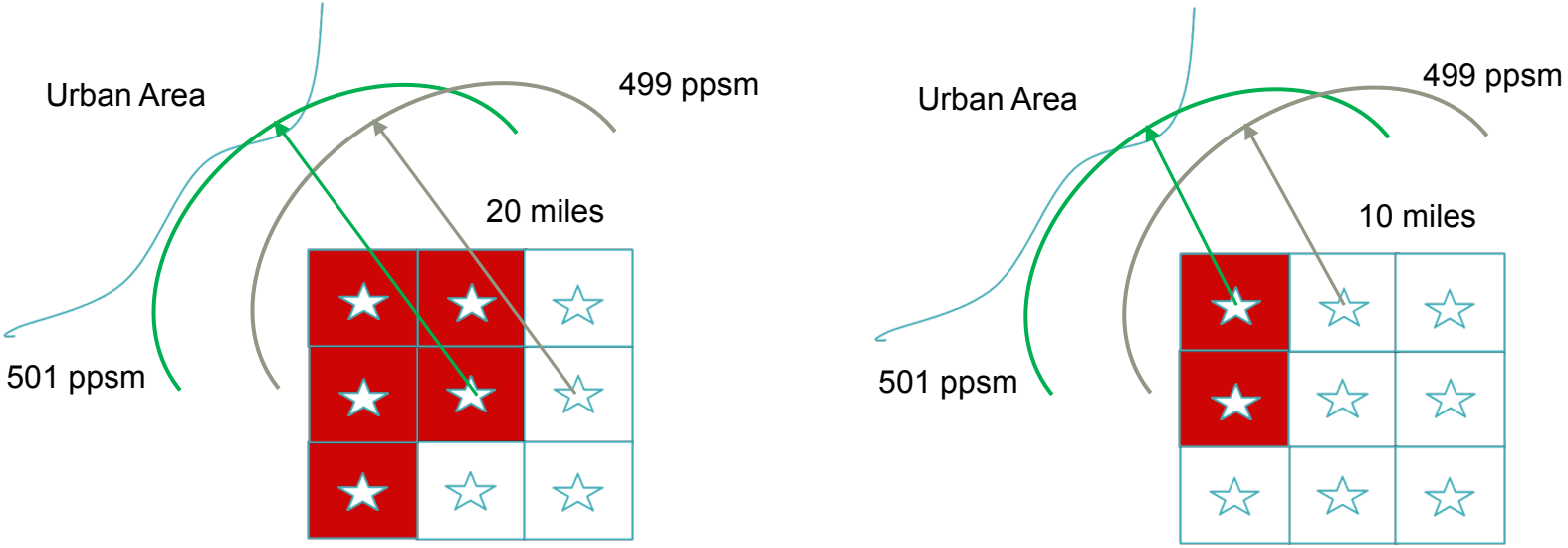
Evaluated urban population distribution and potential for nearby SMR siting:
The figure below shows 500 ppsm limit calculated at 1, 2, 3, 4, 5, and 10 miles. Study evaluated SMR potential of blue-dot sites.



ORNL/TM-2014/300

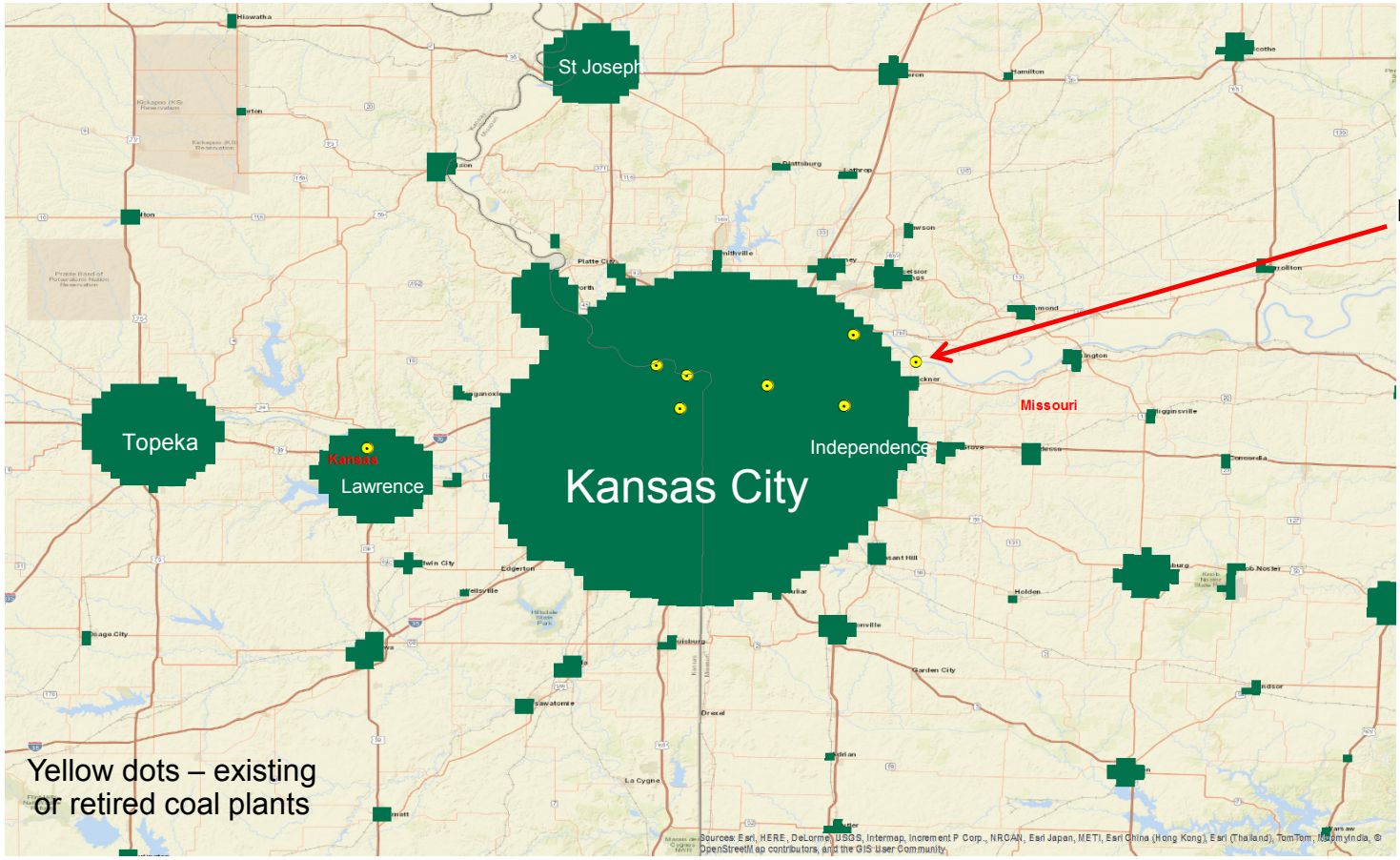
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Population density calculations and visual display



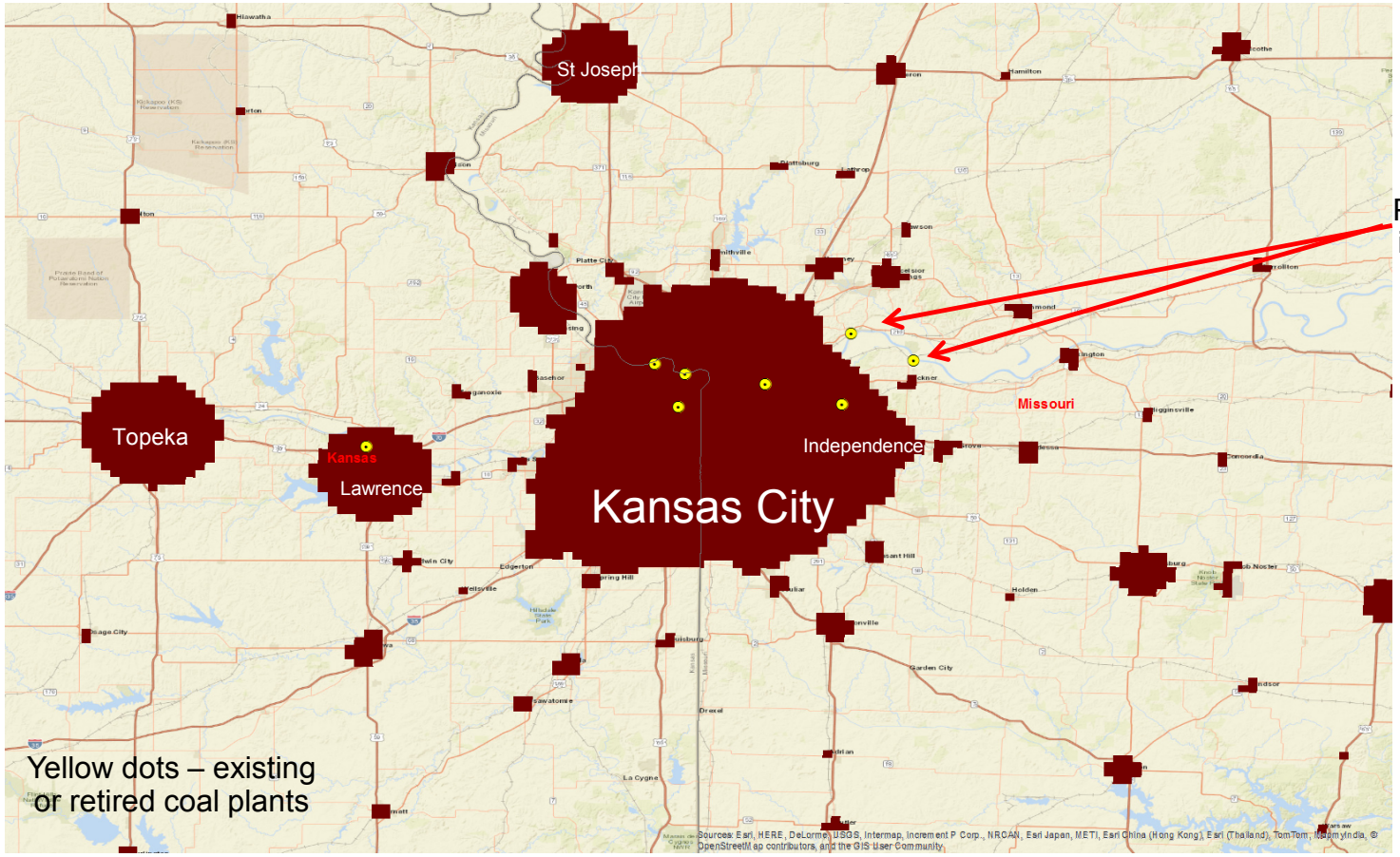
- Population density calculation made for circular area around the center of each database cell
  - Cells ≥ 500 ppsm (red color) are excluded
  - Cells< 500 ppsm (clear) are available
  - Eventually, a 100 m movement away from the population center will result in a transition from just above 500 ppsm to just below 500 ppsm

Population density (500 ppsm @ 20 miles)



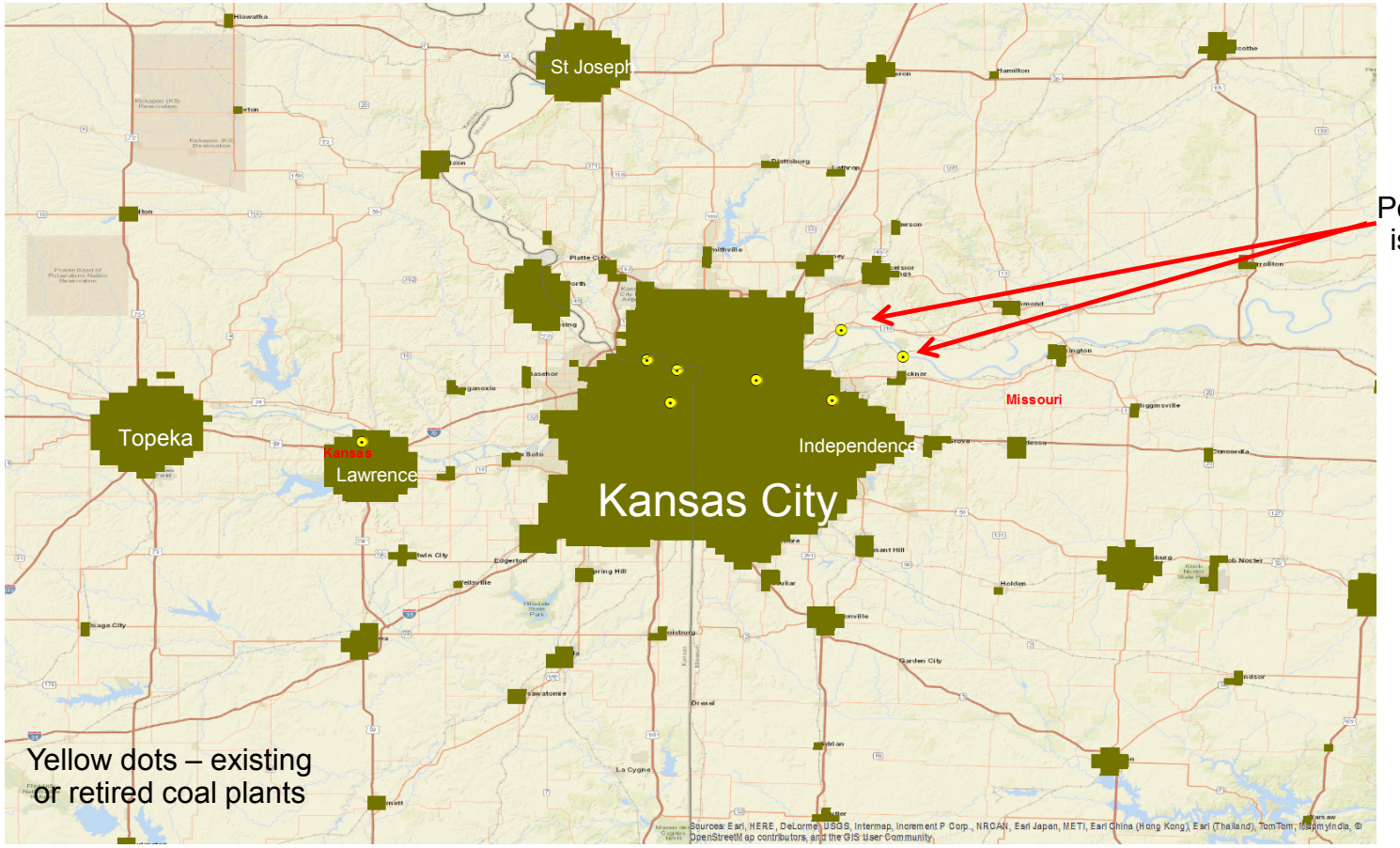
Population not an issue at 20 miles

Population density (500 ppsm @ 10 miles)



Population not an issue at 10 miles

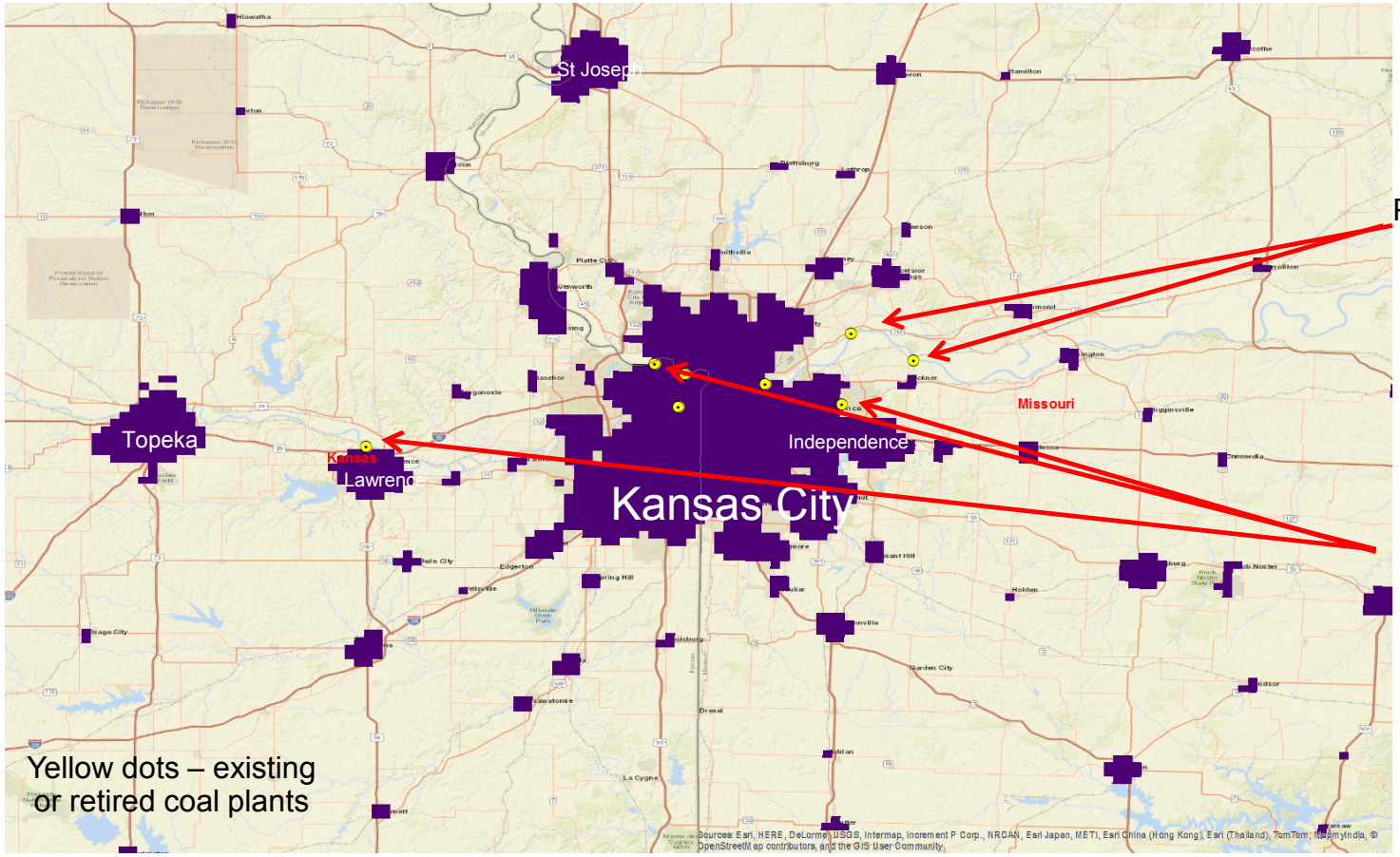
Population density (500 ppsm @ 5 miles)



Population not an issue at 5 miles

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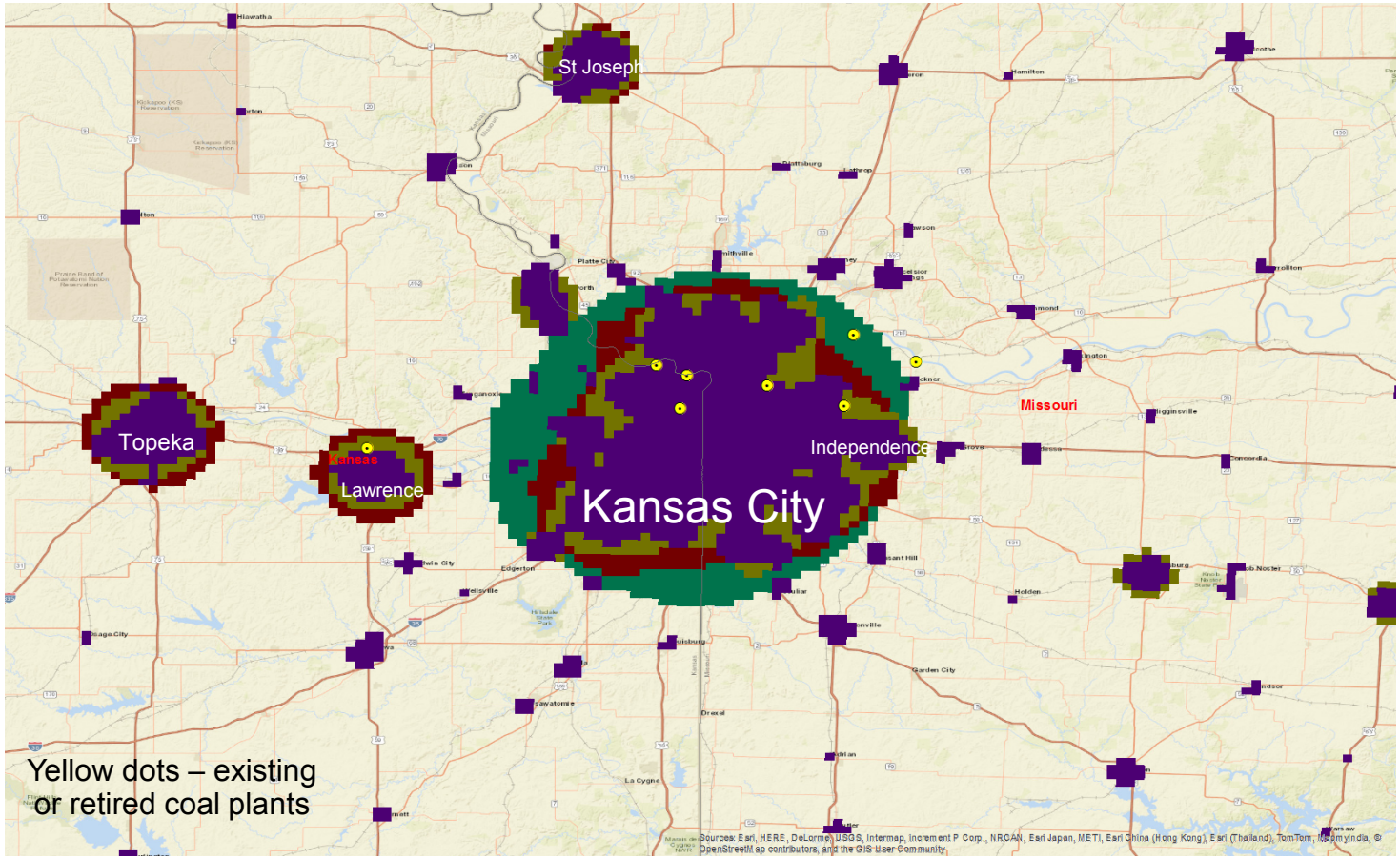
Population density (500 ppsm @ 2 miles)



Population not an issue at 2 miles

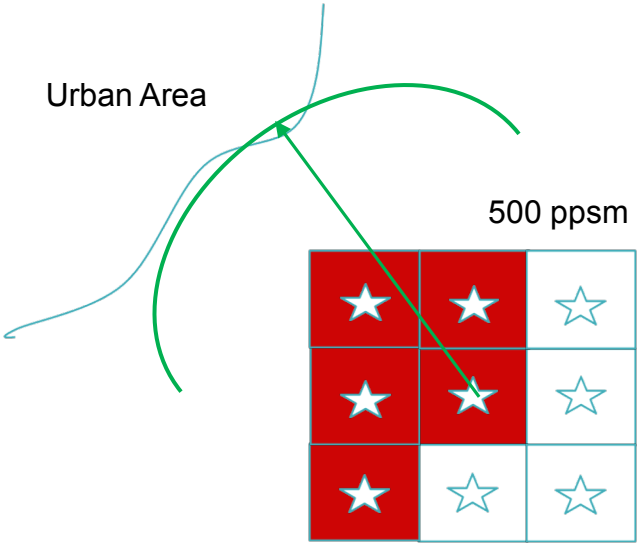
Population may not be an issue at 2 miles

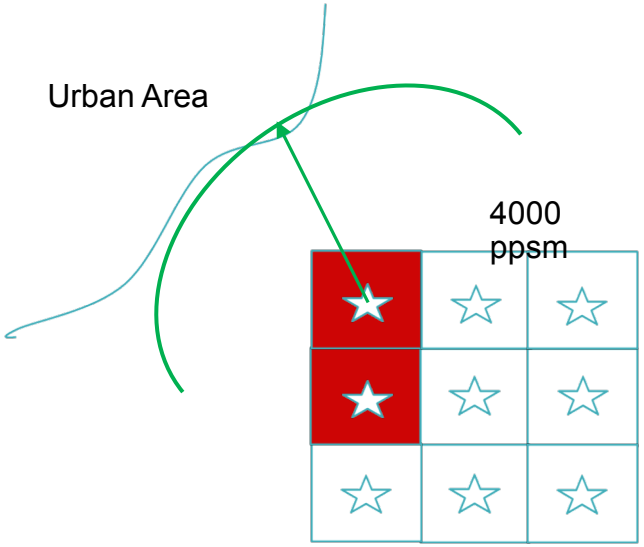
Density comparison (500 ppsm @ 2, 5, 10, & 20 miles)



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The effect of increasing the population density threshold





- Population density calculation made for circular area around the center of each database cell
  - Cells ≥ population density of interest (red color) are excluded
  - Cells< population density of interest (clear) are available
  - Shifting to a higher population density number effectively provides an opportunity for sites closer to an urban area to be considered

NRC asked ORNL to consider siting guidance alternatives for advanced reactors (SMRs, non-LWRs, and micro-reactors)

- Current siting guidance support in RG 4.7, Rev 3, General Site Suitability Criteria for Nuclear Power Stations, focused large LWRs
  - Population Density
  - Distance
- Some reactor vendors business plans include siting reactors close to an industrial partner or at a fossil plant back-fit site
  - Less remote
- Are there alternatives to using a formula distance as the primary population figure of merit for siting?

