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

Presentations for December 14, 2017 Public Meeting
Regulatory Improvements for Advanced Reactors

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

1) NRC Slides
   - Opening / Outline
   - Future Meetings

2) Licensing Modernization Project Slides on Defense in Depth

3) Discussion on higher assay low enriched uranium and fuel cycle (no slides; see ADAMS Accession No. ML17341A604)

4) Discussion on regulatory engagement plans (no slides, see ML17319A210)

5) NRC Slides on Functional Containment Performance Criteria

6) Slides on ASME Section III, Division 5
   - NRC Staff
   - DOE
   - ASME

7) Slides on potential issues related siting assessment related to populations
   - NRC Staff
   - ORNL

Advanced Reactor Program

U.S.NRC
United States Nuclear Regulatory Commission
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Public Meeting on Possible Regulatory Process Improvements for Advanced Reactor Designs

December 14, 2017

Advanced Reactor Program

Telephone Bridge
(888) 793-9929
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Opportunities for public comments and questions at designated times

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Outline

- Introductions
- Licensing Modernization White Paper
  - LMP; Discussion
  - Break @10:30
- Higher Assay LEU & Fuel Cycle
  - NEI; NRC Staff; Discussion
- Regulatory Engagement Plans
  - NEI; Discussion
  - As time allows, continuation of DiD and/or other topics
- *Lunch*
- ASME Code Section III, Division 5
  - NRC; DOE; ASME; Discussion
- Nuclear Plant Siting Considerations
  - NRC Staff; ORNL; Discussion
- Functional Containment Performance Criteria
  - NRC Staff; Discussion
- Other Topics
  - HTGR TWG Update on TRISO topical
  - Other Discussions
  - Future Meetings

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Licensing Modernization Project White Papers

NEI
Higher Assay LEU
&
Fuel Cycle

[White paper]

[11/2 NEI]

NEI
Regulatory Engagement Plan

11/2 NEI

Outline

NRC Staff (NRO)
ASME Section III Division 5

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ASME Section III Division 5

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ASME
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NRC Staff (NRO) Siting Considerations Draft White Paper (ML17333B158)

Advanced Reactor Program

ORNL
Siting Considerations

NRC Staff (NRO) Functional Containment Performance Criteria Draft White Paper (ML17334A155)

Other Topics

- HTGR TWG Limited Scope Topical on TRISO Fuel
- Other Topics
- Future meetings

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Future Stakeholder Meetings

Feb 1	Functional Containment Performance Criteria
	NIC - Fuel Cycle ?
Mar 22	NEI (Consolidated) RIPB Guidance
May 3	
Jun 14	

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ACRS Schedule (tentative)

Date	Committee	Topic
Feb 7	Sub	ARDC
Feb 23	Sub	Functional Containment
Mar	Full	ARDC
Apr	Full	Functional Containment
May 17	Sub	RIPB Guidance
Oct 30	Sub	RIPB Guidance

Public Comments / Questions

Utility-Led Initiative for Licensing Modernization of Technical Requirements for Licensing of Non-Light Water Reactors

Defense-in-Depth Adequacy

Ed Wallace, Karl Fleming

December 14, 2017

Meeting Purpose

- Provide initial overview of LMP Defense- in-Depth framework and approach including:
  - Plant Capability DID description and use
  - Programmatic Capability DID description and use
  - Integration with design processes
  - Integrated Evaluation of DID Adequacy
  - DID Baseline development

Purpose of DID Paper

- Utilize the existing defense-in-depth (DID) philosophy to define a framework and evaluation process for establishing DID and evaluating DID attributes for advanced reactors
- Describe a process that includes an approach for the incorporation of DID protective measures into the plant design and a structured method for the evaluation of DID adequacy
- Describe how the DID evaluation process integrates with the LMP RIPB approaches for design development, PRA, LBE selection and evaluation and SSC safety classification
- When implemented, the LMD DID framework provides a more objective means to answer the question for a specific design: "When is enough, enough?"