Brief history of siting regulations

- 1946 – Atomic Energy Commission formed from Atomic Energy Act
- 1948 – First Exclusion Area (EA) rule of thumb
  - $0.01 \times \sqrt{reactor\ thermal\ power\ (kw)}$
  - Reasonable for small reactors; impractical for larger reactors
  - A 1000 MWe (3000 MWt) reactor would require an EA of 17 miles
  - Assumed 50% of all fission products released in a cloud
  - No public evacuation assumed
  - Whole-body dose over accident duration limited to 300 rem
- 1954 – AEA amended to allow private industry to develop NPPs for electricity production

10 CFR 100, Reactor Site Criteria, established

- Between 1948 and 1962, the AEC licensed plants on a case-by-case basis and did not use formal criteria or metrics for siting
  - 1956 – Construction permit granted for Fermi 1
    - Between Detroit and Toledo
    - Concerns about population density dismissed
  - 1959; small PWR near Jamestown, NY rejected
    - Criticism had grown regarding siting NPPs close to population centers
    - Jamestown officials accused AEC of inconsistent siting standards
  - 1959-1962 AEC developed power reactor siting criteria leading to 10 CFR 100
    - An EA distance that is set by limiting the total whole-body dose of an individual to less than 25 rem or limiting the thyroid dose to less than 300 rem for an accident
    - Low population zone just outside the EA limited by the 25-rem whole-body/300-rem thyroid dose in consideration of any individual exposed over the entire time period of an accident
    - The population center, defined as >25,000 residents, should be 133% further away than evacuation distance

Development of power reactor siting guidance

- 1973 – AEC published WASH-1308, Population Distribution around NPP Sites
  - Acknowledges that engineered safety features (ESF) technology is available to make siting in densely populated areas feasible from the standpoint of meeting the individual dose guidelines in 10 CFR 100
  - ESF operating experience data not abundant
  - Off-normal and accident experience lacking
  - Only ESF feature considered is containment
  - Proposed specific regulatory guidance to limit population density out to 40 miles to approximately 400 ppsm; otherwise establish an EA distance of at least 0.4 miles and an LPZ of at least 2 miles
  - Policy is to limit societal risk by promoting remote siting but opens the door future consideration of additional ESFs

Development of RG 4.7 (Rev. 0), General Site Suitability Criteria for Nuclear Power Stations

- 1974 – RG 4.7 (Rev. 0) is published without specific population density guidance
  - ***Areas of low population density are preferred*** *for nuclear power station sites*
  - *Based on past experience, the Regulatory staff has found that a minimum* ***EA distance of 0.4 mile****, even with unfavorable design basis atmospheric dispersion characteristics, usually provides assurance that ESFs can be designed to bring the calculated dose from a postulated accident within the guidelines of 10 CFR 100*
    - *If the minimum exclusion distance is less than 0.4 mile, it may be necessary to place special conditions on station design (e.g.,* ***added engineered safety features****) before the site can be considered acceptable*
  - *Regulatory staff has found that a distance of* ***3 miles*** *to the outer boundary of the* ***LPZ*** *is usually adequate*

Development of RG 4.7 (Rev. 1)

- 1975 – RG 4.7 (Rev. 1) quickly revises Rev. 0 to include specific population density guidance
  - *If the population density, including weighted transient population, projected at the time of initial operation of a nuclear power station exceeds **500** persons per square mile averaged over any radial distance out to **30 miles** (cumulative population at a distance divided by the area at that distance), or the projected population density over the **lifetime** of the facility exceeds **1,000** persons per square mile averaged over any radial distance out to 30 miles, special attention should be given to the consideration of alternative sites with lower population densities.*

Development of RG 4.7 (Rev. 2)

- 1998 – RG 4.7 (Rev. 2) further refines specific population density guidance
  - *Preferably a reactor would be located so that, at the time of initial site approval and within about 5 years thereafter, the population density, including weighted transient population, averaged over* ***any radial distance out to 20 miles*** *(cumulative population at a distance divided by the circular area at that distance),* ***does not exceed 500 persons per square mile****. A reactor should not be located at a site whose population density is well in excess of the above value.*
- 2014 – RG 4.7 (Rev. 3) consistent with above statement
- Current siting guidance substitutes a combination of distance and population density limits as a surrogate for meeting potential radiological doses to individuals

Basis for Possible Change - SOARCA

- After September 11, 2001, applicants were required to evaluate and establish mitigating actions for the loss of large plant areas due to fires and explosions
  - NRC initiated State-of-the-Art-Reactor Consequence Analyses (SOARCA)
    - Improved evaluation of conservative source term assumptions established by NUREG-0771, NUREG-0772, and NUREG-0773
    - Enhanced modeling and simulation capabilities
    - Significant reduction in predicted release of cesium and iodine
- Beyond Design Basis Accidents can still lead to denial of land use over a large area
- Must look at advanced reactor attributes for transformational changes to siting guidance

Advanced reactor attributes compared to large LWRs (1)

SMRs

- Generally provide less than 300 MWe
- Operated individually or as part of a multi-module plant
- Near-term SMRs typically based on LWR technology with integrated components (SG, pressurizer, etc.) inside the reactor vessel
- Passive safety features and longer coping times
- More coolant water per kW
- Small source term

Non-LWRs

- A non-LWR may be defined as an SMR, or it may be a larger reactor
- Non-LWRs use a coolant other than water
- May use alternate fuel forms
- Particle-based fuel forms have higher melting temperatures and more robust barriers to radiation release
- LMRs and MSRs operate at low pressure and large margins to boiling (and essentially no driving force following an accident)
- Longer coping times

Advanced reactor attributes compared to large LWRs (2)

Micro-reactors

- Generally provide less than 25 MWe
- Factory fabricated
- Transportable by truck, shipping vessel, airplane, or rail car
- Self-regulating, not requiring many specialized operators, and utilizing passive safety systems to prevent any potential for overheating or reactor meltdown
- Small source term
- Very small site footprint



Additional bases for Possible Change

- Advanced reactors tend to preclude or severely mitigate accident categories by design
  - There is margin to consider changes in siting distances without compromising the overall defense-in-depth balance for public safety
- Passive safety systems tend to increase the reliability of an appropriate and successful accident response
  - Control of heat removal
  - Control of reactivity
  - Barrier approach to radionuclide retention
- Many advanced reactors considering below-grade siting
- NUREG-1537, non-power reactor siting approach
- Changes in public perception of risk

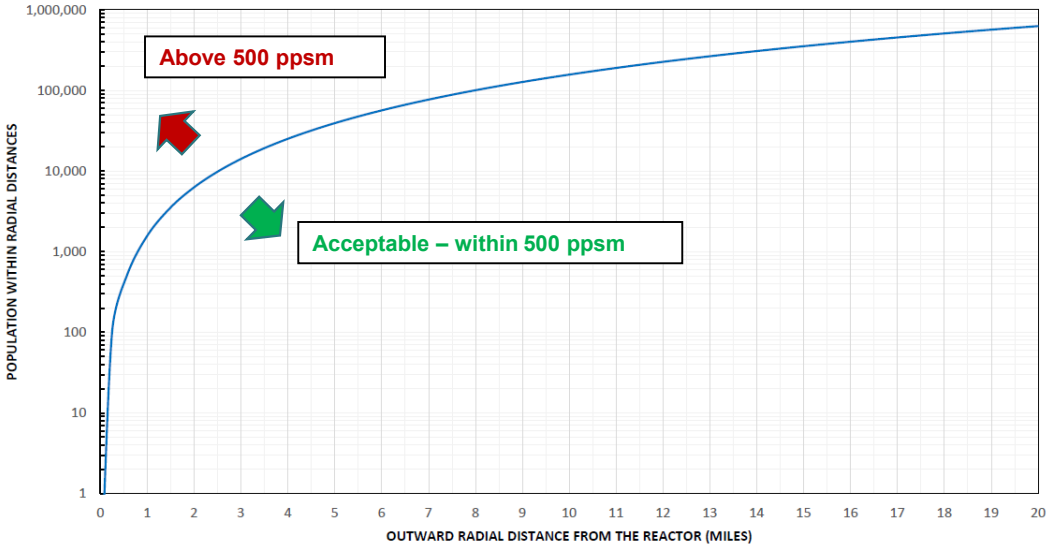
Given there are ample bases to consider advanced reactor alternatives to current large LWR siting guidance; the question is what can be measured?

- The first-principle elements for reasonable assurance of adequate protection of public health and safety are:
  - Source term
  - Time
  - Distance
  - Shielding
- Source term, hold-up time, and shielding are functions of design
  - Advanced reactors provide opportunities for improvements in all 3 areas
- However, **distance, integrated dose, and population density remain as the primary elements for siting guidance**

Currently accepted societal risk for siting based on population density and distance

RG 4.7 Societal Risk Curve

- 500 ppsm
- Evaluated out to 20 miles from the reactor
- RG 4.7 guidance implies that the 10 CFR 100 dose limits will be satisfied if the guidance limitations on population density in the plant vicinity are met



A. Costa, *Siting Considerations Related to Population for Small Modular and Non-Light Water Reactors*, NRC, presented at NRC Public Meeting on Regulatory Improvements for Advanced Reactors, December 14, 2017 (ML17354B219)

Source term, dose, and societal risk

- Many advanced reactors are expected to have one or more attributes—such as smaller cores, smaller source terms, passive safety features, enhanced release barriers, and reduced operating pressures—that will provide for smaller releases in the event of an accident
- Estimates of the dose to the public from an advanced reactor technology DBA requires analyses that are generally based on proprietary information
- However, if the dose to the public from an LWR DBA is assumed to be within limits specified in 10 CFR 100 using the conservative guidance from RG 4.7, then some societal risk ratios can be considered for advanced reactors

Calculating the basis for a societal risk ratio

- The area of a circle is given by $A = \pi r^2$, where A is the area and r is the radius of the circle
- The area swept out by a 20-mile circle is 1,256.6 square miles
- Multiplying the population density limitation of 500 ppsm for this area, an LWR should be limited to no closer than 20 miles to a population center of ~ 628,000 people
- An average dose per person (such as 25 rem) could be applied to this 20-mile population to establish a conservative societal risk value in person-rem
- However, a ratio of acceptable societal dose is the goal, so a person-rem value is not necessary

Assumptions for setting up a source term ratio

- A reasonable approximation for comparison is that at a fixed distance under identical meteorology conditions, radiation dose to the public is linear with the source term
- This means that if the radiological mixture is similar, then a reduction in source term that produces half the radiological isotopes would produce half the dose at a point in space
- Using this assumption, the cumulative accident dose to a population limited to a density of 500 ppsm out to a radius of 20 miles from an LWR that meets 10 CFR 100 limits can be ratioed and would be **equivalent to the cumulative advanced reactor accident dose received by an equivalent number of people over a smaller area**

Mathematical approach

- This relation can be seen in the following formula:

  $$SR = \pi r^2 \times D \times PPSM,$$

  where

  SR = accepted societal risk in terms of large LWRs (person-rem),

  r = assumed radius of consideration,

  D = assumed LWR dose to the public in the area of consideration, and

  PPSM = assumed population density in the area of consideration

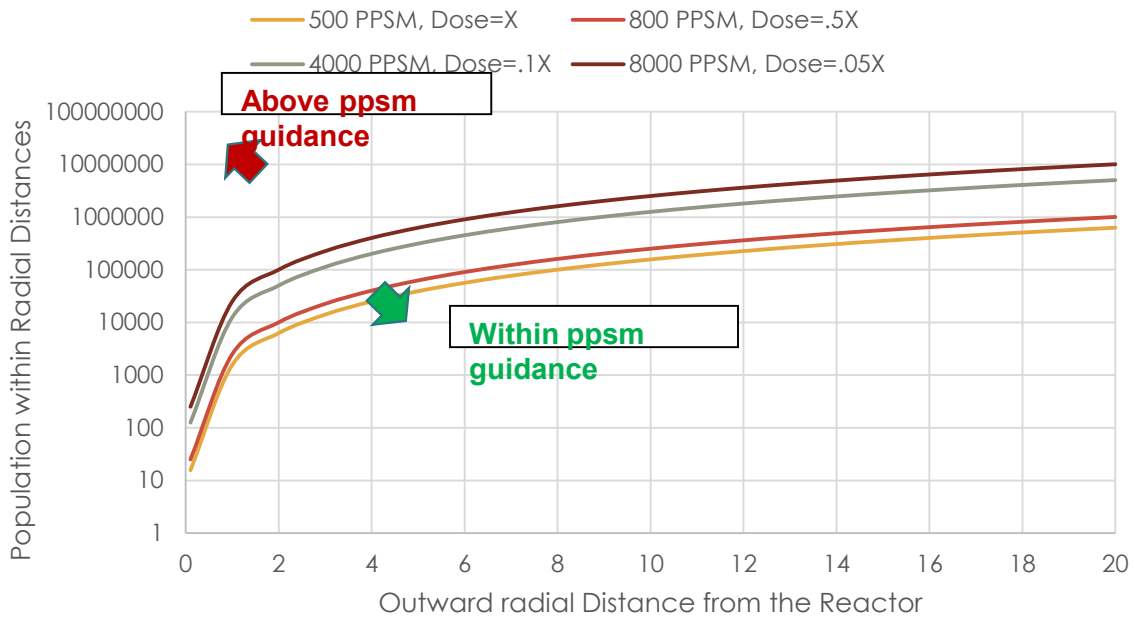
- A fraction of the LWR dose can be evaluated for an advanced reactor DBA compared to the conservative source term dose associated with an LWR DBA (for example the advanced reactor dose may = 0.1 D)

As an example, assume the advanced reactor source term is 1/10 the LWR source term

- A ratio of equivalent societal risks can lead to an impacted area due to the proximity to an advanced reactor that is a tenth the size of the large LWR equivalent
- Such an area relative to a circle with a 20-mile radius would be 125.7 square miles and would be equivalent to a circle with a radius of just 6.3 miles
- An uncertainty margin for the assumption of linear dose with source term, the location of the population relative to a radiation plume, and radiological mixture can be applied for added conservatism
- If the desired margin is 25%, then a circle with an area of 157.1 square miles (1.25 × 125.7) and a radius of 7.1 miles can be considered as a bounding comparable risk to the public

The equivalent societal risk curve can be recalculated

- Resulting population density of 4,000 ppsm (628,000/157.1) to be considered for siting a hypothetical advanced reactor with a tenth the source term of a large LWR plus additional uncertainty margin.



- Also shown are evaluations at 0.5 and 0.05 x the large LWR guidance curve plus 25% margin
- Maintains higher population density evaluation out to 20 miles from the reactor

The area holding 25,000 people at the given population densities of interest can be solved for the minimum 10 CFR 100 LPZ distance

25,000 people population center		
PPSM	Distance (miles)	LPZ (miles)
500	4.0	3.0
800	3.2	2.4
4,000	1.4	1.1
8,000	1.0	0.7

- Using 4,000 ppsm (1/10 source term ratio), the LPZ is reduced from 3 miles to 1.1 miles
- At this distance, the LPZ is likely approaching a reasonable EA boundary for a utility with an advanced reactor

Identified siting alternatives

- Using higher population density based on an advanced reactor bounding ratio out to 20 miles will provide closer siting opportunities on the edges of large population centers but still forces consideration for standoff siting from areas of very high population density
- An alternative would be to continue using 500 ppsm limited to the smaller evaluated area
  - 1/10 source term ratio would be 6.3 miles
  - 1/10 source term ratio plus 25% margin would be 7.1 miles

Summary

- Analyses of many advanced reactor designs and attributes are expected to show that reactor accidents will
  - Result in a significantly reduced source term and/or
  - Limit any radioactive material fallout to within the site boundary or be limited to within a short distance of the reactor boundary
- Therefore, proposing a fixed reduction in the source term, such as an order of magnitude, for advanced reactors seems achievable.
- Such an approach would require each advanced reactor technology to demonstrate some minimum reduction in the source term to permit more liberal bounding guidance in RG 4.7 for advanced reactor siting
- Another option is to perform individual analyses for each technology, but this risks the original consistency argument

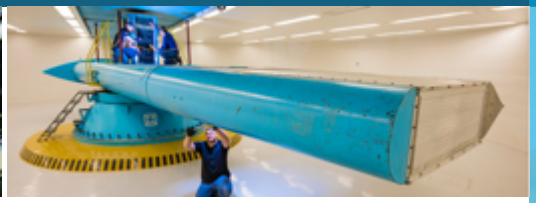
Sandia National Laboratories

Non-Light Water Reactor Mechanistic Source Term Study









PRESENTED BY

Matt Dennis, Dave Luxat, Andrew J. Clark, and Zac Jankovsky

Sandia National Laboratories is a multimission laboratory managed and operated by National Technology & Engineering Solutions of Sandia, LLC, a wholly owned subsidiary of Honeywell International Inc., for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-NA0003525.

Outline

Scope and limitations of the analysis

Scenario selection and examples

Source term factors

Source term factor value assignment

Scenarios and results

- HTGR
- MSR
- SFR

High level conclusions

Scope and Limitations

Goal of the analysis

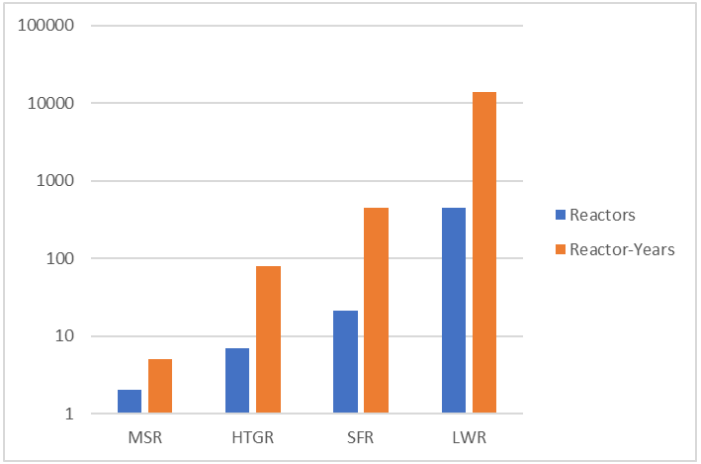
- Identify accident scenarios that could lead to off-site radiological release for non-LWR design concepts.
- Generalized evaluation of fission product retention barrier role in limiting off-site releases.
- Qualitative (and semi-quantitative) impact of non-LWR source term characteristics on off-site consequences.