Foundation Documents

- NRC DID Philosophy
- NRC NUREG KM-009 Historical Review and Observations of Defense-in-Depth
- Next Generation Nuclear Plant Defense-in-Depth Approach, INL/EXT 09-17139, December 2009
- Draft LMP White Papers on PRA, LBE Selection and SSC Classification

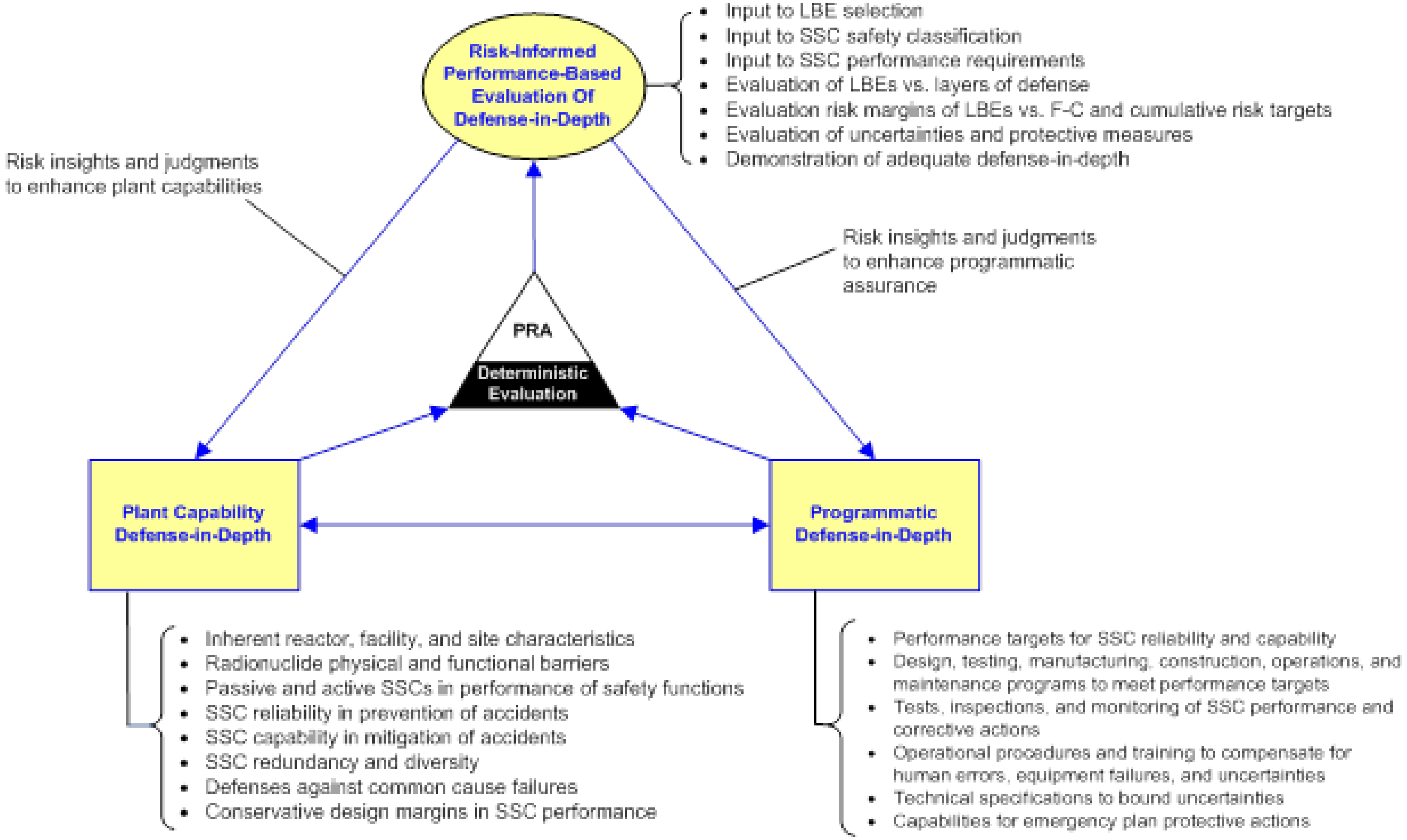
Framework General Objectives

- Systematic and reproducible
- Sufficiently complete
- Available for timely input to design decisions
- Risk-informed and performance-based
- Reactor technology inclusive
- Compatible with applicable regulatory requirements

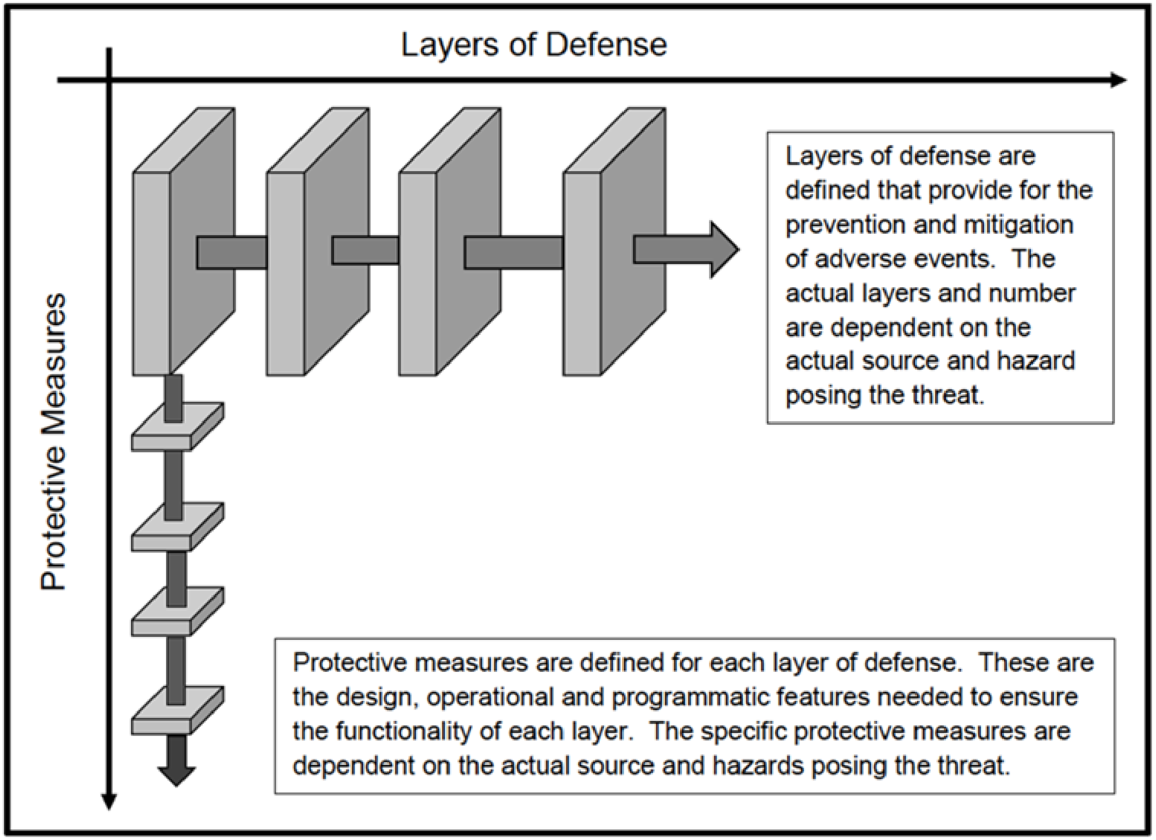
Basic Definitions

- **Plant Capability Defense-in-Depth**
  - Plant Functional Capability DID- This is the capability of systems and features designed to prevent occurrence of undesired licensing basis events (LBE) or mitigate the consequences of such events.
  - Plant Physical Capability DID- This capability is introduced through SSC robustness, design margins, and physical barriers to limit the consequences of a radionuclide release
- **Programmatic Defense-in-Depth**
  - Used to address uncertainties when evaluating Plant Capability DID and where programmatic protective strategies provide additional confidence of plant performance for life of the plant.
  - Used to incorporate special treatments in design, construction, testing, operations and maintenance
- **Risk-Informed Evaluation of Defense-in-Depth**
  - Provides a systematic, holistic, integrated and transparent process for examining the DID adequacy achieved by the combination of Plant Capability and Programmatic elements.