Overview of presentation scope

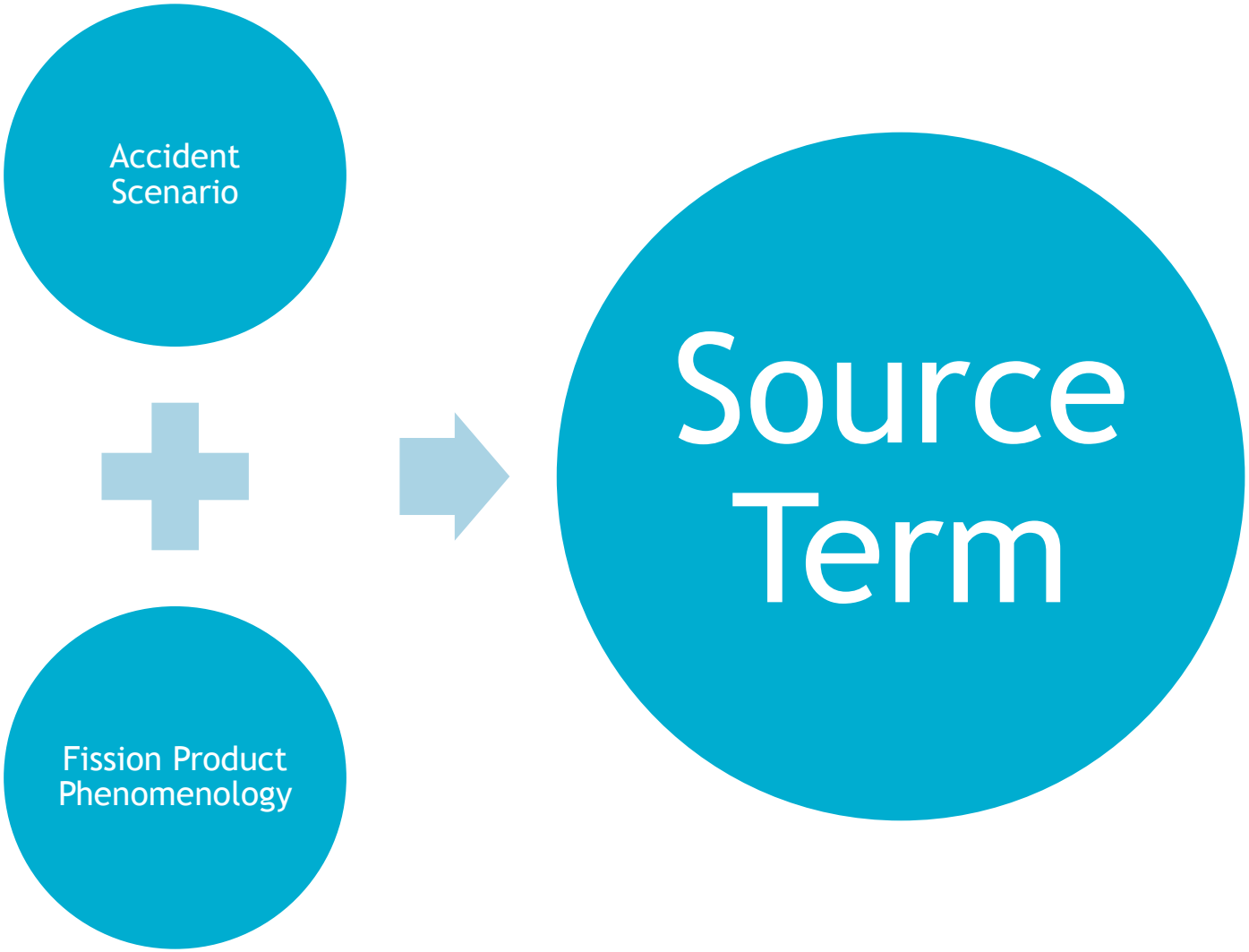
- Representative non-LWR concepts
  - Liquid-fuel, fluoride salt MSR; e.g., Molten Salt Reactor Experiment (MSRE)
  - Helium-cooled, TRISO-fueled, pebble bed HTGR; e.g., AVR reactor in Germany
  - Pool-type, metal-fueled SFR; e.g., EBR-II
- Evaluation of accident scenarios contributing to source terms
- Identify qualitative features of non-LWR source terms

Limitations

- Current state of knowledge for each reactor type
- Diversity of designs
  - Current advanced reactor designs differ widely and so will the accident insights
  - E.g., HTGR pebble bed vs prismatic core designs.



Overall Considerations in Mechanistic Source Term Determination



Scenario Selection and Examples

Scenarios selected to assess characteristics of design basis and beyond design basis source terms across three advanced reactor concepts.

Scenarios capture classes of accident scenarios based on initiating events, reactivity control, heat removal, and containment failures.

- Reactivity control failures
  - Accidents tend to progress rapidly
- Heat removal failures
  - Failures can occur when reactor is critical or in a subcritical state
  - Failures in critical state tend to progress rapidly and lead to early source terms
  - Timing largely influenced by coolant heat capacity when failure occurs after reactor is subcritical
- Containment failures
  - Compromise of a barrier may influence timing and severity

Examples

- MSR primary coolant leak (heat removal, containment)
- HTGR break in helium pressure boundary (heat removal, containment)
- SFR transient overpower (reactivity control)

Source Term Characteristics

Source terms developed to capture potential human health impact and effect on off-site population

Type and magnitude of radionuclide release necessary to assess immediate or long-term health effects from individual exposure

Timing of radionuclide release impacts extent to which off-site population is affected (i.e., whether mitigating measures can be taken to reduce potential for exposure)

Source term evolution characterized in terms of

- Small or large release magnitudes for noble gases, halogens and volatile classes of radionuclides
- Early, late or long-term releases to the environment are highly dependent on reactor type and initiating event.

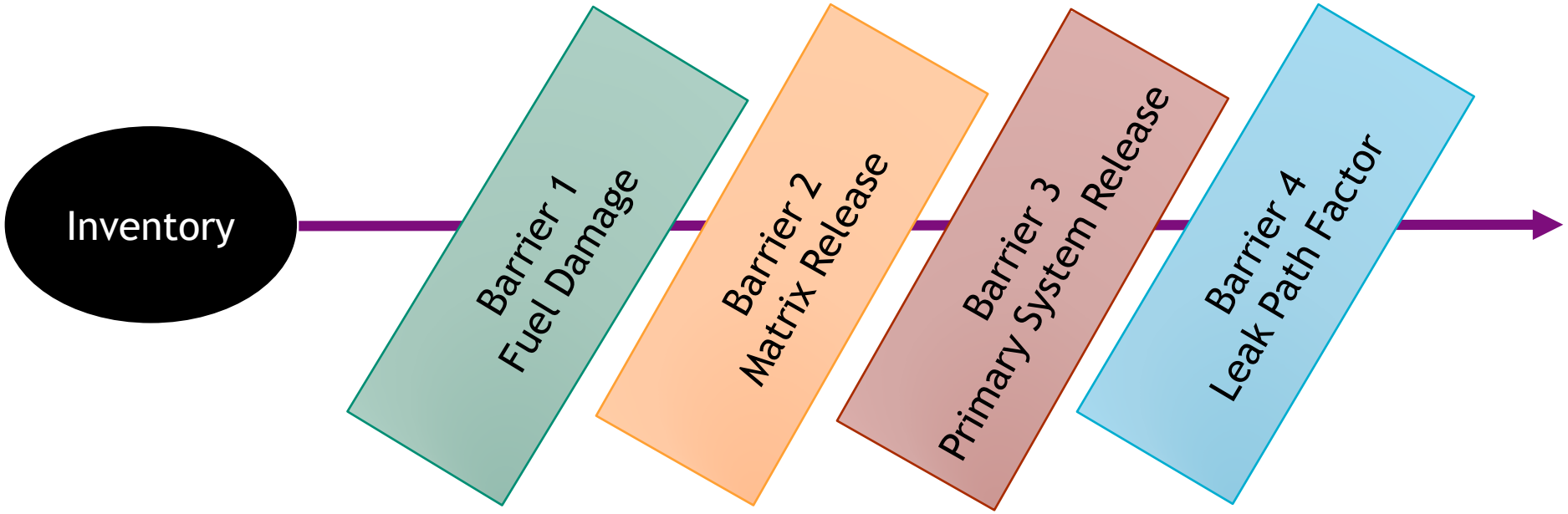
Source Term Evolution – Framework for Evaluation

When scoping a mechanistic source term (MST) calculation, the designer must consider the barriers that prevent the release of radionuclides.

DOE-STD-3009-2014 and DOE-HDBK-3010-94 provide standard approaches and guidance for performing source term calculations.

- The approach considers the barriers that prevent the release of radioactivity to the environment.

For the assessment of many different reactor coolants and designs, the DOE approach to source term analysis is adopted.



DOE Source Term Calculations – Barrier Approach

The source term is characterized by:

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

Where,

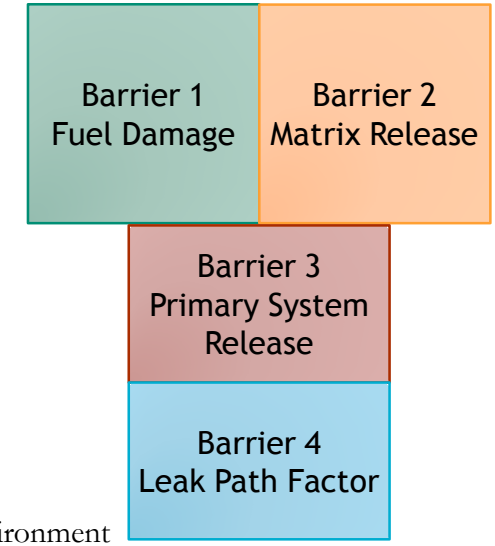
- $ST(S_i, RN_j, t)$ - Source term as a function of the accident scenario (S_i), radionuclide class (RN_j), and timing (t)
- $I(RN_j)$ - Inventory
- $F(S_i, t)$ - Fuel damage
- $MR(S_i, RN_j, t)$ - Matrix release
- $PSR(S_i, RN_j, t)$ - Primary system release
- $LPF(S_i, RN_j, t)$ - Leak path factor

Each of these parameters, and their functional dependencies, are evaluated for each reactor and the given scenarios.

Source Term Factors

Phenomena relating to retention of radionuclides

- Retention in solid fuel
  - Fission products can remain within the fuel matrix
  - Gaseous fission products retained in the fuel by cladding preventing release into the coolant
- Retention in molten fuel
  - Molten fuel systems combine the fuel and working fluid (coolant) responsible for carrying away heat
- Retention in liquid working fluids
  - Fission products leave liquids through transport to surfaces and release into interfacing gaseous environments
- Transport in gaseous environments
  - Gas flows can carry fission products into other regions of the plant or ultimately into the environment
- Fission product removal from gas volumes or flows
  - Fission products can deposit on surfaces (e.g., containment walls) and be removed from gas volumes or gaseous flows
  - Gravitational settling, inertial impaction on surfaces, phoretic mechanisms, turbulent deposition

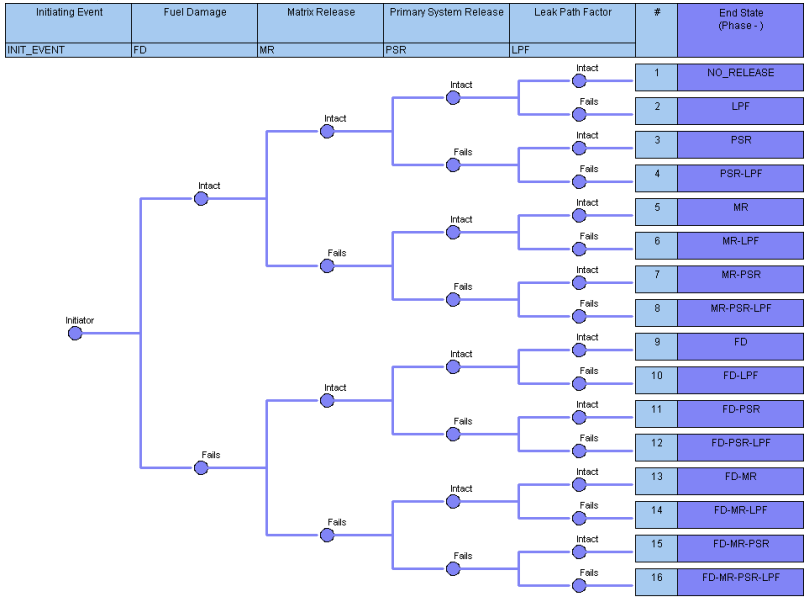


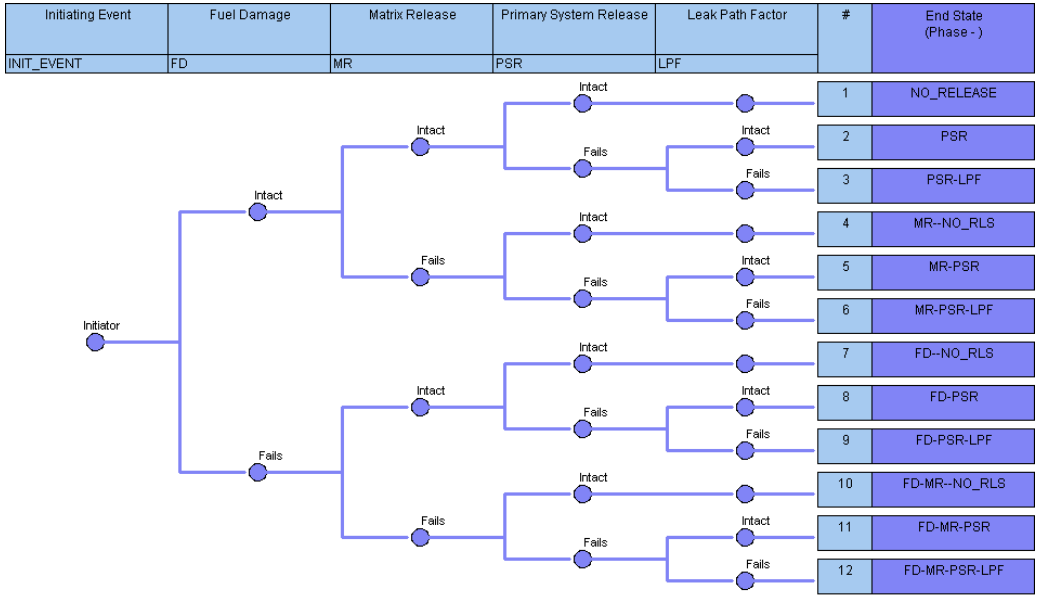
These all represent barriers to release of fission products directly into environment

Progressive Source Term Evolution

Barrier approach to MST

- Consider challenges to barriers from inventory to environmental release
- From an event tree perspective, the barrier approach to MST calculations can be modeled as shown below.
  - Naïve approach shows 16 end-states (2^n relationship, where $n=4$ represents the number of barriers)
  - The number of end-states can be reduced based on assumptions of the design.
    - HTGR example: some residual amount of radionuclides escape fuel and matrix during normal operation, however, as long as the primary system remains intact, RNs should not escape out of the primary system. As such, 16 end states can be reduced to 12 or fewer end states.
    - This reduction in end-states is important in limiting the consequence analysis to only release end-states.







HTGR Scenarios

High-Temperature Gas Reactors Worldwide

First the Dragon Reactor 20 MW(t) in the UK (criticality on August 23, 1964).

- 2MPa system pressure, 350/750 °C inlet/outlet temperatures

Closely followed by Peach Bottom, 115 MW(t).

- 2.4 MPa, 350/750 °C inlet/outlet temperatures

AVR, 46MW(t) first prototype pebble bed reactor (Germany).

- 100,000 6 cm diameter spheres.
- 1.1 MPa, 270/950 °C inlet/outlet temperatures

FSV HTGR, 842MW(t) – first reactor to use stacked columns of prismatic fuel elements.

- 4.8 MPa, 405/775 °C inlet/outlet temperatures

Modular HTGR (mHTGR) – Fort St. Vrain –

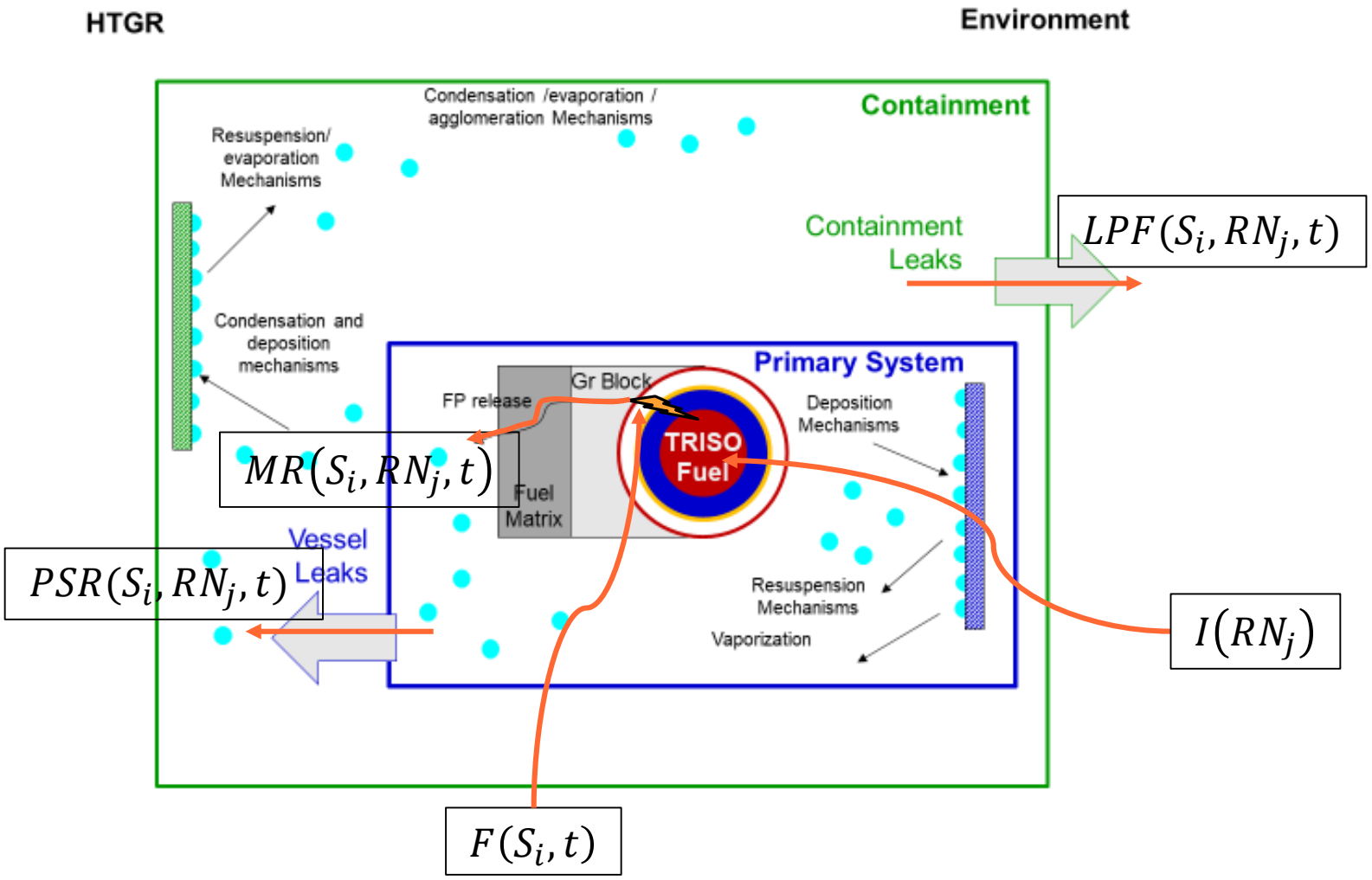
- General Atomics Design for 2000-4000 MW(t), where each module rated for 350 MW(t)

NGNP conceptual design.

DOE Source Term Calculations – Five Factor Formula

The source term is characterized by:

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



HTGR – Inventory

The source term is characterized by:

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

Initial inventory

The initial inventory is referenced from NGNP Source Term Report for a 250MW(t) pebble bed reactor with a reactor outlet temperature of 700°C.

Class Name	Representative Nuclide	Initial Inventory (kg)
Noble Gases	Xe	2.84E-01
Alkali Metals	Cs	8.73E+00
Alkaline Earths	Ba	5.12E+00
Halogens	I	8.03E-02
Chalcogens	Te	3.66E-02
Platiniods	Ru	4.64E-01
Early Transition Elements	Mo	0.00E+00
Tetravalent	Ce	3.43E+01
Trivalents	La	2.45E-02
Uranium	U	0.00E+00
More Volatile Main Group	Cd	9.44E-02
Less Volatile Main Group	Ag	1.03E-02

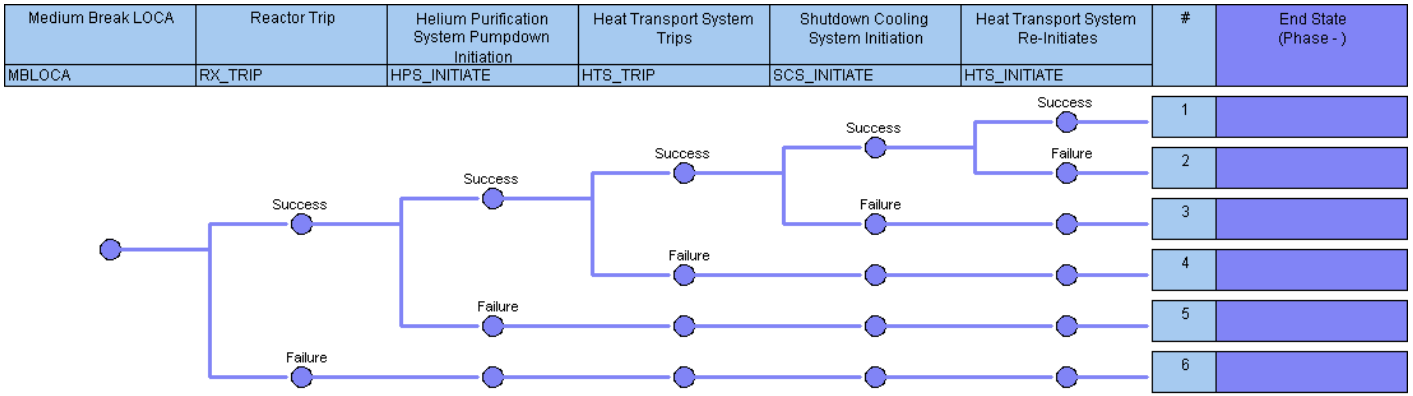
*Note that initial inventory for the HTGR reactor does not consider early transition elements and uranium RN classes. These classes should be accounted for in follow-on work.

→mHTGR (350MW(t)) estimated an inventory of 2.0E09 Ci

→ NGNP (250MW(t)) estimated an inventory of 3.0E08 Ci

HTGR Scenario – Break in Helium Pressure Boundary

1. Break in helium pressure boundary (HPB) – Early release due to depressurization. Long term release due to core heatup from the fuel and then from HPB.
   - Leak or break in the HPB piping
   - Reactor trip
   - Loss of heat transport to steam generator
   - Loss of shutdown cooling
   - Immediate depressurization of helium in the HPB
   - Opening of the reactor building vent to relieve helium pressure

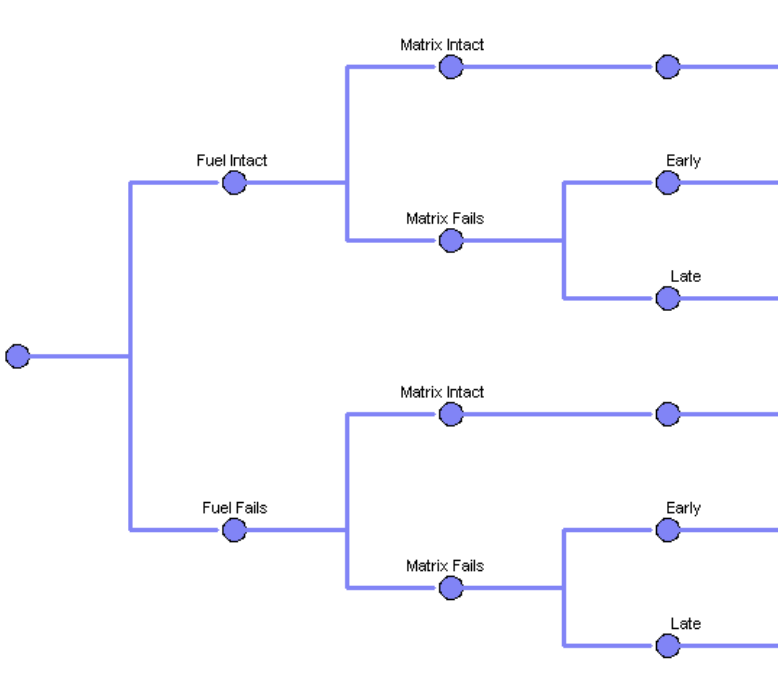


The mHTGR PRA assessed this accident sequence. The systems that respond during a LOCA accident are represented in the event tree.