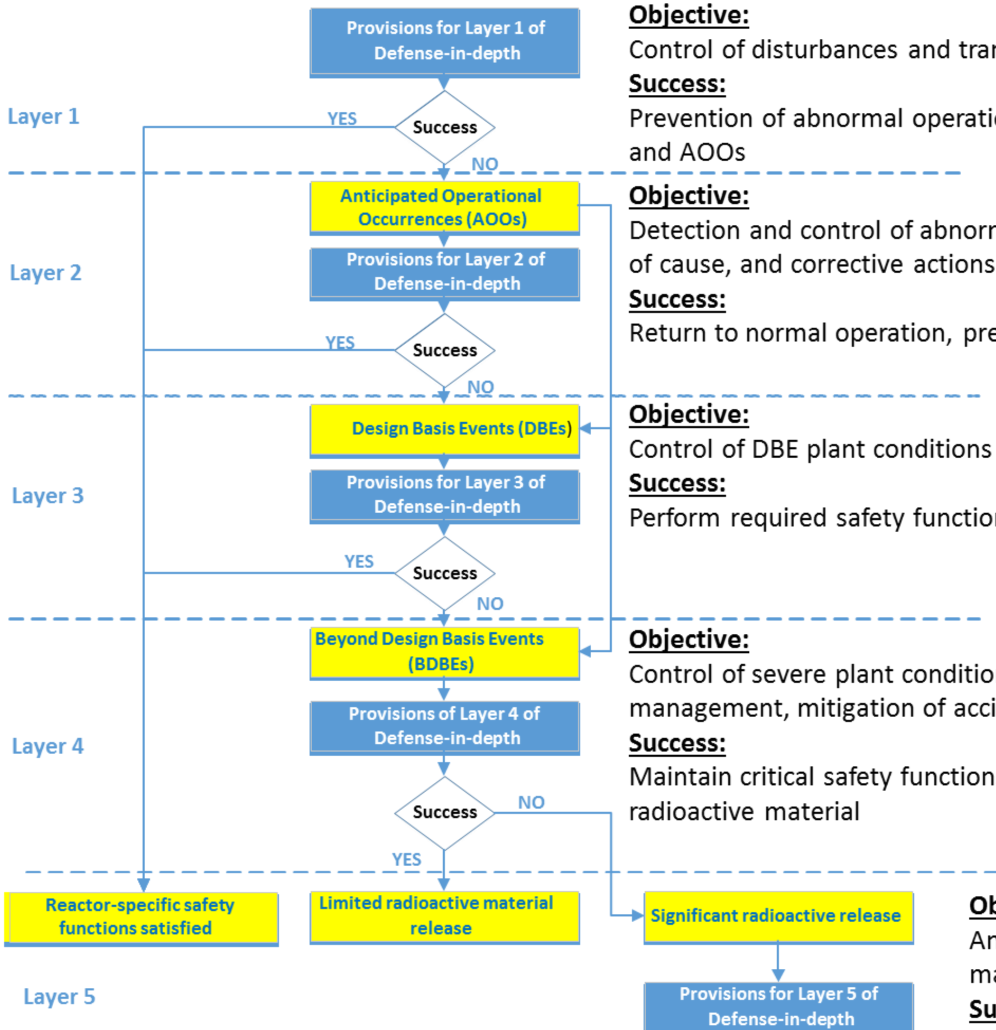
Builds on NGNP Approach to DID



Embracing "Layers of Defense" Concept from NUREG/KM-0009



Layer Concept Adapted from IAEA



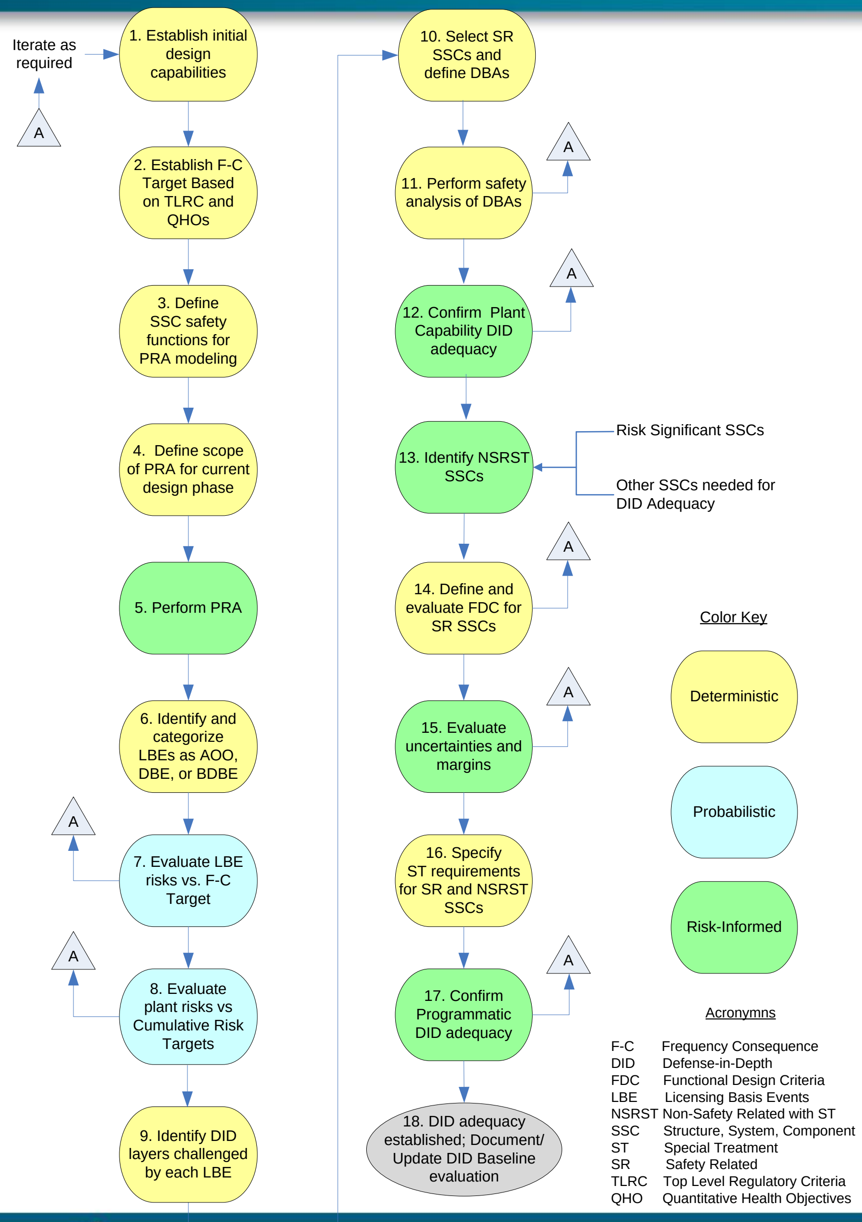
Objective:
Control of disturbances and transients
Success:
Prevention of abnormal operation, initiating events, and AOOs