NGNP Used a Similar Barrier Approach

The NGNP Safety Basis and Approach document used a similar approach here. The barriers considered by the NGNP study are:

- The fuel particle kernel → **Fuel Damage**
- The fuel particle coatings (silicon carbide and pyrocarbon coating) → **MR**
- The core graphite and carbonaceous materials → **MR**
- The helium pressure boundary → **PSR**
- The reactor building → **LPF**



Medium Break LOCA	TRISO Fuel Incremental Failures	RN Matrix Release from TRISO Fuel	TRISO Timing	Primary System Release	Leak Path Factor	#	End State (Phase -)	Comments (Phase -)
MBLOCA	FUEL	MATRIX	TRISO_TIMING	PSR	LPF			
						1	OK	Core OK - Normal Ops RLS
						2	OK	Core OK - Cntmnt Fails
						3	MR_EARLY-PSR	MR Early - PSR - Cntmnt OK
						4	MR_EARLY-PSR-LPF	MR Early - PSR - Cntmnt Fails
						5	MR_LATE-PSR	MR Late - PSR - Cntmnt OK
						6	MR_LATE-PSR-LPF	MR Late - PSR - Cntmnt Fails
						7	OK	FD - Very Small PSR
						8	OK	FD - Very Small PSR - Cntmnt Fails
						9	FD-MR_EARLY-PSR	FD - MR Early - PSR - Cntmnt OK
						10	FD-MR_EARLY-PSR-LPF	FD - MR Early - PSR - Cntmnt Fails
						11	FD-MR_LATE-PSR	FD - MR Late - PSR - Cntmnt OK
						12	FD-MR_LATE-PSR-LPF	FD - MR Late - PSR - Cntmnt Fails

HTGR Scenario – Break in Helium Pressure Boundary

Medium Break LOCA	TRISO Fuel Incremental Failures	RN Matrix Release from TRISO Fuel	TRISO Timing	Primary System Release	Leak Path Factor	#	End State (Phase -)	Comments (Phase -)
MBLOCA	FUEL	MATRIX	TRISO_TIMING	PSR	LPF			
					Containment Holdup	1	OK	Core OK - Normal Ops RLS
		Matrix Intact		LOCA	Containment Leak	2	OK	Core OK - Cntmnt Fails
	Fuel Intact		Early	LOCA	Containment Holdup	3	MR_EARLY-PSR	MR Early - PSR - Cntmnt OK
		Matrix Fails			Containment Leak	4	MR_EARLY-PSR-LPF	MR Early - PSR - Cntmnt Fails
			Late	LOCA	Containment Holdup	5	MR_LATE-PSR	MR Late - PSR - Cntmnt OK
					Containment Leak	6	MR_LATE-PSR-LPF	MR Late - PSR - Cntmnt Fails
		Matrix Intact		LOCA	Containment Holdup	7	OK	FD - Very Small PSR
					Containment Leak	8	OK	FD - Very Small PSR - Cntmnt Fails
	Fuel Fails		Early	LOCA	Containment Holdup	9	FD-MR_EARLY-PSR	FD - MR Early - PSR - Cntmnt OK
		Matrix Fails			Containment Leak	10	FD-MR_EARLY-PSR-LPF	FD - MR Early - PSR - Cntmnt Fails
			Late	LOCA	Containment Holdup	11	FD-MR_LATE-PSR	FD - MR Late - PSR - Cntmnt OK
					Containment Leak	12	FD-MR_LATE-PSR-LPF	FD - MR Late - PSR - Cntmnt Fails

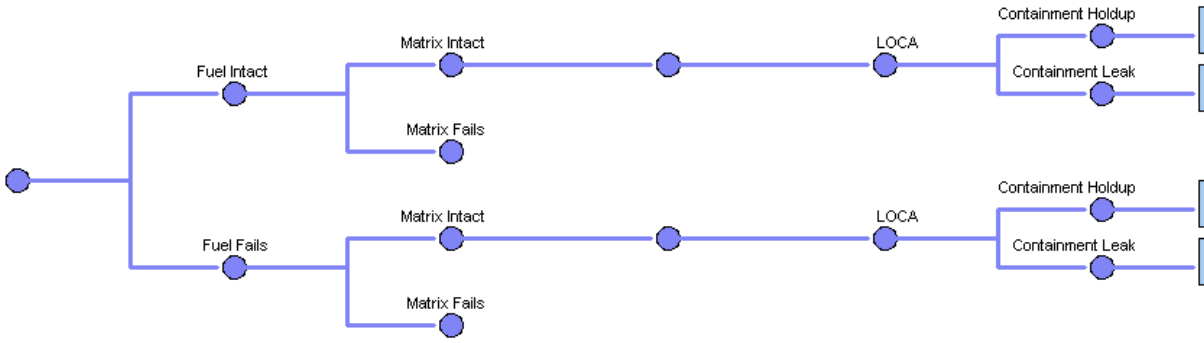
Using the DOE source term approach, an event tree can be constructed to represent the barriers that prevent the release of radionuclides. Note that the event tree branching is dependent on the engineered safety features of the plant.

When constructing an event tree for the break in HPB, it was found that timing considerations were only important for fuel and matrix releases.

- Because the PSR parameter is directly related to the LOCA scenario, the release is assumed to occur
- Leak path factor will be dependent on containment response to the accident

HTGR MBLOCA in HPB – Negligible to Minimal Release Pathways

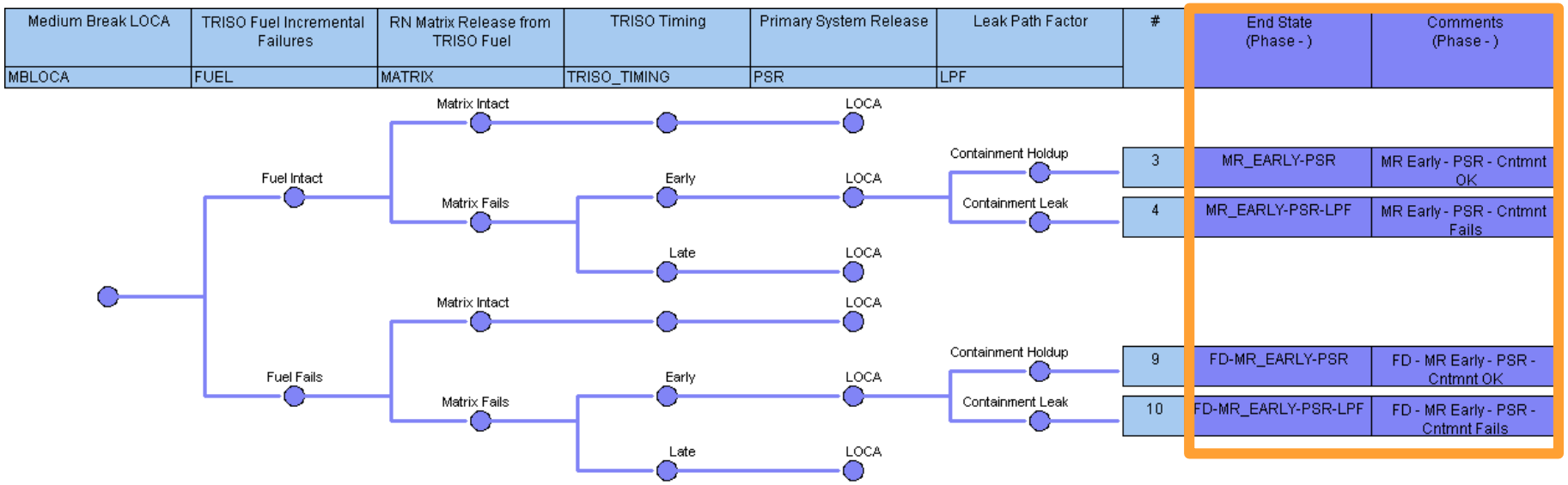
Medium Break LOCA	TRISO Fuel Incremental Failures	RN Matrix Release from TRISO Fuel	TRISO Timing	Primary System Release	Leak Path Factor	#	End State (Phase -)	Comments (Phase -)
MBLOCA	FUEL	MATRIX	TRISO_TIMING	PSR	LPF			
						1	OK	Core OK - Normal Ops RLS
						2	OK	Core OK - Cntmnt Fails
						7	OK	FD - Very Small PSR
						8	OK	FD - Very Small PSR - Cntmnt Fails



Assume that if the matrix, which is comprised of the pellet buffer layers (pyrolytic carbon and silicon carbide layers) and graphite fuel matrix, remains intact that RN release from TRISO is significant reduced.

- End State #1
  - Fuel and matrix remain intact.
  - RNs from normal operation are released from primary system into containment.
  - Containment leakage is minimal
  - ➢ RN release is nearly negligible.
- End State #2
  - Fuel and matrix remain intact.
  - RNs from normal operation are released from primary system into containment.
  - Containment leakage increases.
  - ➢ RN release is minimal.
- End State #7
  - Fuel fails but matrix remains intact → matrix is able to withhold a significant amount of RNs.
  - RNs from normal operation are released from primary system into containment.
  - Containment leakage is minimal
  - ➢ RN release is nearly negligible.
- End State #8
  - Fuel fails but matrix remains intact → matrix is able to withhold a significant amount of RNs.
  - RNs from normal operation are released from primary system into containment.
  - Containment leakage increases.
  - ➢ RN release is minimal.

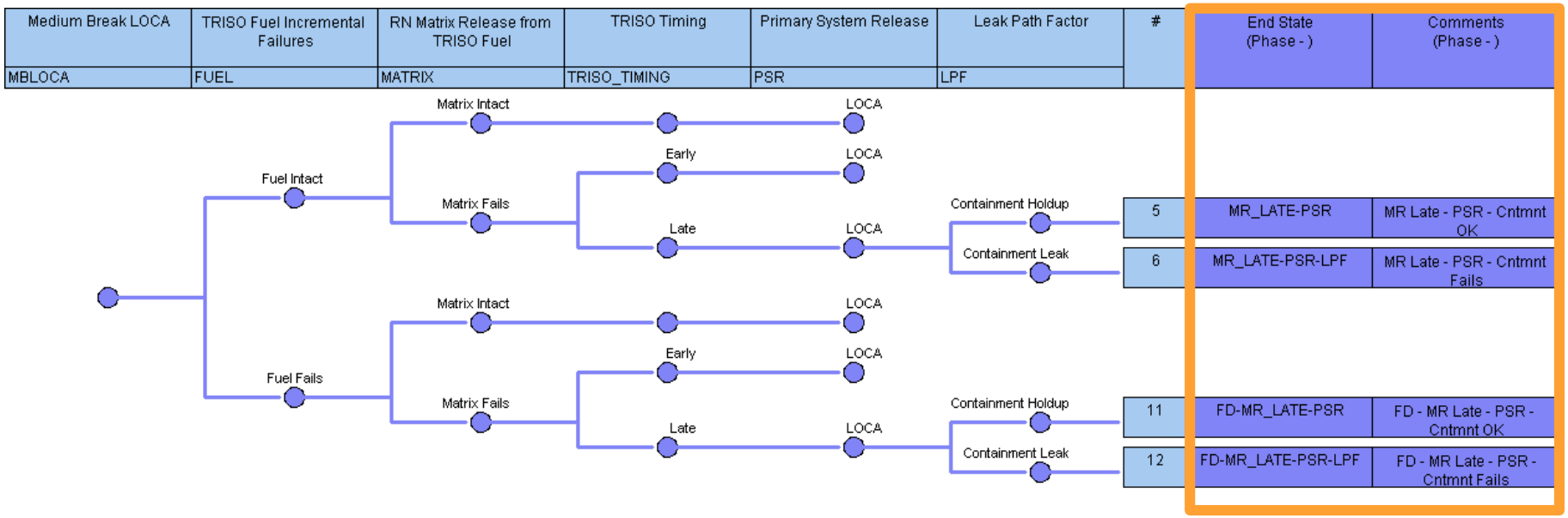
HTGR MBLOCA in HPB – Early Release Pathways



If the matrix fails, a significant release of RNs from the TRISO fuel may occur. The following end states may lead to early releases.

- End State #3
  - Fuel does not experience significant additional failure but the matrix fails early.
  - RNs from normal operation are released from primary system into containment.
  - Containment leakage is minimal
  - ➢ RN release is nearly negligible and early.
- End State #4
  - Fuel does not experience significant additional failure but the matrix fails early.
  - RNs from normal operation are released from primary system into containment.
  - Containment leakage increases.
  - ➢ RN release is minimal and early.
- End State #9
  - Fuel and matrix experience significant early failures.
  - Significant quantities of RNs are released from primary system into containment.
  - Containment leakage is minimal
  - ➢ RN release is moderate and early.
- End State #10
  - Fuel and matrix experience significant early failures.
  - Significant quantities of RNs are released from primary system into containment.
  - Containment leakage increases.
  - ➢ RN release is significant and early.

HTGR MBLOCA in HPB – Late Release Pathways

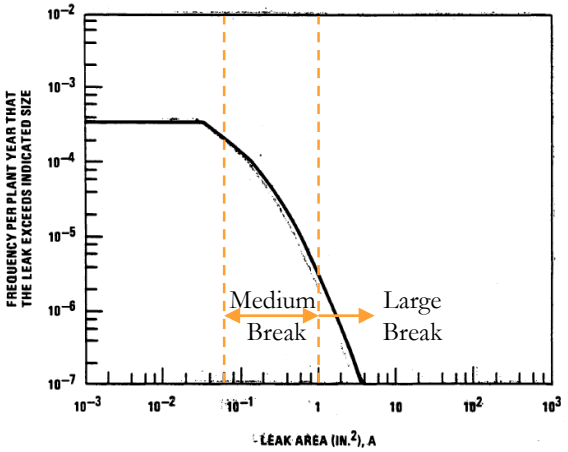


If the matrix fails, a significant release of RNs from the TRISO fuel may occur. The following end states may lead to late releases.

- End State #5
  - Fuel does not experience significant additional failure but the matrix fails late.
  - RNs from normal operation are released from primary system into containment.
  - Containment leakage is minimal
  - ➢ RN release is nearly negligible and late.
- End State #6
  - Fuel does not experience significant additional failure but the matrix fails late.
  - RNs from normal operation are released from primary system into containment.
  - Containment leakage increases.
  - ➢ RN release is minimal and late.
- End State #11
  - Fuel and matrix experience significant late failures.
  - Significant quantities of RNs are released from primary system into containment.
  - Containment leakage is minimal
  - ➢ RN release is moderate and late.
- End State #12
  - Fuel and matrix experience significant late failures.
  - Significant quantities of RNs are released from primary system into containment.
  - Containment leakage increases.
  - ➢ RN release is significant and late.

HTGR Scenario – Break in Helium Pressure Boundary

Medium Break LOCA	TRISO Fuel Incremental Failures	RN Matrix Release from TRISO Fuel	TRISO Timing	Primary System Release	Leak Path Factor	#	End State (Phase -)	Comments (Phase -)
MBLOCA	FUEL	MATRIX	TRISO_TIMING	PSR	LPF			
	Fuel Intact	Matrix Intact		LOCA	Containment Holdup	1	OK	Core OK - Normal Ops RLS
					Containment Leak	2	OK	Core OK - Cntmnt Fails
		Matrix Fails	Early	LOCA	Containment Holdup	3	MR_EARLY-PSR	MR Early - PSR - Cntmnt OK
					Containment Leak	4	MR_EARLY-PSR-LPF	MR Early - PSR - Cntmnt Fails
			Late	LOCA	Containment Holdup	5	MR_LATE-PSR	MR Late - PSR - Cntmnt OK
					Containment Leak	6	MR_LATE-PSR-LPF	MR Late - PSR - Cntmnt Fails
	Fuel Fails	Matrix Intact		LOCA	Containment Holdup	7	OK	FD - Very Small PSR
					Containment Leak	8	OK	FD - Very Small PSR - Cntmnt Fails
		Matrix Fails	Early	LOCA	Containment Holdup	9	FD-MR_EARLY-PSR	FD - MR Early - PSR - Cntmnt OK
					Containment Leak	10	FD-MR_EARLY-PSR-LPF	FD - MR Early - PSR - Cntmnt Fails
			Late	LOCA	Containment Holdup	11	FD-MR_LATE-PSR	FD - MR Late - PSR - Cntmnt OK
					Containment Leak	12	FD-MR_LATE-PSR-LPF	FD - MR Late - PSR - Cntmnt Fails

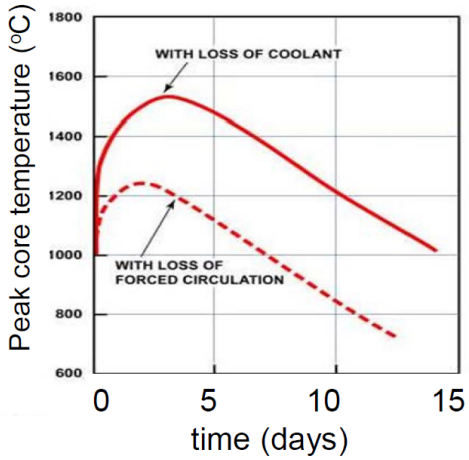


Frequency of primary coolant system leaks as a function of leak area [mHTGR PRA, Vol. 1, Appendix A].

- For LOCA initiating events, only MBLOCA and LBLOCA are considered.

HTGR Scenario – Break in Helium Pressure Boundary

Medium Break LOCA	TRISO Fuel Incremental Failures	RN Matrix Release from TRISO Fuel	TRISO Timing	Primary System Release	Leak Path Factor	#	End State (Phase -)	Comments (Phase -)
MBLOCA	FUEL	MATRIX	TRISO_TIMING	PSR	LPF			
	Fuel Intact	Matrix Intact		LOCA	Containment Holdup	1	OK	Core OK - Normal Ops RLS
					Containment Leak	2	OK	Core OK - Cntmnt Fails
		Matrix Fails	Early	LOCA	Containment Holdup	3	MR_EARLY-PSR	MR Early - PSR - Cntmnt OK
					Containment Leak	4	MR_EARLY-PSR-LPF	MR Early - PSR - Cntmnt Fails
			Late	LOCA	Containment Holdup	5	MR_LATE-PSR	MR Late - PSR - Cntmnt OK
					Containment Leak	6	MR_LATE-PSR-LPF	MR Late - PSR - Cntmnt Fails
	Fuel Fails	Matrix Intact		LOCA	Containment Holdup	7	OK	FD - Very Small PSR
					Containment Leak	8	OK	FD - Very Small PSR - Cntmnt Fails
		Matrix Fails	Early	LOCA	Containment Holdup	9	FD-MR_EARLY-PSR	FD - MR Early - PSR - Cntmnt OK
					Containment Leak	10	FD-MR_EARLY-PSR-LPF	FD - MR Early - PSR - Cntmnt Fails
			Late	LOCA	Containment Holdup	11	FD-MR_LATE-PSR	FD - MR Late - PSR - Cntmnt OK
					Containment Leak	12	FD-MR_LATE-PSR-LPF	FD - MR Late - PSR - Cntmnt Fails



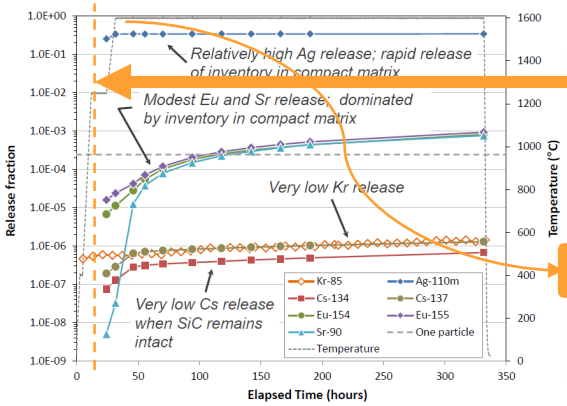
Condition	Design specifications for TRISO failure fraction		AGR-1 95% confidence
	NGNP[a]	HTR-Modul[a]	
In-service	$\leq 2.0\times10^{-4}$	$\leq 1.6\times10^{-4}$	$\leq 1.1\times10^{-5}$
1600°C accident	$\leq 6.0\times10^{-4}$	$\leq 6.6\times10^{-4}$	$\leq 6.6\times10^{-5}$ [b]

[a] Values for NGNP and HTR-Modul taken from NGNP Fuel Qualification White Paper, INL/EXT-10-18610, Idaho National Laboratory, 2010.
[b] Value obtained by combining statistics from 1600 and 1700°C AGR-1 safety tests (0 failures out of 45,550 particles)

According to design specifications and test data, differences in the 95% confidence intervals between in-service and elevated fuel temperatures failures are minimal.

HTGR Scenario – Break in Helium Pressure Boundary

Medium Break LOCA	TRISO Fuel Incremental Failures	RN Matrix Release from TRISO Fuel	TRISO Timing	Primary System Release	Leak Path Factor	#	End State (Phase -)	Comments (Phase -)
MBLOCA	FUEL	MATRIX	TRISO_TIMING	PSR	LPF			
					Containment Holdup	1	OK	Core OK - Normal Ops RLS
		Matrix Intact		LOCA	Containment Leak	2	OK	Core OK - Cntmnt Fails
	Fuel Intact		Early	LOCA	Containment Holdup	3	MR_EARLY-PSR	MR Early - PSR - Cntmnt OK
		Matrix Fails			Containment Leak	4	MR_EARLY-PSR-LPF	MR Early - PSR - Cntmnt Fails
			Late	LOCA	Containment Holdup	5	MR_LATE-PSR	MR Late - PSR - Cntmnt OK
					Containment Leak	6	MR_LATE-PSR-LPF	MR Late - PSR - Cntmnt Fails
		Matrix Intact		LOCA	Containment Holdup	7	OK	FD - Very Small PSR
					Containment Leak	8	OK	FD - Very Small PSR - Cntmnt Fails
	Fuel Fails		Early	LOCA	Containment Holdup	9	FD-MR_EARLY-PSR	FD - MR Early - PSR - Cntmnt OK
		Matrix Fails			Containment Leak	10	FD-MR_EARLY-PSR-LPF	FD - MR Early - PSR - Cntmnt Fails
			Late	LOCA	Containment Holdup	11	FD-MR_LATE-PSR	FD - MR Late - PSR - Cntmnt OK
					Containment Leak	12	FD-MR_LATE-PSR-LPF	FD - MR Late - PSR - Cntmnt Fails



Test conditions	# particles	SiC failures		
		# failures	Failure fraction	Failure fraction (95% Conf)
Irradiation	298000	4	1.3E-05	≤3.1E-05
1600°C	33100	3	9.1E-05	≤2.4E-04
1700°C	12450	7	5.6E-04	≤1.1E-03
1800°C	16500	23	1.4E-03	≤2.0E-03

In addition to these failures, a fraction of the released RNs will be attenuated in the pebble bed and graphite surrounding the core. This attenuation is conditional on the release fraction.

HTGR – Matrix Release (Break in HPB)

The source term is characterized by:

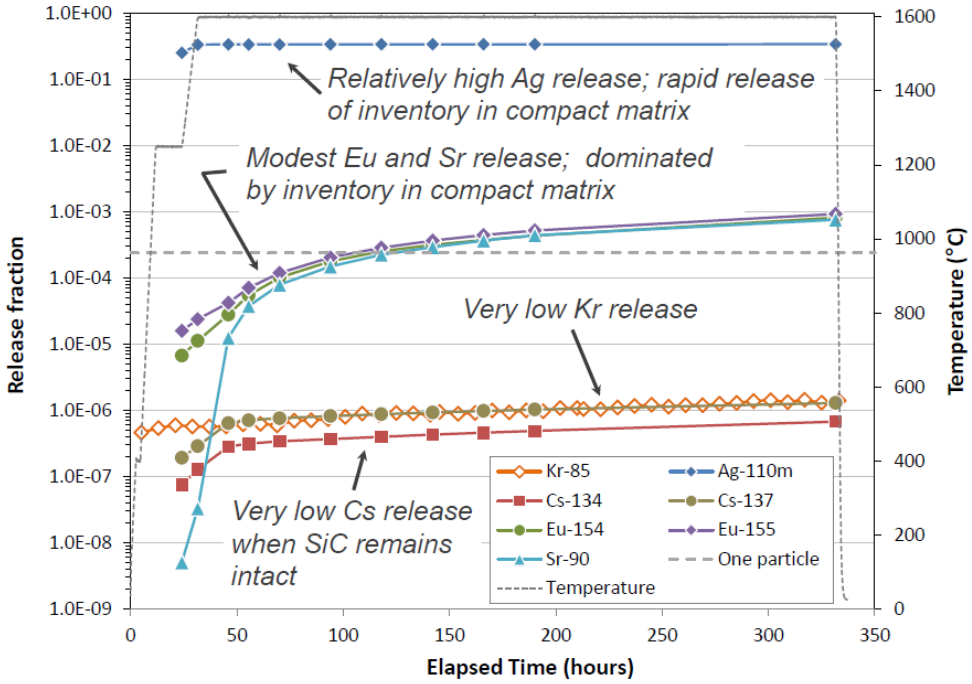
$$ST(S_i, RN_j, t) = I(RN_j) \cdot F(S_i, t) \cdot MR(S_i, RN_j, t) \cdot PSR(S_i, RN_j, t) \cdot LPF(S_i, RN_j, t)$$

Matrix Release

Matrix release rates are strongly dependent on SiC failures and radionuclide classes.

Experimental data is used to estimate the release fractions and release timings for each RN class.

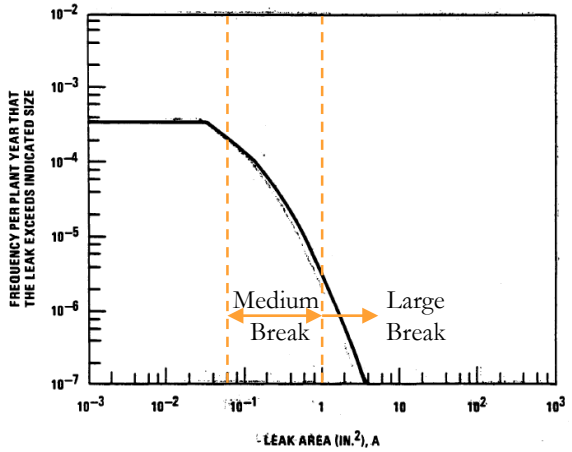
INL AGR-1 safety test results for RN release fractions from TRISO fuel.



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HTGR Scenario – Break in Helium Pressure Boundary

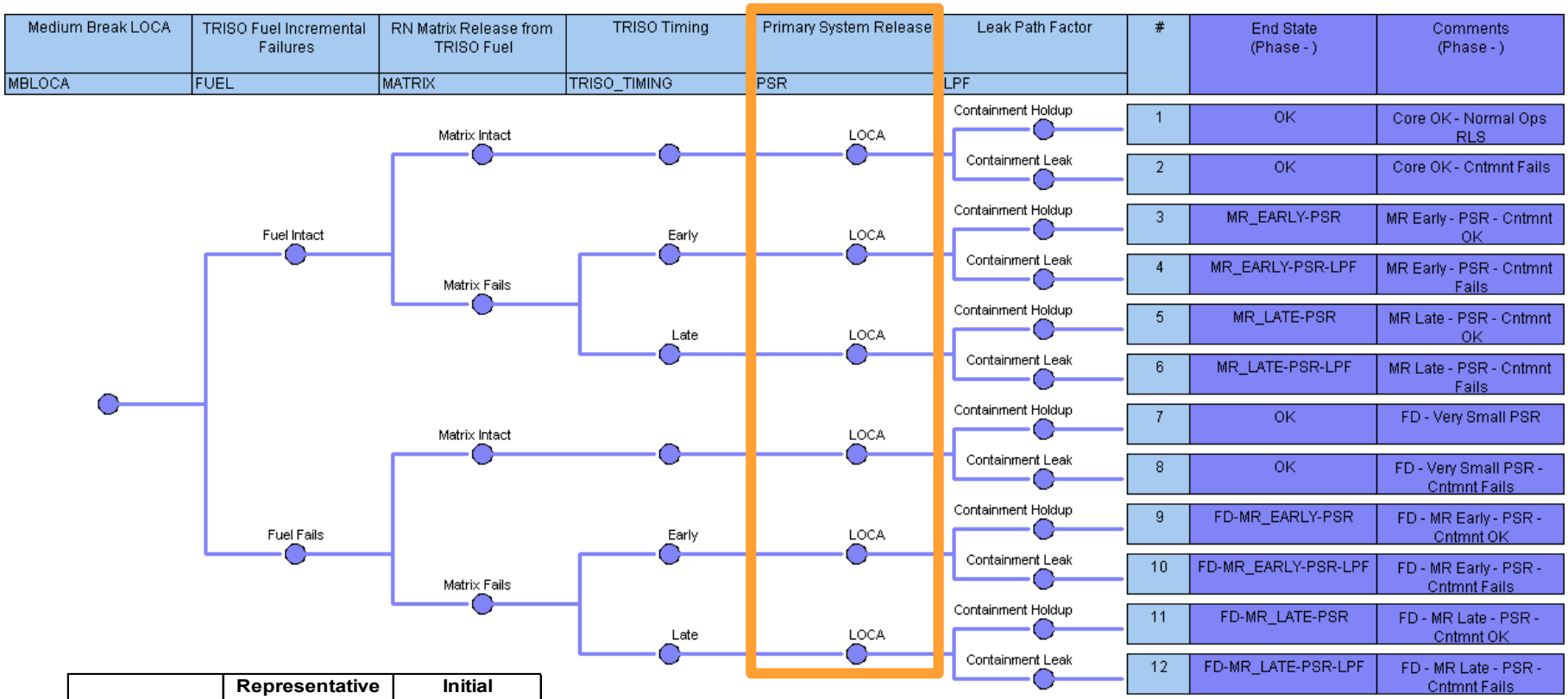
Medium Break LOCA	TRISO Fuel Incremental Failures	RN Matrix Release from TRISO Fuel	TRISO Timing	Primary System Release	Leak Path Factor	#	End State (Phase -)	Comments (Phase -)
MBLOCA	FUEL	MATRIX	TRISO_TIMING	PSR	LPF			
		Matrix Intact		LOCA	Containment Holdup	1	OK	Core OK - Normal Ops RLS
					Containment Leak	2	OK	Core OK - Cntmnt Fails
	Fuel Intact		Early	LOCA	Containment Holdup	3	MR_EARLY-PSR	MR Early - PSR - Cntmnt OK
		Matrix Fails			Containment Leak	4	MR_EARLY-PSR-LPF	MR Early - PSR - Cntmnt Fails
			Late	LOCA	Containment Holdup	5	MR_LATE-PSR	MR Late - PSR - Cntmnt OK
					Containment Leak	6	MR_LATE-PSR-LPF	MR Late - PSR - Cntmnt Fails
		Matrix Intact		LOCA	Containment Holdup	7	OK	FD - Very Small PSR
					Containment Leak	8	OK	FD - Very Small PSR - Cntmnt Fails
	Fuel Fails		Early	LOCA	Containment Holdup	9	FD-MR_EARLY-PSR	FD - MR Early - PSR - Cntmnt OK
		Matrix Fails			Containment Leak	10	FD-MR_EARLY-PSR-LPF	FD - MR Early - PSR - Cntmnt Fails
			Late	LOCA	Containment Holdup	11	FD-MR_LATE-PSR	FD - MR Late - PSR - Cntmnt OK
					Containment Leak	12	FD-MR_LATE-PSR-LPF	FD - MR Late - PSR - Cntmnt Fails



For LOCA initiating events, the primary system release parameter is strongly correlated to the initiating event.

- Basically, it is assumed that there is a break in the system. There will be an attenuation of RNs in the primary system loop. This node accounts for the RN attenuation.
- Additionally, RN release will depend on the leak size. This again correlates to the initiating event.

HTGR Scenario – Break in Helium Pressure Boundary

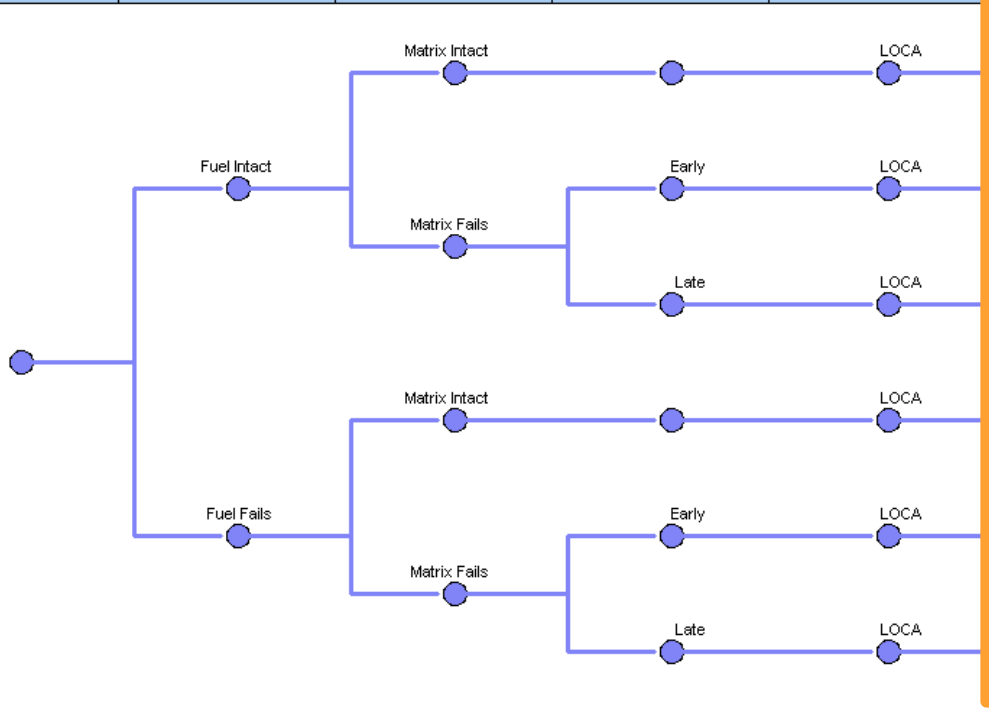


Class Name	Representative Nuclide	Initial Inventory (kg)
Noble Gases	Xe	2.84E-01
Alkali Metals	Cs	8.73E+00
Alkaline Earths	Ba	5.12E+00
Halogens	I	8.03E-02
Chalcogens	Te	3.66E-02
Platiniods	Ru	4.64E-01
Early Transition Elements	Mo	0.00E+00
Tetravalent	Ce	3.43E+01
Trivalents	La	2.45E-02
Uranium	U	0.00E+00
More Volatile Main Group	Cd	9.44E-02
Less Volatile Main Group	Ag	1.03E-02

RN classes will have different release rates from the primary system based on gas and aerosol physics.

For example, noble gases will not deposit on reactor internals or in piping systems, whereas aerosols such as CsI, can deposit in the reactor internals.

HTGR Scenario – Break in Helium Pressure Boundary



Medium Break LOCA	TRISO Fuel Incremental Failures	RN Matrix Release from TRISO Fuel	TRISO Timing	Primary System Release	Leak Path Factor
MBLOCA	FUEL	MATRIX	TRISO_TIMING	PSR	LPF

#	End State (Phase -)	Comments (Phase -)
1	OK	Core OK - Normal Ops RLS
2	OK	Core OK - Cntmnt Fails
3	MR_EARLY-PSR	MR Early - PSR - Cntmnt OK
4	MR_EARLY-PSR-LPF	MR Early - PSR - Cntmnt Fails
5	MR_LATE-PSR	MR Late - PSR - Cntmnt OK
6	MR_LATE-PSR-LPF	MR Late - PSR - Cntmnt Fails
7	OK	FD - Very Small PSR
8	OK	FD - Very Small PSR - Cntmnt Fails
9	FD-MR_EARLY-PSR	FD - MR Early - PSR - Cntmnt OK
10	FD-MR_EARLY-PSR-LPF	FD - MR Early - PSR - Cntmnt Fails
11	FD-MR_LATE-PSR	FD - MR Late - PSR - Cntmnt OK
12	FD-MR_LATE-PSR-LPF	FD - MR Late - PSR - Cntmnt Fails

Leak path factor dependent on containment conditions following LOCA initiating event.

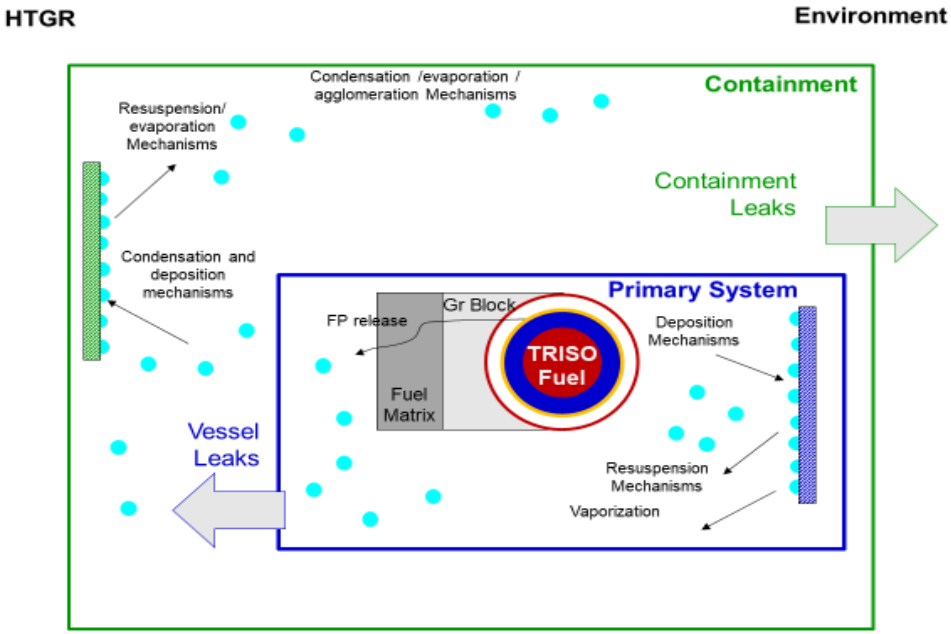
- For MBLOCA and LBLOCA, it is expected that high pressure/temperature reactor coolant (helium gas) will pressurize containment leading to increased containment leakage.

Conclusions – HTGR, Break in HPB

The source term barrier approach is able to capture important phenomenon for HTGR scenarios.

Operational experience, operational testing, and experiments conducted by the national labs is used to inform parameter estimates.

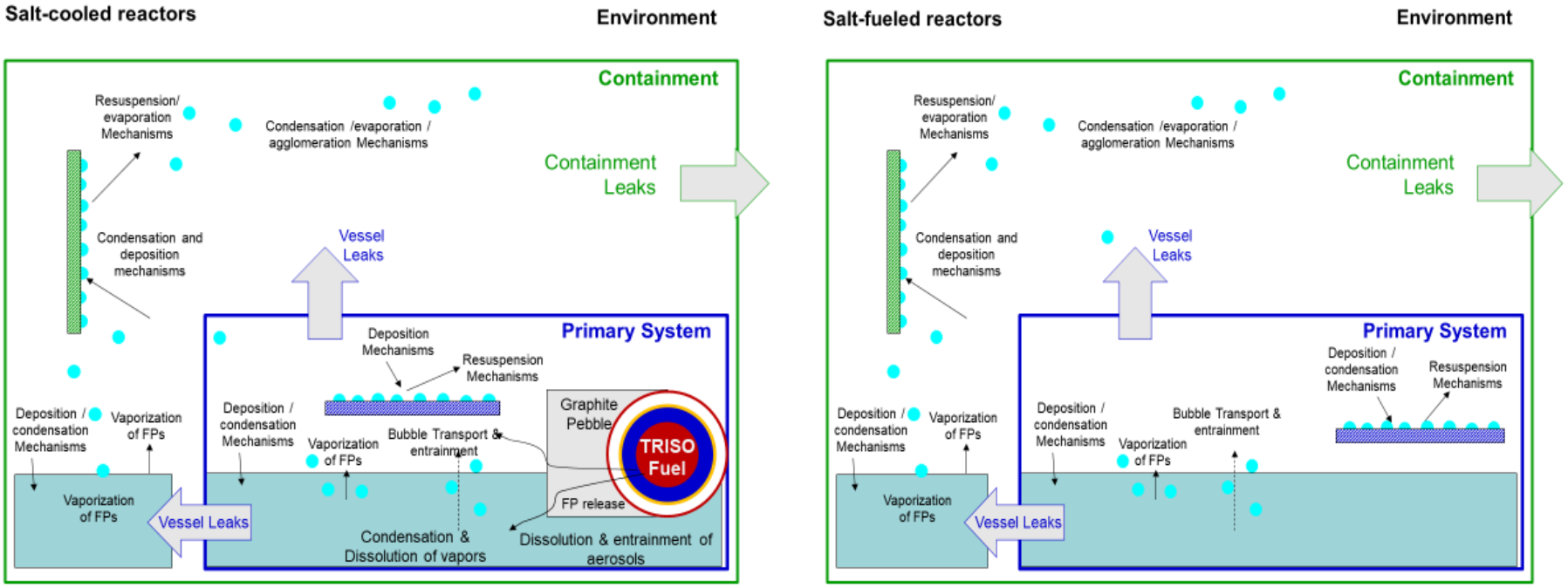
The initiating events, conditional probabilities, and end states are modeled to predict the amount of RNs released to the environment.