Objective:
Detection and control of abnormal operation and AOOs, identification of cause, and corrective actions
Success:
Return to normal operation, prevention of DBEs.

Objective:
Control of DBE plant conditions within the design basis
Success:
Perform required safety functions, prevent BDBEs

Objective:
Control of severe plant conditions, including on-site accident management, mitigation of accident consequences
Success:
Maintain critical safety functions for the retention of radioactive material

Objective:
Anticipatory emergency planning and off-site accident management to mitigate accident doses to the public
Success:
Prevent adverse public health and safety impacts



Integrated Process for Design, PRA, LBE, SSC and DID

- Steps reflect information logic, not serial requirements
- Iterative process as design matures
- Baseline DID adequacy determination when all steps satisfactorily address

Plant Capability DID Attributes (Box 12)

Attribute	Evaluation Focus
Initiating Event and Accident Sequence Completeness	PRA Documentation of Initiating Event Selection and Event Sequence Modeling
	Insights from reactor operating experience, system engineering evaluations, expert judgment
Layers of Defense	Multiple Layers of Defense
	Extent of Layer Functional Independence
	Functional Barriers
	Physical Barriers
Functional Reliability	Inherent Reactor Features that contribute to performing safety functions
	Passive and Active SSCs performing safety functions
	Redundant Functional Capabilities
	Diverse Functional Capabilities
Prevention and Mitigation Balance	SSCs performing prevention functions
	SSCs performing mitigation functions
	No Single Layer /Feature Exclusively Relied Upon

Guidelines for Plant Capability DID Adequacy

<table>
<tr><th rowspan="2">Layer[1]</th><th colspan="2">Layer Guideline</th><th colspan="2">Overall Guidelines</th></tr>
<tr><th>Quantitative</th><th>Qualitative</th><th>Quantitative</th><th>Qualitative</th></tr>
<tr><td>1) prevent off-normal operation and AOOs</td><td colspan="2">Maintain frequency of plant transients within designed cycles; meet user requirements for plant reliability and availability[2]</td><td rowspan="5">Meet F-C target for all LBEs and cumulative risk metric targets with substantial[4] margins</td><td rowspan="5">No single design or operational feature[3], no matter how robust, is exclusively relied upon to satisfy the five layers of defense.</td></tr>
<tr><td>2) control abnormal operation, detect failures, and prevent DBEs</td><td>Maintain frequency of all DBEs < 10^{-2}/ plant-year</td><td>Minimize frequency of challenges to safety related SSCs</td></tr>
<tr><td>3) control accidents within the analyzed design basis conditions and prevent BDBEs</td><td>Maintain frequency of all BDBEs < 10^{-4}/ plant-year</td><td>No single design or operational feature[3] relied upon to meet quantitative objective for all DBEs</td></tr>
<tr><td>4) control severe plant conditions, mitigate consequences of accidents</td><td rowspan="2">Maintain individual risks from all LBEs < QHOs with sufficient[4] margins</td><td rowspan="2">No single barrier[3] or plant feature relied upon to limit releases in achieving quantitative objective for all BDBEs</td></tr>
<tr><td>5) deploy adequate offsite protective actions and prevent adverse impact on public health and safety</td></tr>
</table>

Acronyms:

AOO	Anticipated Operational Occurrence
DBE	Design Basis Event
BDBE	Beyond Design Basis Event
QHO	Quantitative Health Objective
F-C	Frequency-Consequence

Notes:

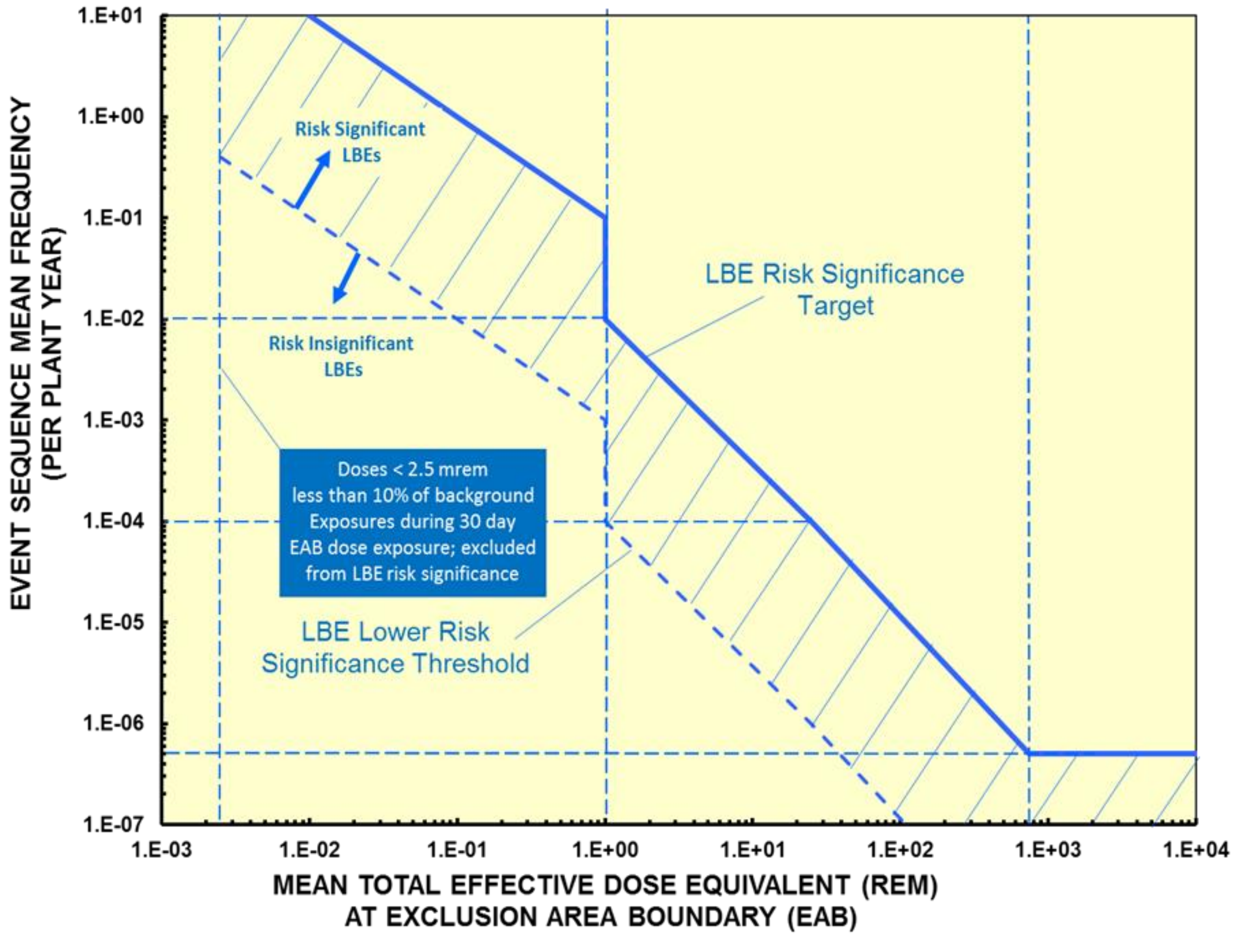
[1] The plant design and operational features and protective strategies employed to support each layer should be functionally independent

[2] Non-regulatory user requirements for plant reliability and availability and design targets for transient cycles should limit the frequency of initiating events and transients and thereby contribute to the protective strategies for this layer of defense-in-depth. Quantitative and qualitative targets for these parameters are design specific.

[3] This criterion implies no excessive reliance on programmatic activities or human actions and that at least two independent means are provided to meet this objective.