MSR Scenarios

Source Term Evolution Pathways – Molten Salt Reactors



Source Term Considerations for MSR Scenarios (1/3)

MSR accident source terms are heavily influenced by the presence of water[1]

- Fuel salt has no rapid chemical reactions with containment liner, atmosphere, or water
- Steam production drives containment pressure and opening of rupture discs to the vapor-condensing system
  - MSRE experienced numerous water leaks into reactor and drain tank containment cells[2]

During MSRE maximum credible accident, containment returns to near atmospheric pressure after approximately 5 minutes of venting to vapor-condensing system[1]

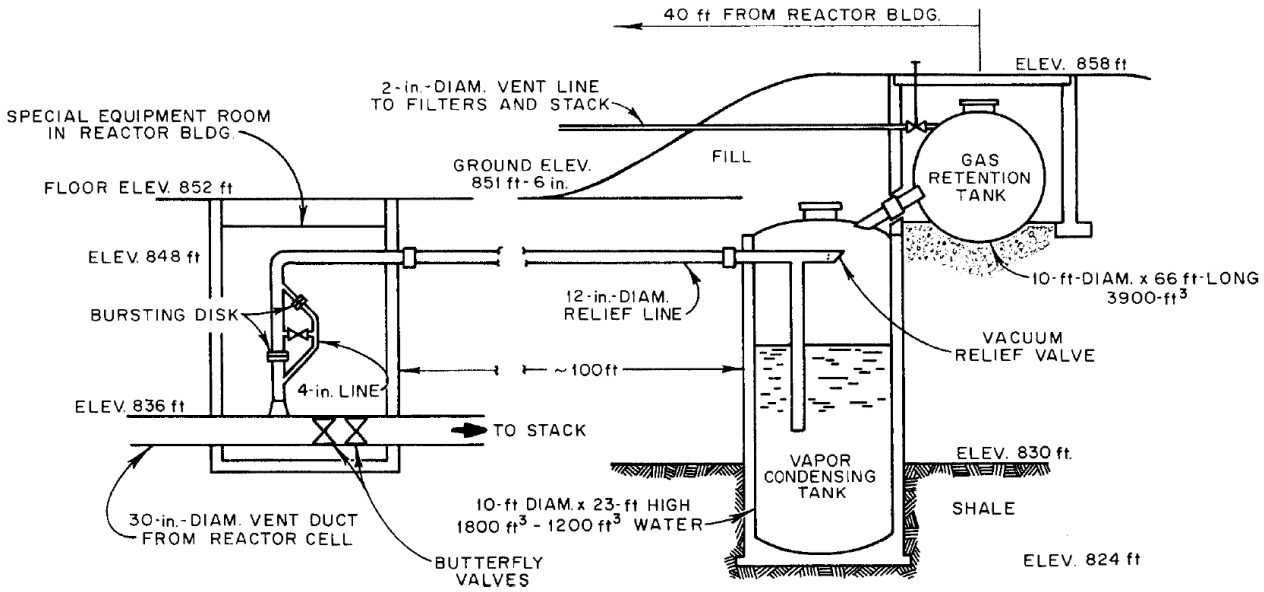
Thermal characteristics of containment vessels for MSRE were considered sufficient to handle thermal loads, including decay heat, from fuel salt spills[1]

[1]: ORNL-TM-732, p.240-257
[2]: ORNL-TM-3039

Source Term Considerations for MSR Scenarios (2/3)

Routing through the vapor-condensing system is assumed to scrub 95% of all classes of radionuclides[1]

- MSRE experienced significant leaks of the lines from containment to the vapor-condensing system[2]
  - Unavailability of vapor-condensing system is assumed to result in an unscrubbed release



[1]: ORNL-TM-732, p.244
[2]: ORNL-TM-3039

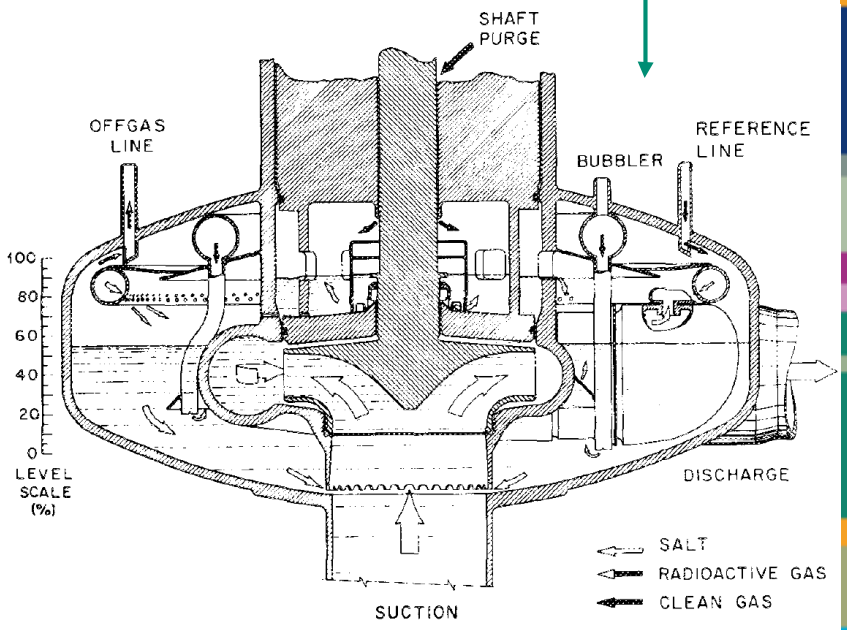
Source Term Considerations for MSR Scenarios (3/3)

Noble gases (and precursors) constantly produced throughout the primary system

- Partially stripped in cover gas space of fuel pump (highest point of system) to improve reactivity[1,2]
- MSRE had significant diffusion of xenon into core graphite
  - MSBR called for significant reduction of graphite permeability[3]

It is assumed that noble gases escape the primary system in the same fraction as fuel salt if a break occurs

- Once gas temperatures equilibrate, Xe and Kr have higher density than the nitrogen containment atmosphere



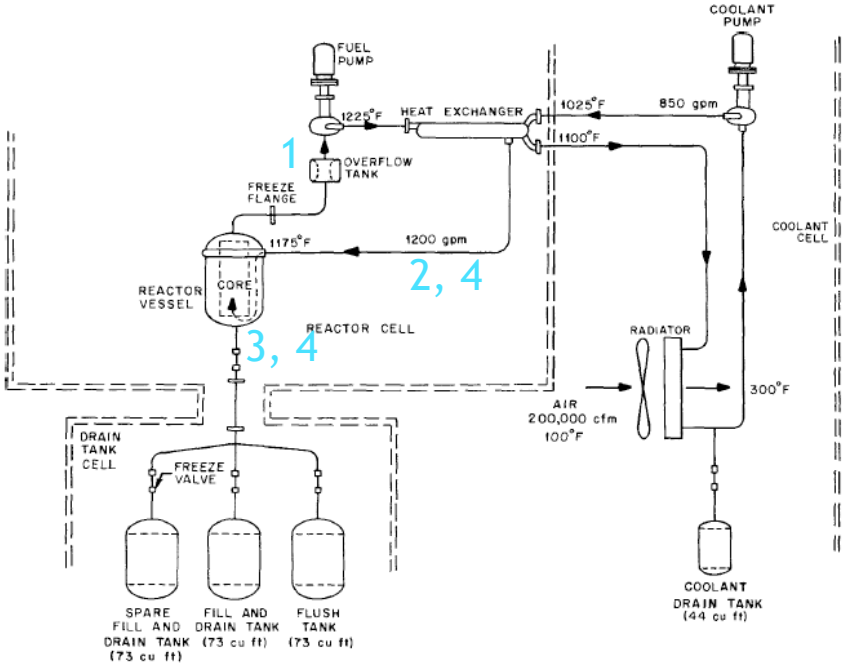
[1]: ORNL-4865
[2]: ORNL-TM-3464
[3]: ORNL-4389

Development of MSR Scenario PCL (1/3)

Primary coolant leak (PCL) spills to reactor cell

Sequences:

1. Small break (5% of inventory in a day)
   - MSRE experienced one small fuel salt leak at a freeze flange[1]
2. Recirculation line break (40% of inventory in 15s)[2]
3. Drain line break (100% of inventory in 370s)[2]
4. Coincident recirculation and drain line breaks (100% of inventory in 280s)[2]



[1]: ORNL-TM-3039, p.93
[2]: ORNL-TM-732, p.241

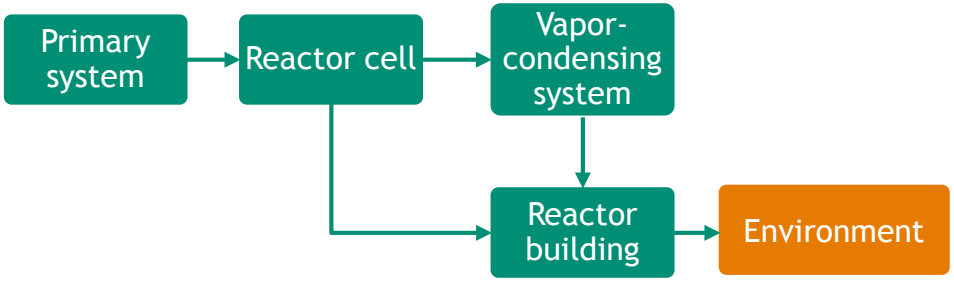
Development of MSR Scenario PCL (2/3)

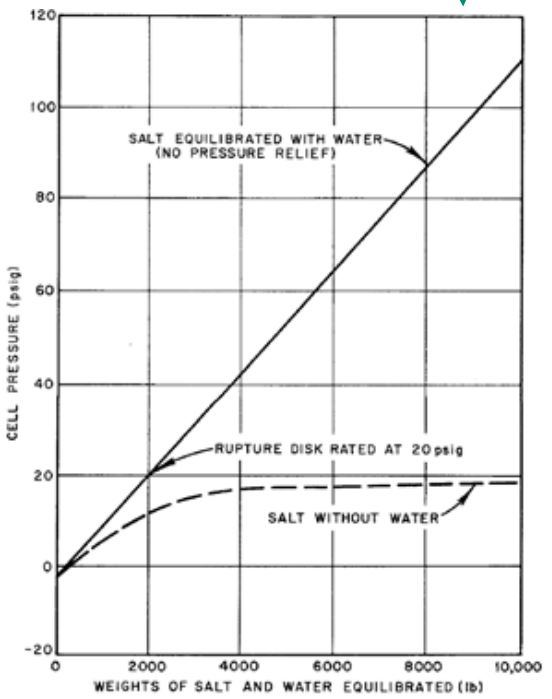
Only a large leak (>=20% of salt inventory, sequences 2, 3, and 4) with water ingress is assumed to rupture containment

- Conceivable that a single event causes both a PCL and water spill into reactor cell
  - Fuel pump motor is water-cooled
- MSRE experienced numerous water leaks into reactor cell from containment cooling system[1]

After containment rupture, steam flows (ideally) to vapor-condensing system or to reactor building

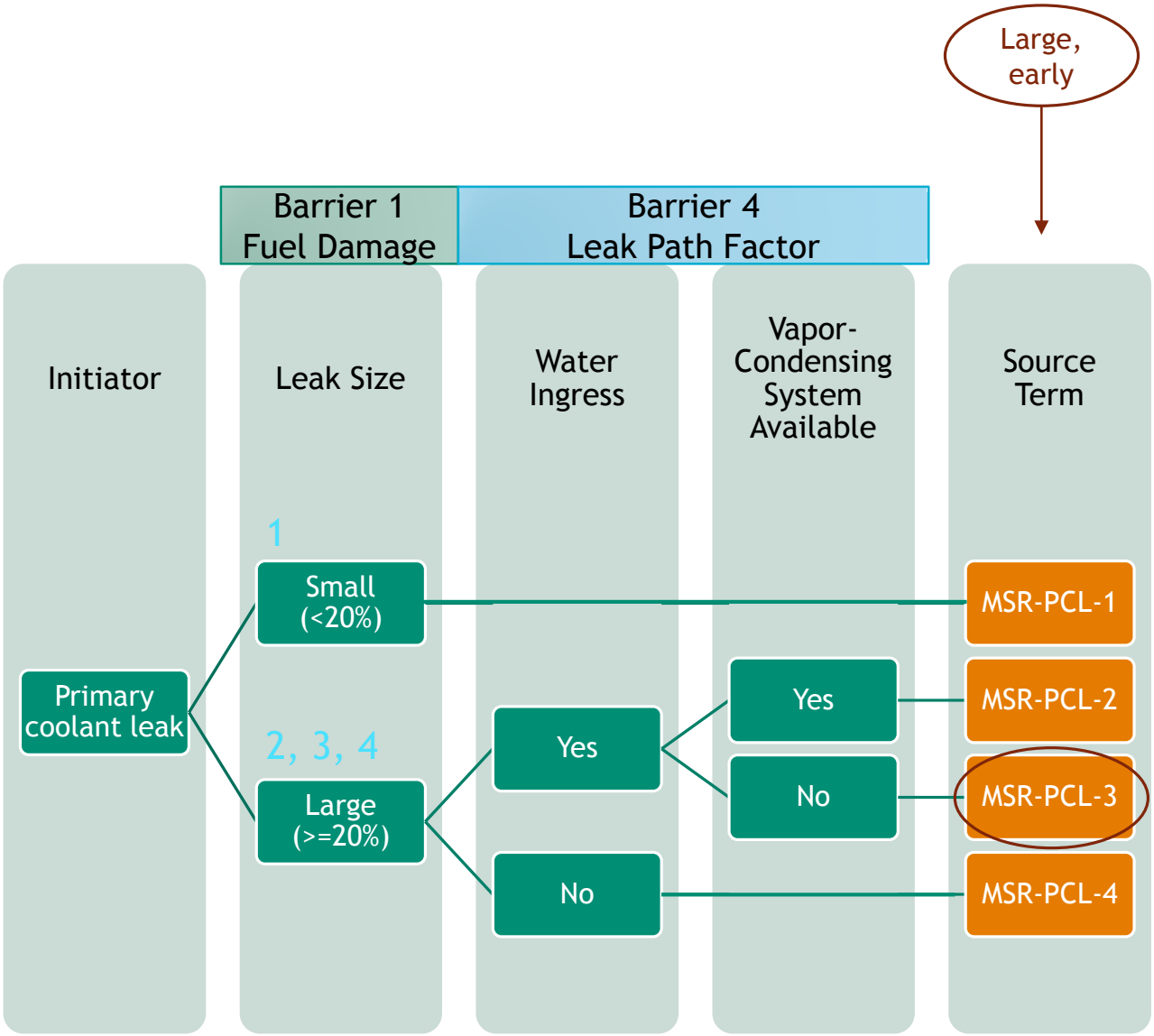
Both the reactor building and vapor-condensing system vent to the reactor building stack





[1]: ORNL-TM-3039

Development of MSR Scenario PCL (3/3)



MSR Scenario PCL - FD

For the liquid fuel MSR, FD represents loss of inventory from the primary system

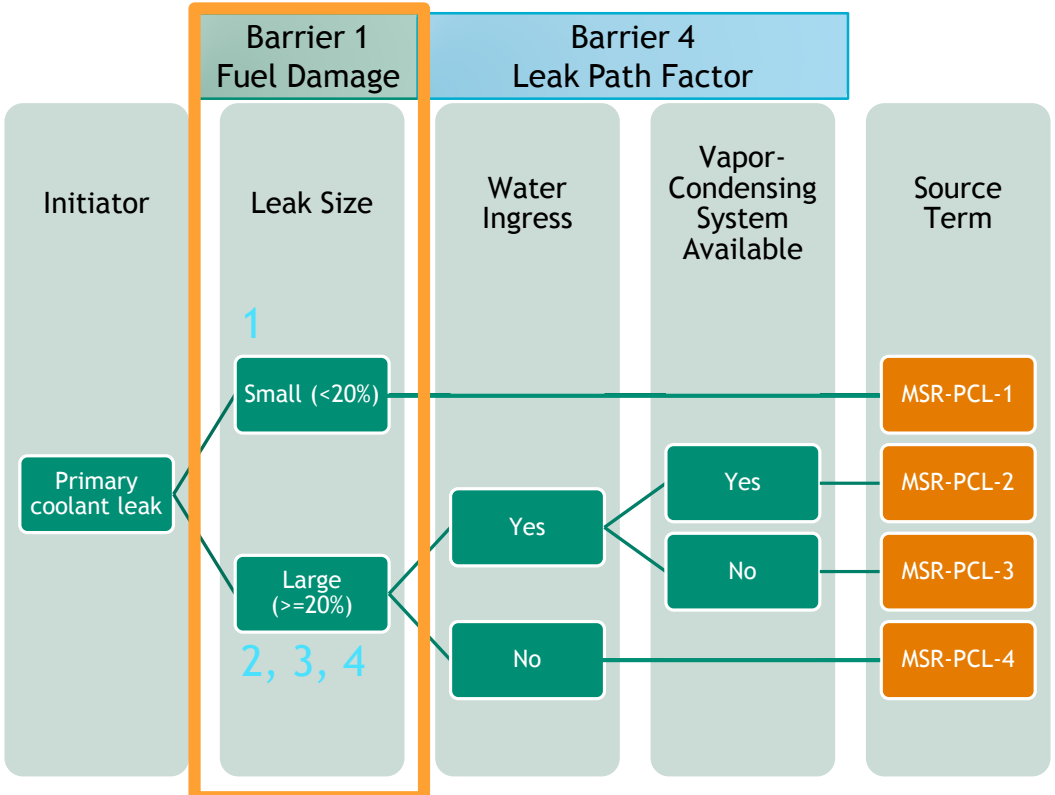
- Used in lieu of PSR for consistency in order of barriers

Sequence 1

- Release from primary system is small and late
- End state MSR-PCL-1 does not overpressurize containment even with water ingress

Sequences 2, 3, 4

- Release from primary system is large and early
- Source term depends largely on subsequent events



MSR Scenario PCL - MR

For the chosen MSR sequences, MR is immediate and depends only on the radionuclide class

- Examination of MSRE fuel revealed that numerous elements form stable fluoride salts[1]
  - Solubility in accident conditions may differ

Barrier 2
Matrix Release

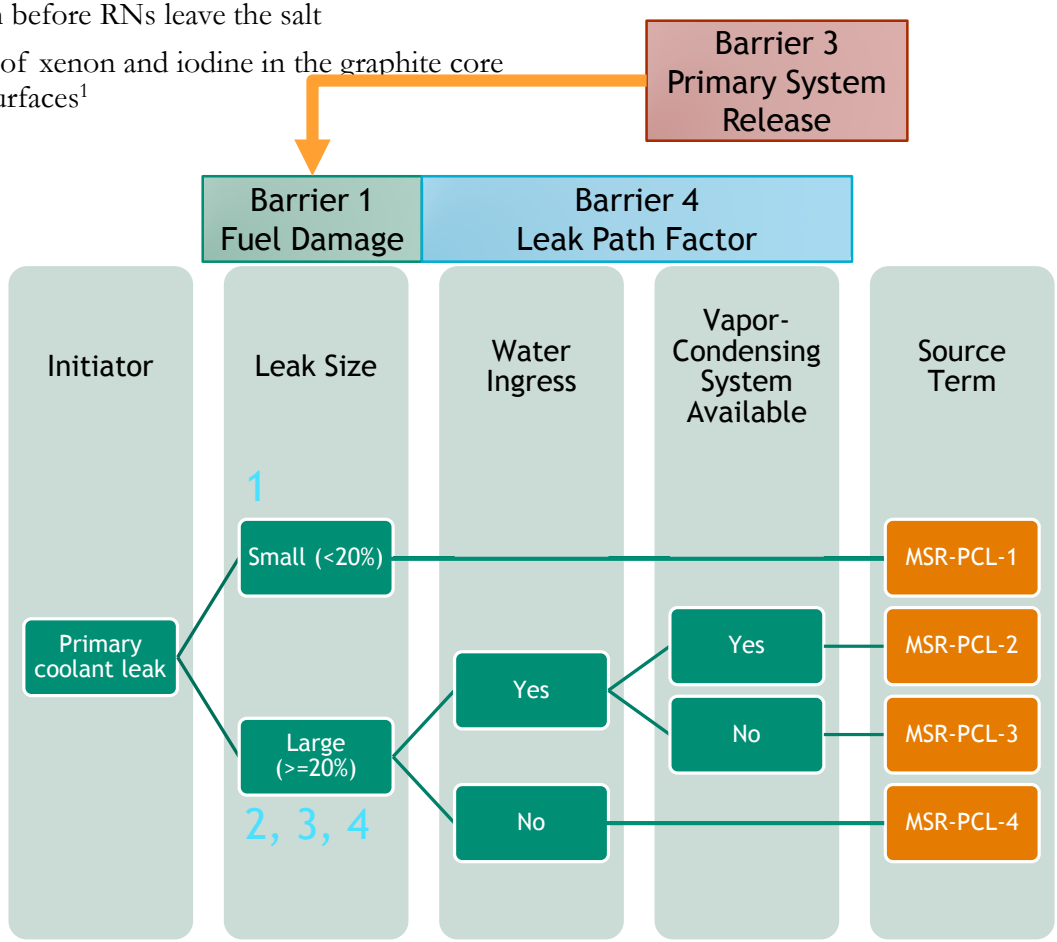
MACCS Class Name	Rep. Element	ORNL-4865 Class Name	MR Mag. Bin
Noble Gases	Xe	Noble Gases	High
Alkali Metals	Cs	Stable salt seekers (noble gas precursors)	Medium
Alkaline Earths	Ba	Stable salt seekers (noble gas precursors)	Medium
Halogens	I	Iodine	Medium
Chalcogens	Te	Tellurium, antimony	Medium
Platinoids	Ru	Noble metals	Medium
Early Transition Elements	Mo	Noble metals	Medium
Tetravalent	Ce	Stable salt seekers	Low
Trivalent	La	Stable salt seekers	Low
Uranium	U	Stable salt seekers	Low
More Volatile Main Group	Cd	Stable salt seekers	Medium
Less Volatile Main Group	Ag	Noble metals	Medium
Boron	B	Stable salt seekers	Medium

[1]: ORNL-4865

MSR Scenario PCL - PSR

For the chosen liquid fuel MSR sequences, PSR is immediate and total

- The concept of fuel damage does not apply to a liquid fuel MSR but FD is used to reflect fuel salt loss to maintain the order of barriers
  - I.e., the salt is assumed to leave the primary system before RNs leave the salt
  - MSRE experienced significant unintended holdup of xenon and iodine in the graphite core as well as noble metal plating on primary system surfaces[1]

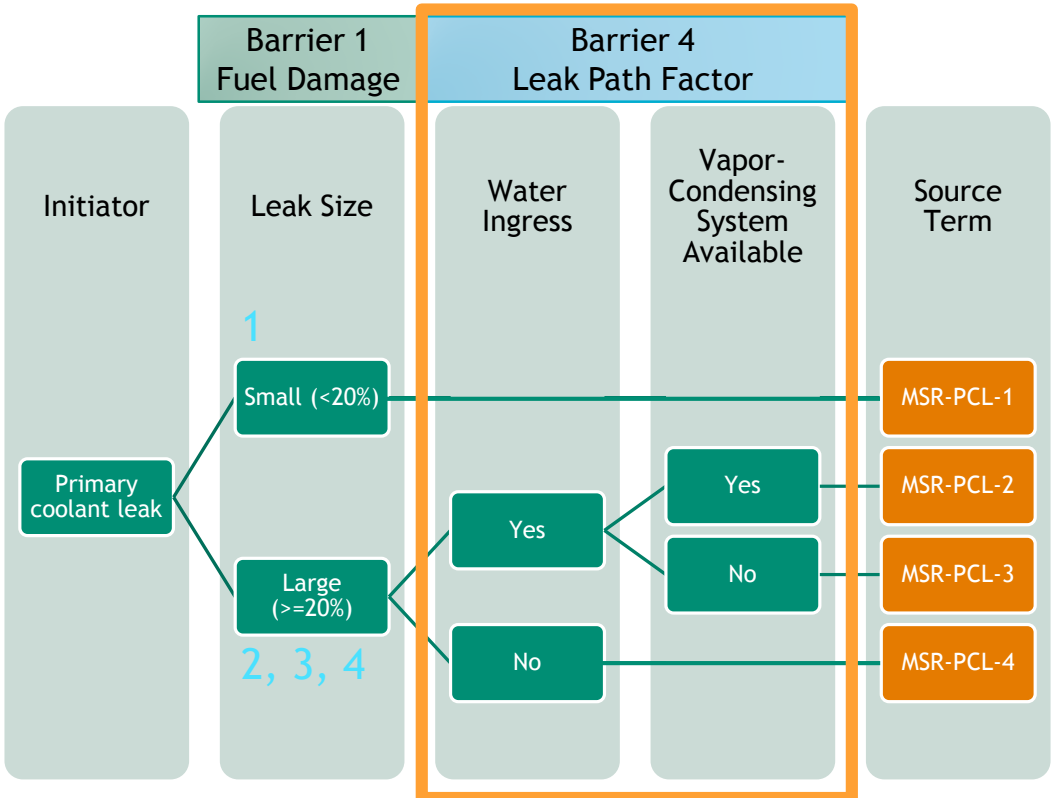


[1]: ORNL-4865

MSR Scenario PCL – LPF

The LPF reflects radionuclide releases from containment

- Without water ingress, driving pressure is low and containment does not rupture or leak
- Class-specific values obtained from MSRE maximum credible accident
  - Analysis in [1] included time history of material release from containment
- Vapor-condensing system may be available to scrub release
  - Available results in maximum credible accident releases
  - Unavailable results in twenty times the maximum credible accident releases at each time step



[1]: ORNL-TM-732, p.240-257

MSR Scenario PCL – Source Terms

MSR-PCL-1/4

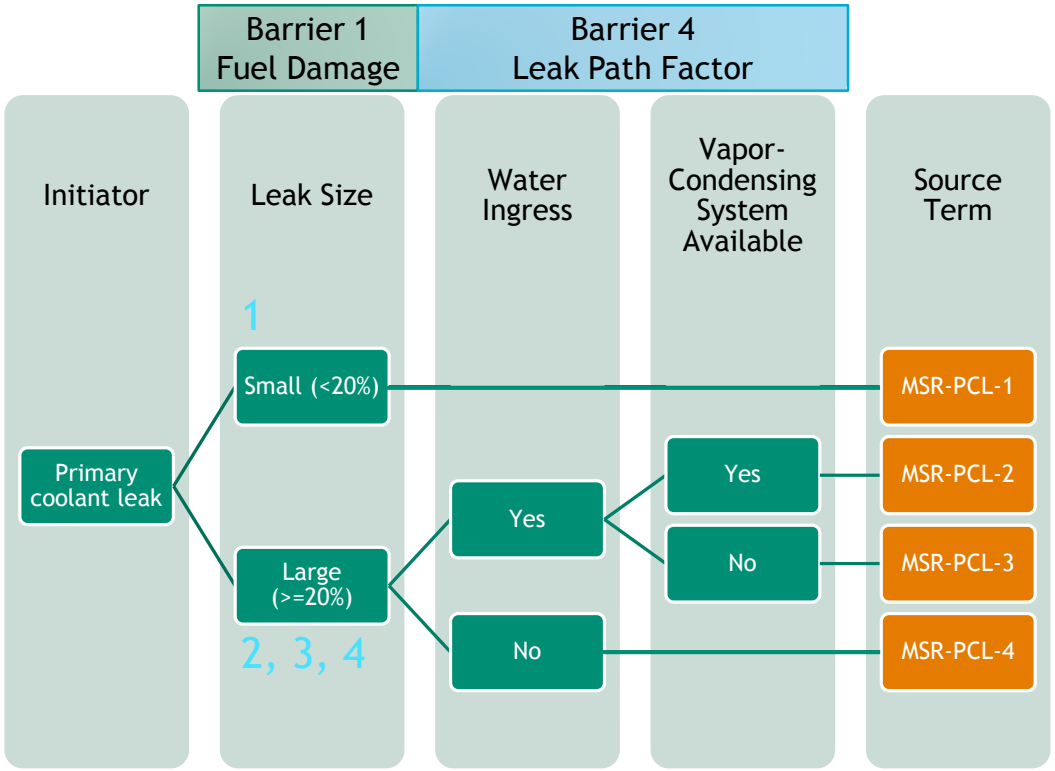
- Containment does not rupture to vapor-condensing system
- < Containment design leakage early
  - Diminishes as salt cools on containment structures

MSR-PCL-2

- Scrubbed early release

MSR-PCL-3

- Unscrubbed early release



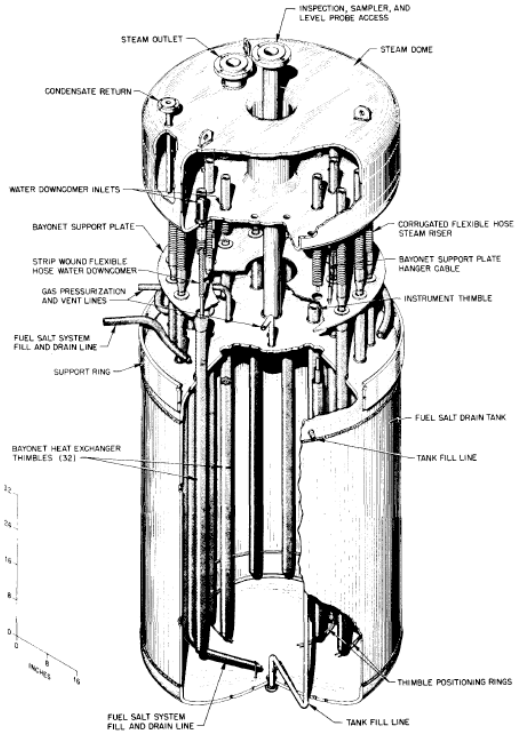
[1]: ORNL-TM-732, p.240-257

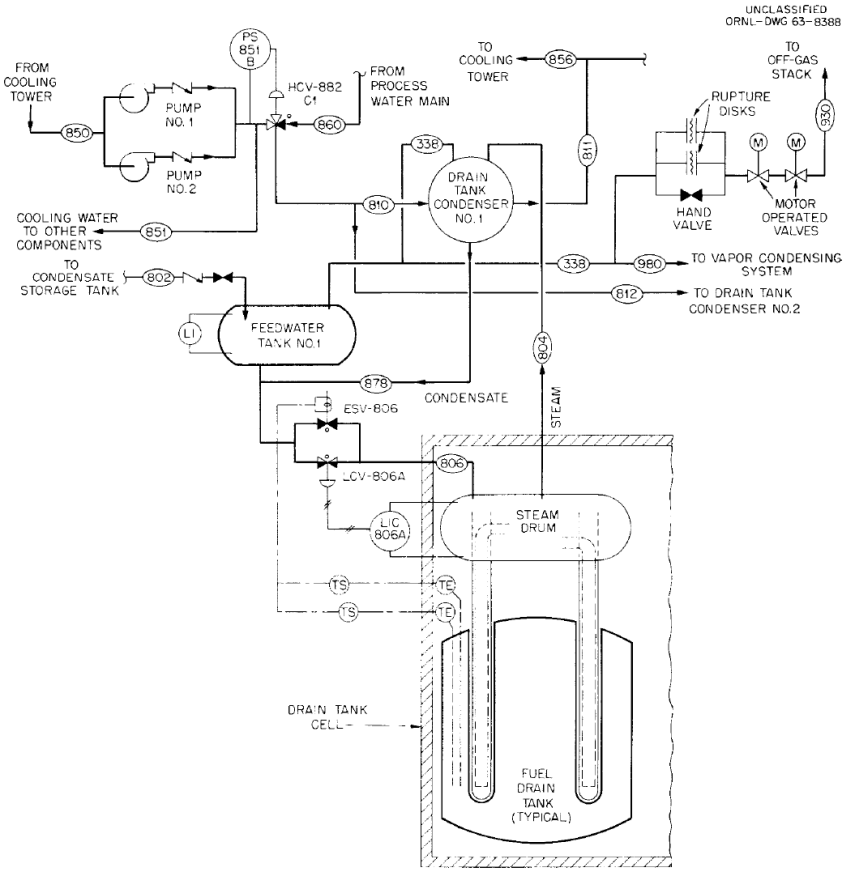
Development of MSR Scenario LOHR (1/3)

Loss of decay heat removal (LOHR) may eventually spill to drain tank cell

Sequences:

1. Loss of feedwater with reserve tank available, steam thimble failure
2. Loss of feedwater with reserve tank available, no steam thimble failure
3. Loss of feedwater with reserve tank unavailable

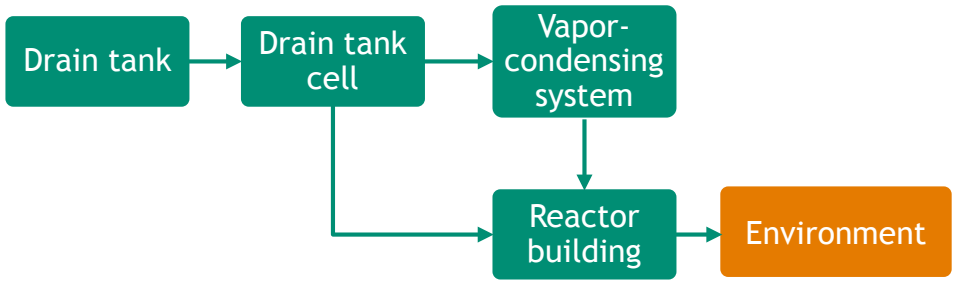


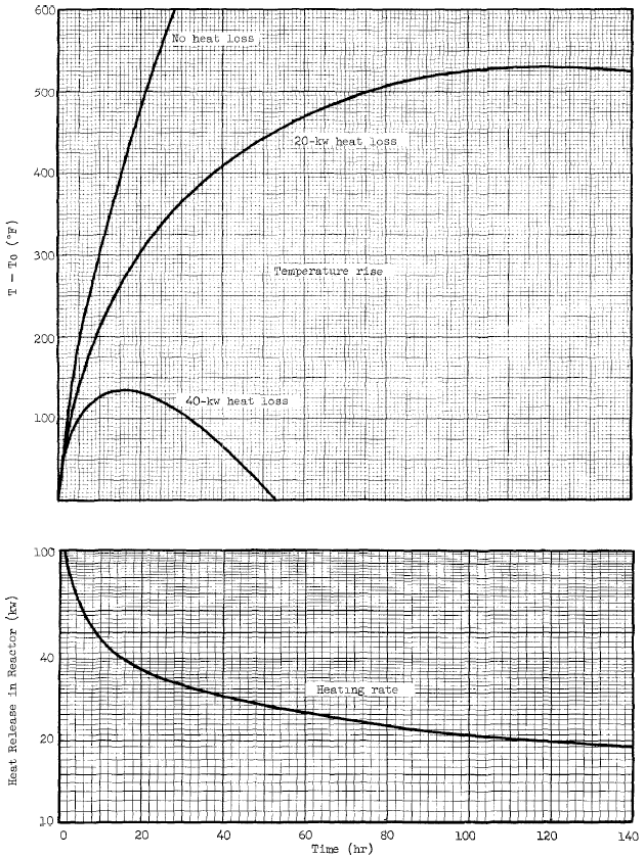


Development of MSR Scenario LOHR (2/3)

Loss of decay heat removal (LOHR) may eventually spill to drain tank cell

- Drain tank failure due to high temperature creep or overpressurization by steam
  - MSRE often operated with reserve tank empty due to valve leaks and so cooling may be unavailable[1]
    - Vessel is assumed to fail after a day if reserve is not available[2]
  - Steam thimble failure may lead to water ingress
    - Conceivable that a single event causes both a loss of the feedwater system and a water leak
    - Vessel is assumed to fail if water ingress occurs
- Containment failure depends on water ingress
- After containment rupture, steam flows either to reactor building or (ideally) to vapor-condensing system





[1]: ORNL-TM-3039, p.134
[2]: ORNL-TM-732, p.227

Development of MSR Scenario LOHR (3/3)

MSR-LOHR-1/2 (1): containment rupture within minutes

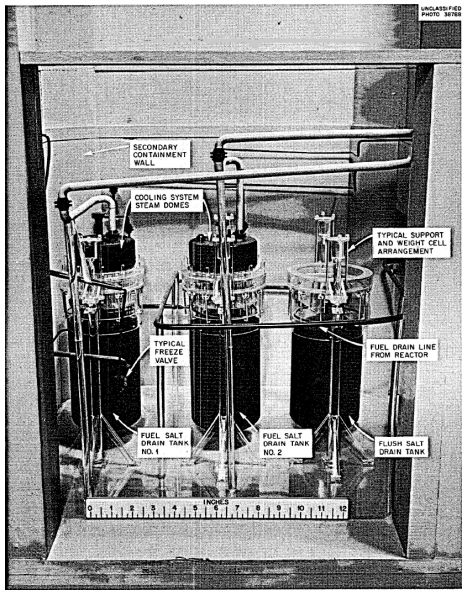
- In-tank water ingress assumed to occur promptly due to thermal shock to steam thimbles

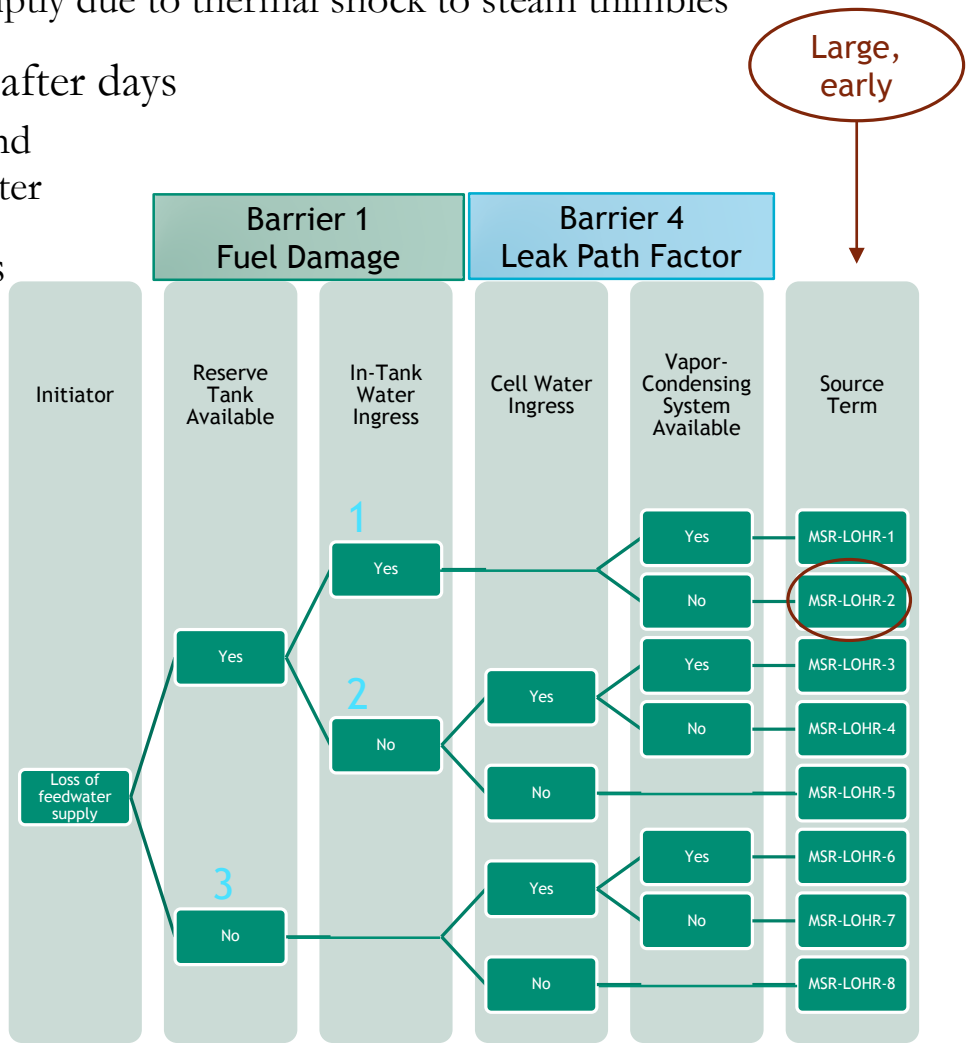
MSR-LOHR-3/4 (2): containment rupture after days

- Reserve maintains temperature for 8 hours and then tank heats to failure, exposing salt to water

MSR-LOHR-6/7 (3): containment ruptures after a day

- Tank heats to failure and then salt encounters water





MSR Scenario LOHR - FD

For the liquid fuel MSR, FD represents loss of inventory from the primary system

- Used in lieu of PSR for consistency in order of barriers

Sequence 1

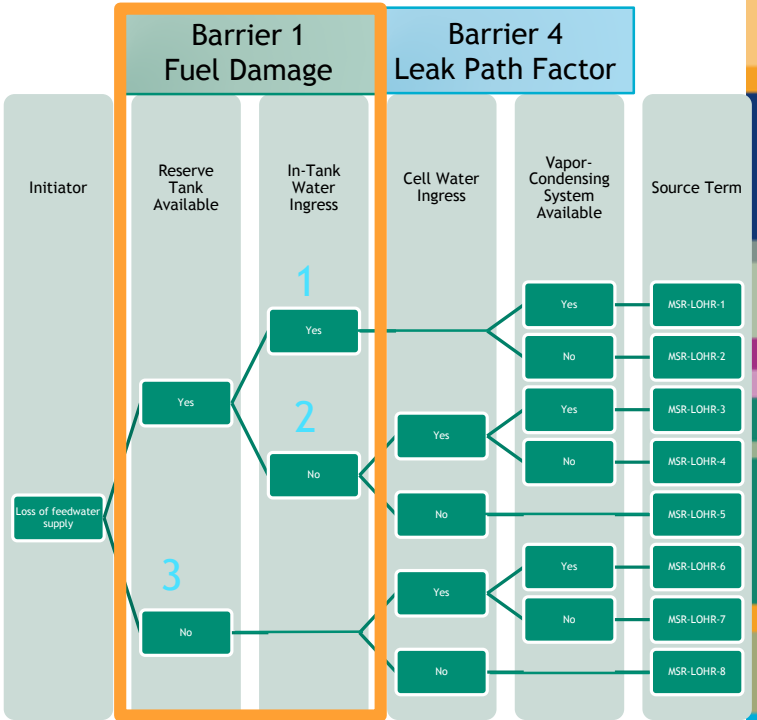
- Release from primary system is large and early
- Water ingress promotes overpressurization of drain tank

Sequence 2

- Release from primary system is large and late
- Availability of reserve tank delays start of significant drain tank heatup
- Drain tank fails due to high temperature creep after days

Sequence 3

- Release from primary system is large and late
- Drain tank fails due to high temperature creep after a day



MSR Scenario LOHR – MR (identical to PCL MR)

For the chosen MSR sequences, MR is immediate and depends only on the radionuclide class

- Examination of MSRE fuel revealed that numerous elements form stable fluoride salts[1]
  - Solubility in accident conditions may differ

Barrier 2
Matrix Release

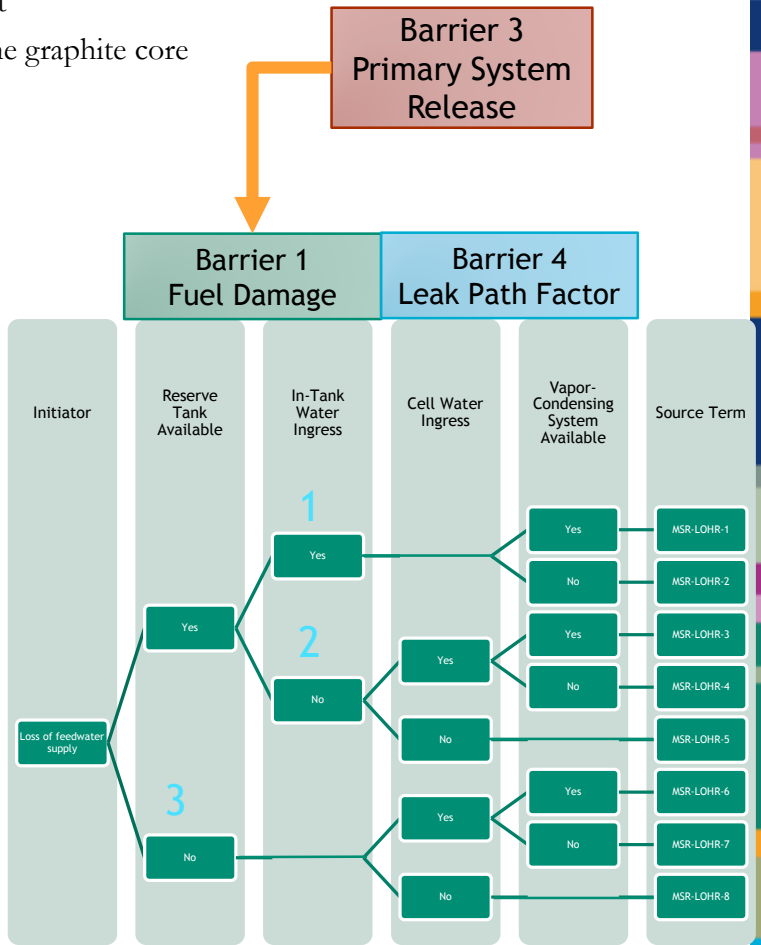
MACCS Class Name	Rep. Element	ORNL-4865 Class Name	MR Mag. Bin
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Alkali Metals	Cs	Stable salt seekers (noble gas precursors)	Medium
Alkaline Earths	Ba	Stable salt seekers (noble gas precursors)	Medium
Halogens	I	Iodine	Medium
Chalcogens	Te	Tellurium, antimony	Medium
Platinoids	Ru	Noble metals	Medium
Early Transition Elements	Mo	Noble metals	Medium
Tetravalent	Ce	Stable salt seekers	Low
Trivalent	La	Stable salt seekers	Low
Uranium	U	Stable salt seekers	Low
More Volatile Main Group	Cd	Stable salt seekers	Medium
Less Volatile Main Group	Ag	Noble metals	Medium
Boron	B	Stable salt seekers	Medium

[1]: ORNL-4865

MSR Scenario LOHR – PSR (identical to PCL PSR)

For the chosen liquid fuel MSR sequences, PSR is immediate and total

- The concept of fuel damage does not apply to a liquid fuel MSR but FD is used to reflect fuel salt loss to maintain the order of barriers
  - I.e., the salt is assumed to leave the primary system before RNs leave the salt
  - MSRE experienced significant unintended holdup of xenon and iodine in the graphite core as well as noble metal plating on primary system surfaces[1]

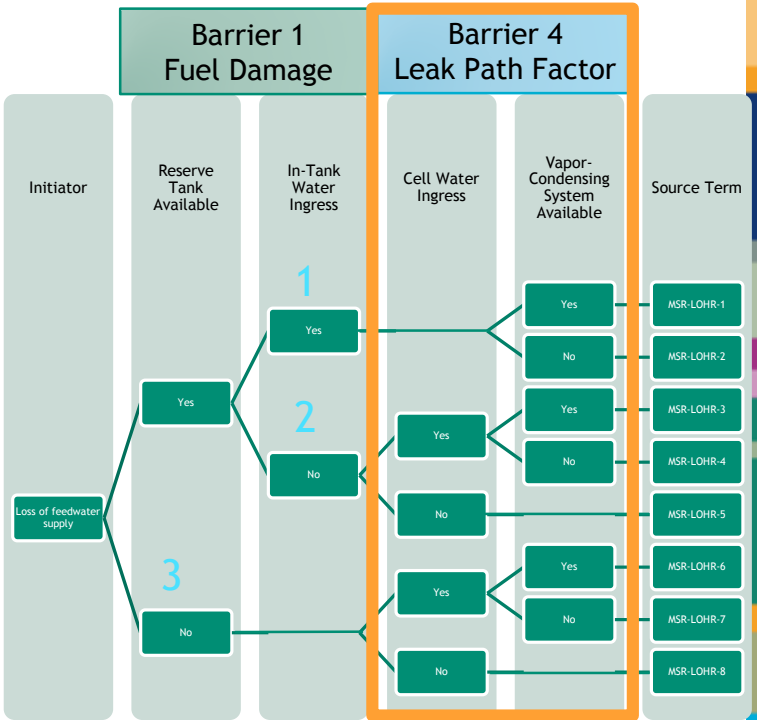


[1]: ORNL-4865

MSR Scenario LOHR – LPF

The LPF reflects radionuclide releases from containment

- Without water ingress, driving pressure is low and containment does not rupture or leak
  - In-tank and other water ingress are considered equally capable of pressurizing containment
- Class-specific values obtained from MSRE maximum credible accident
  - Analysis in [1] included time history of material release from containment
- Timing starts at release of salt into containment
- Vapor-condensing system may be available to scrub release
  - Available results in maximum credible accident releases
  - Unavailable results in twenty times the maximum credible accident releases at each time step