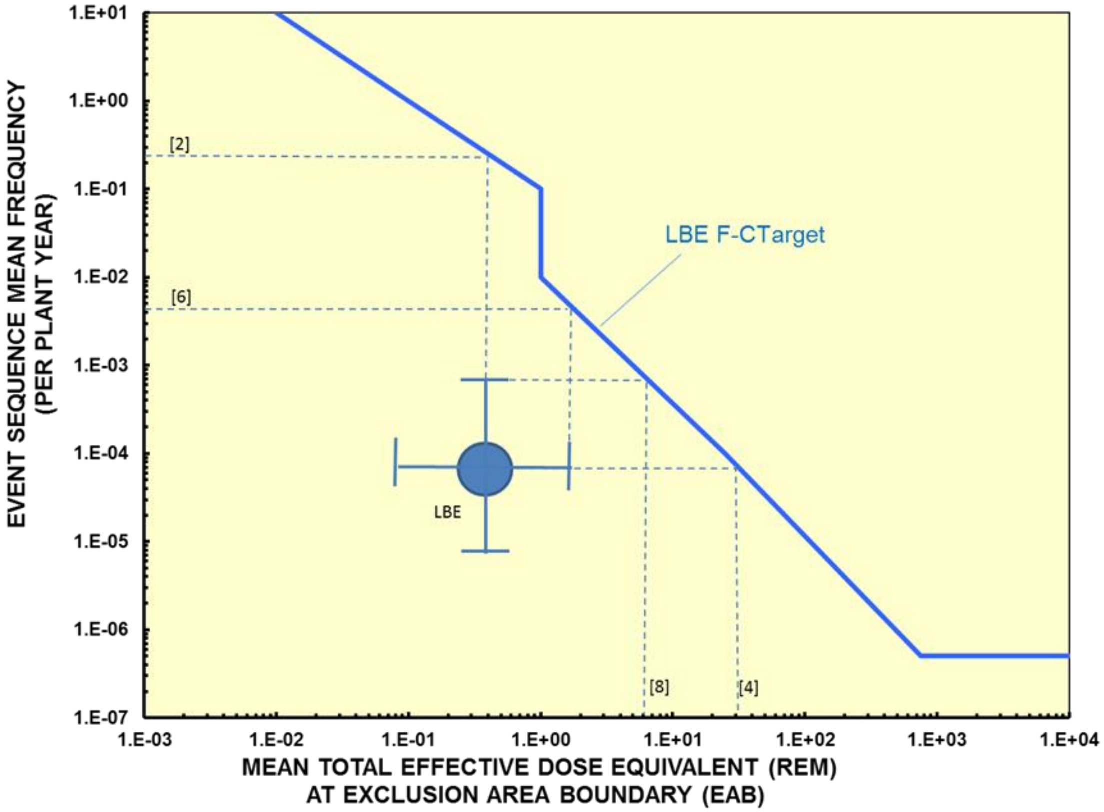
[4] The level of margins between the LBE risks and the QHOs provides objective evidence of the plant capabilities for defense-in-depth as to be decided by the Integrated Decision Panel.

LBE Frequency-Consequence Target Framework



- Used to define risk significant LBEs
- Risk significant criteria derived from PRA standards
- Absolute metrics preferred to relative metrics

Risk Margin Definition



- Risk margins defined for F-C target and cumulative risk targets
- Margins defined based on mean and upper 95%tile frequencies and doses
- Risk margins are indicative of DID adequacy and demonstration of larger safety margins

Margin Analysis Summary

LBE Category	Limiting LBE[1] Name	Limiting LBE[1] Mean Freq./plant-yr.	Limiting LBE[1] Mean Dose (Rem)	F-C Target Freq. at LBE Dose/plant-yr. [2]	F-C Target Mean Frequency Margin[3]	F-C Target Dose at LBE Freq.(Rem) [4]	F-C Target Dose Margin[5]
AOO	AOO-5	4.00E-02	2.50E-04	4.00E+02	1.00E+04	1.00E+00	4.00E+03
DBE	DBE-10	1.00E-02	2.00E-03	6.00E+01	6.00E+03	1.00E+00	5.00E+02
BDBE	BDBE-2	3.00E-06	4.00E-03	2.50E+01	8.30E+06	2.50E+02	6.00E+04

[1] Limiting LBE is LBE with highest risk significance in LBE Category

[2] Frequency value measured at the LBE mean Dose level from the F-C target, See [2] in Figure 2-11

[3] Ratio of the frequency in note [2] to the LBE mean frequency, mean frequency margin

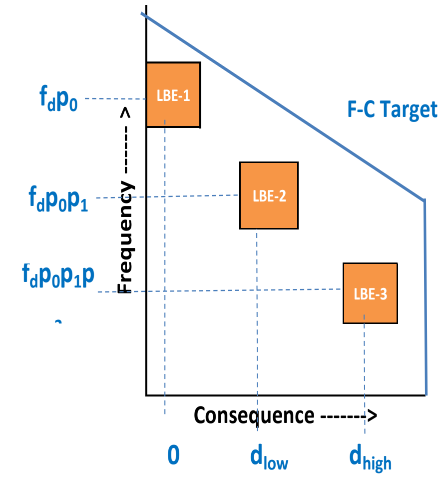
[4] Dose value measured at the LBE mean frequency from the F-C target, See [4] in Figure 2-11