[1]: ORNL-TM-732, p.240-257

MSR Scenario LOHR – Source Terms

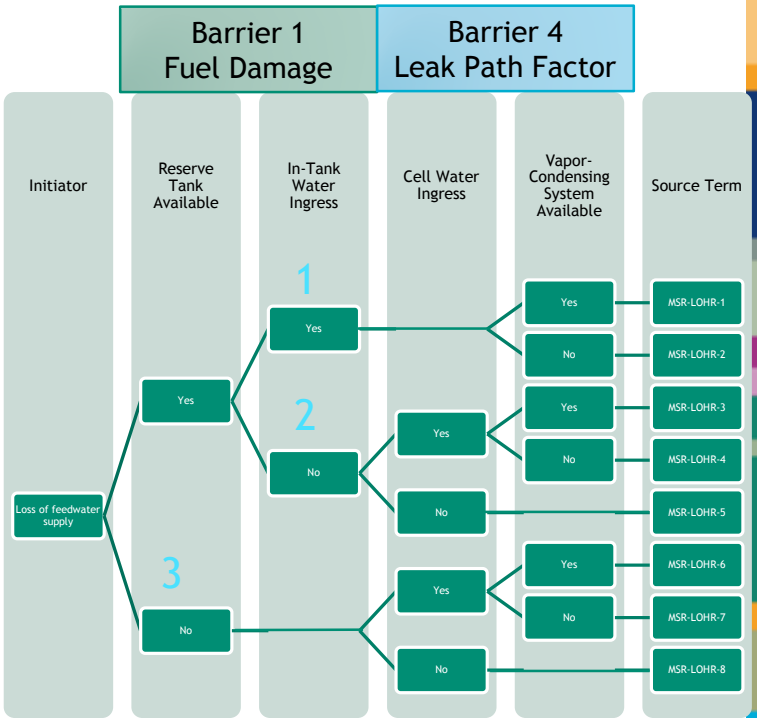
MSR-LOHR-1
- Scrubbed early release

MSR-LOHR-2
- Unscrubbed early release

MSR-LOHR-3/6
- Scrubbed late release

MSR-LOHR-4/7
- Unscrubbed late release

MSR-LOHR-5/8
- Containment does not rupture to vapor-condensing system
- < Containment design leakage early
  - Diminishes as salt cools on containment structures



MSR Source Term Discussion (1/2)

MSR release scenarios vary from release within minutes to delay of days

- Holdup in drain tank delays release even with no cooling
  - Uncertainty in Hastelloy N high temperature creep behavior
- Water ingress required for release beyond 1% design containment leakage

[1]: ORNL-TM-3122, p.20

MSR Source Term Discussion (2/2)

Source term development reflects MSRE design and circa 1969 state of knowledge

- Larger scale designs may invalidate certain assumptions
  - E.g., conduction to containment vessel may no longer be sufficient for ex-vessel decay heat removal
- By 1970, designs proposed cooling the drain tank via a salt loop to eliminate a significant source of water inside containment[1]

Additional work should focus on adapting the study to more recent designs

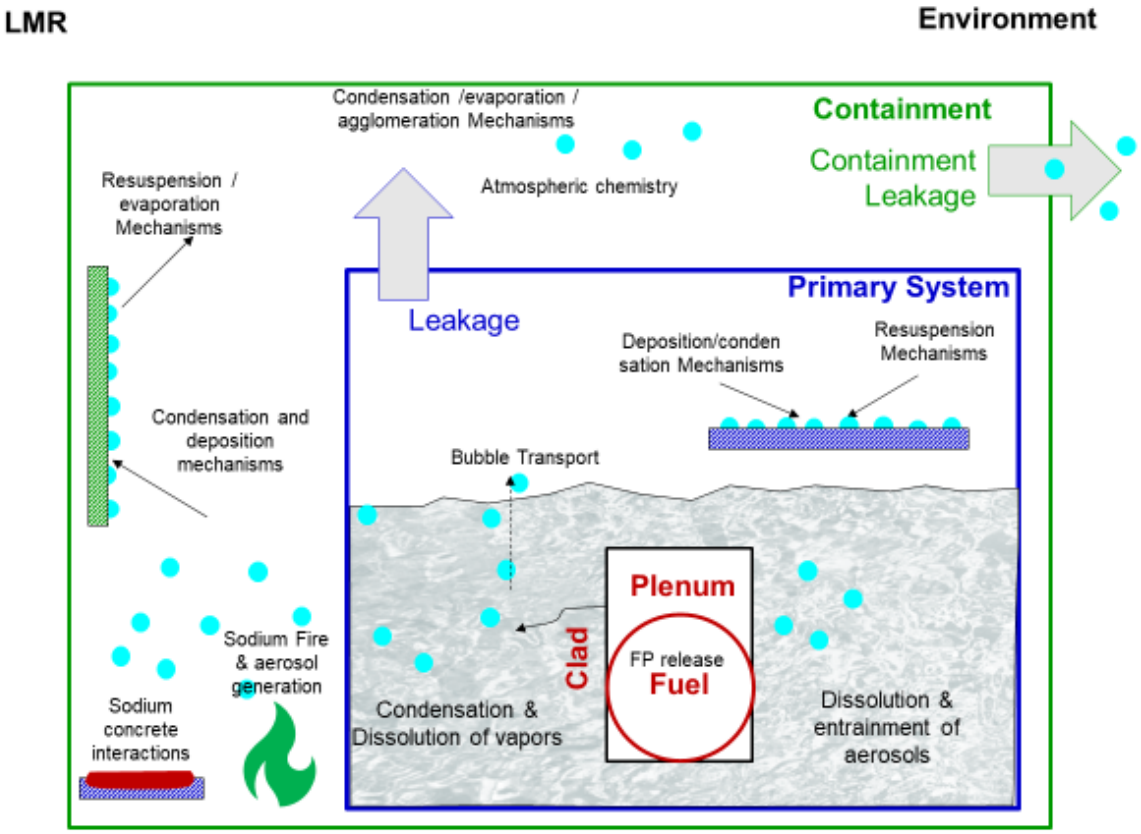
- MSRE: 20MWt, liquid fuel, fluoride salt, piping to heat exchangers, thermal spectrum
  - Kairos: 300MWt, TRISO fuel, fluoride salt, piping to heat exchangers, thermal spectrum
  - Terrapower: 1000 MWe, liquid fuel, chloride salt, heat exchangers in reactor vessel, fast spectrum

[1]: ORNL-TM-3122, p.20

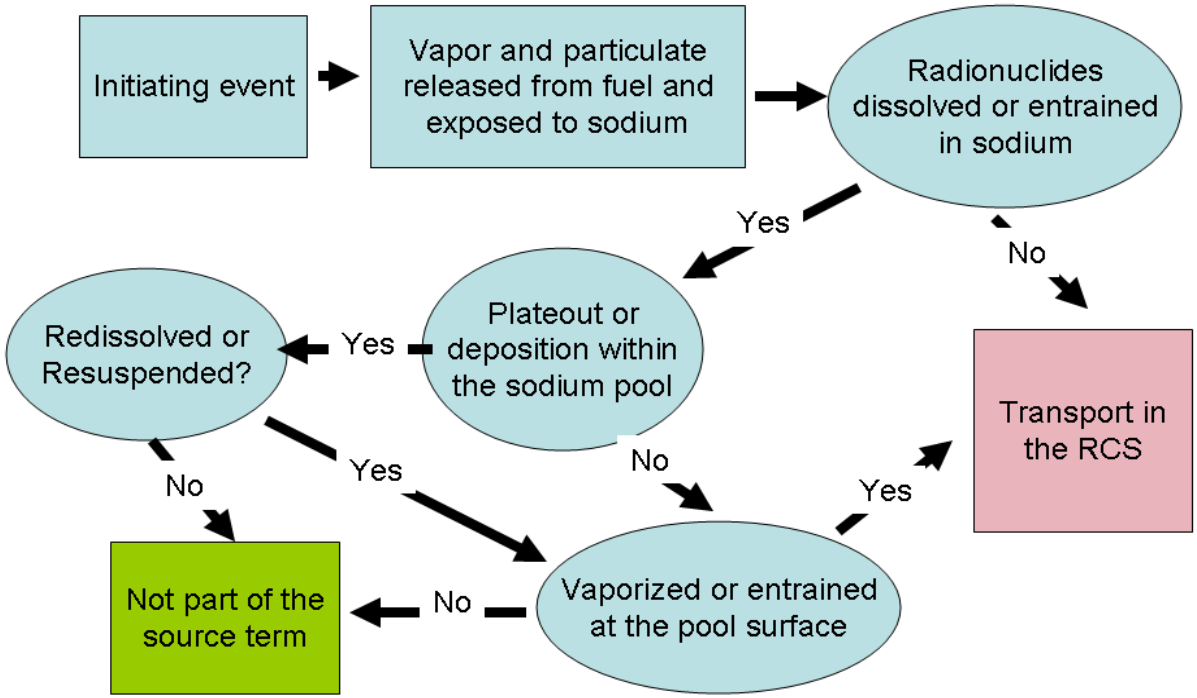


SFR Scenarios

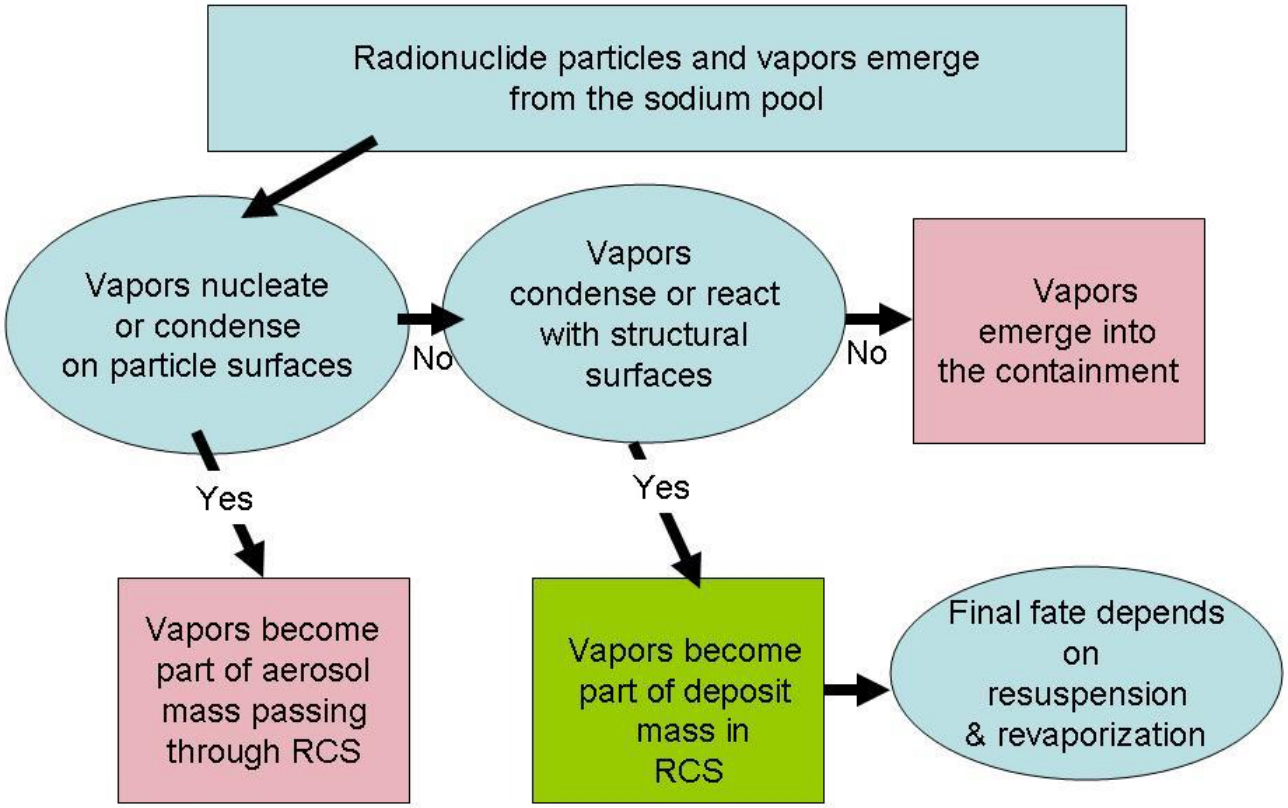
Source Term Evolution Pathways – Liquid Metal Reactors



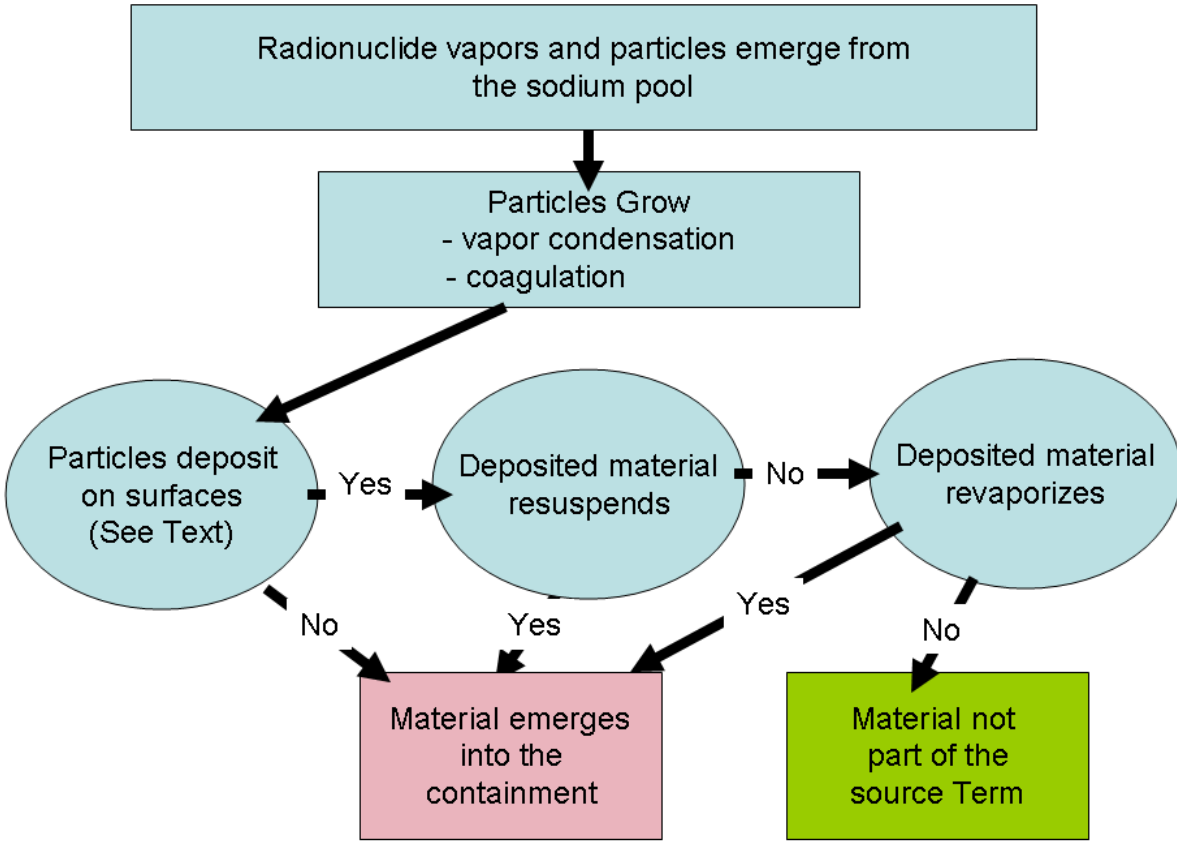
Source Term Evolution – Transport into RCS



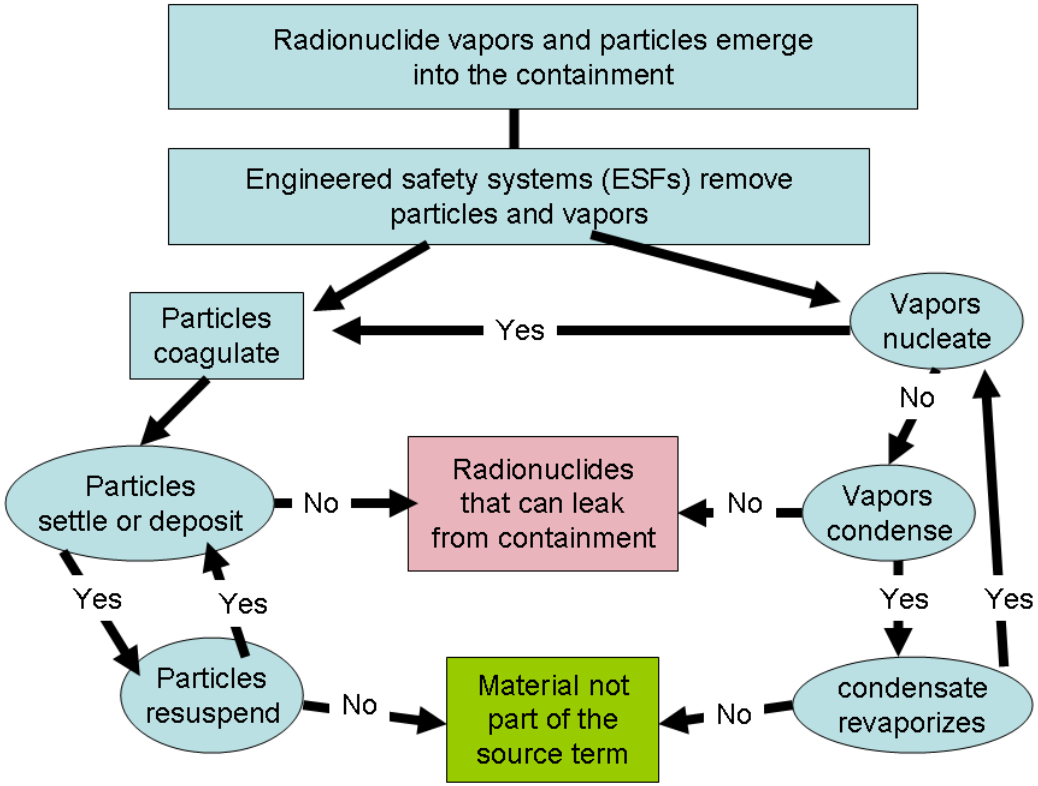
Source Term Evolution – Vapor Release from Sodium Pool



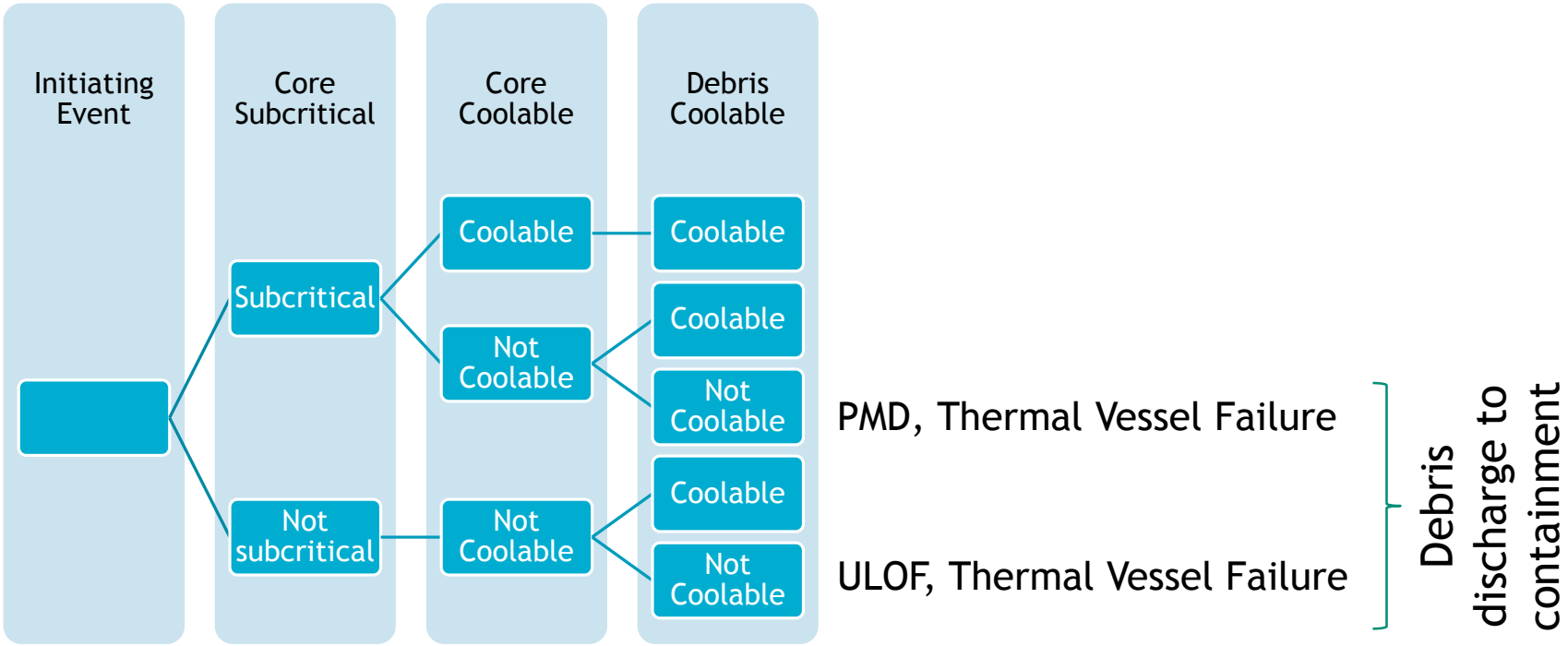
Source Term Evolution – Particle Release from Sodium Pool



Source Term Evolution – Behavior in Containment



SFR Source Term Scenarios



PMD – protected meltdown

- Core disruption occurs from subcritical state
- Primary system vessel fails thermally before core damage with salt draining into containment

ULOF – unprotected loss of flow

- Core disruption occurs before reactor shutdown due to loss of flow creating imbalance between heat removal and generation
- Vessel lower head thermally fails due to interaction with relocated debris
- Debris then sodium drains into containment



Conclusions

Overall Conclusions

The four-factor approach to MST allows for a similar process to be used for different reactor types

- Unavoidable differences in organization of each SME

Large, early releases were identified for each reactor in a way that allows for further examination and quantification of risk

- Typically lower frequency scenarios
- Necessary to consider within risk-informed framework

A number of scenarios with large release late in the event identified

- Characteristic of advanced reactor designs that progress to fuel damage very slowly for events initiated from subcritical state
- Advanced reactor concepts typically have large thermal inertia delaying progression to fission product release