[5] Ratio of the Dose in Note [4] to the LBE mean dose, Mean Dose Margin

Evaluation of SSC Functions and Layers of Defense

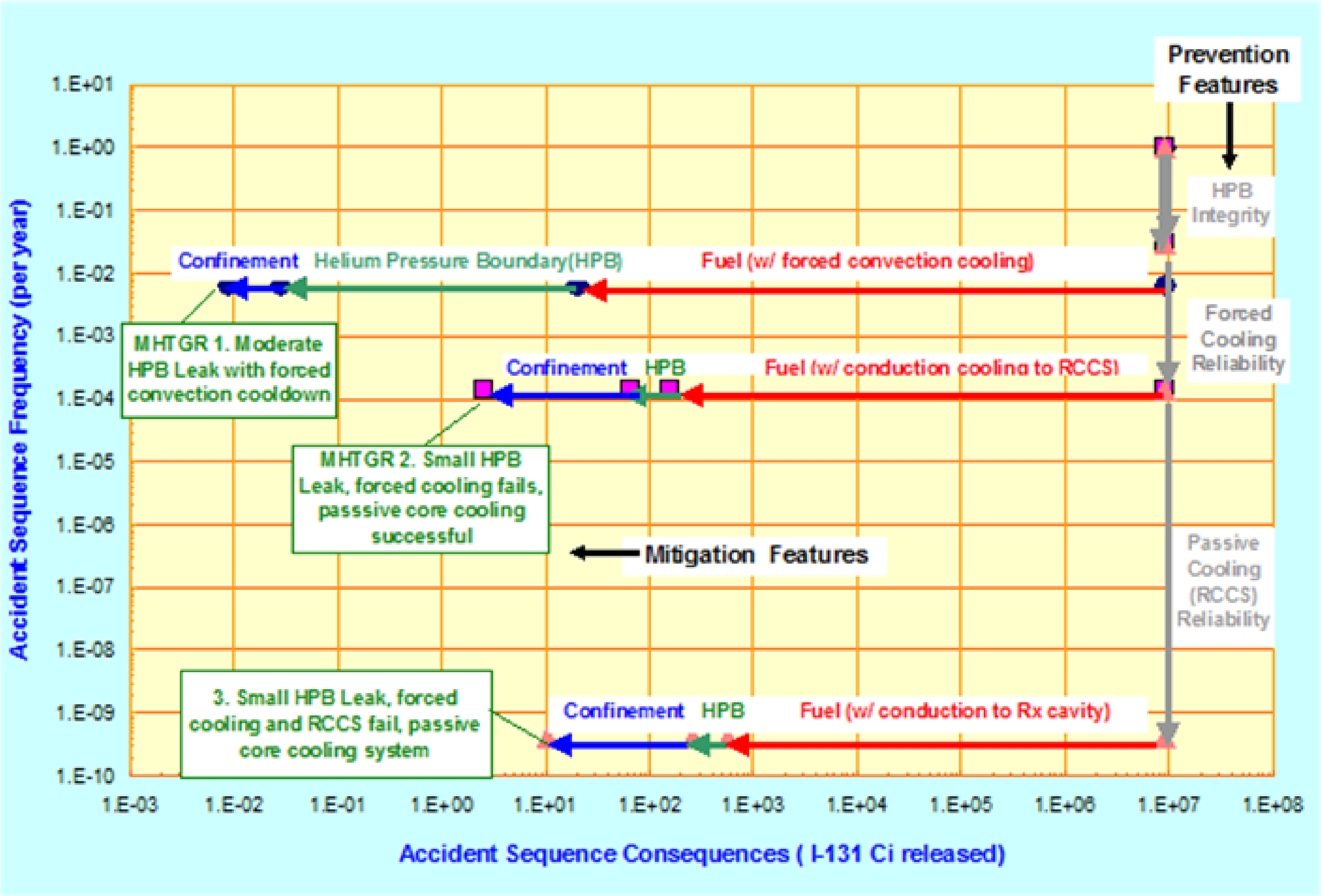
Plant Distrubance	Plant features prevent Inititating event?	SSC$_1$ Prevents Fuel Damage?	SSC$_2$ Limits Release?	LBE	End State	Defense-in-Depth Layers Challenged [1]	Frequency	Dose
f_d	*Yes*			N/A	Disturbance controlled with no plant trip	Layer 1	f_d	0
	p_0 *No*	*Yes*		1	No fuel damage or release	Layer 2	$f_d p_0$	0
		p_1 *No*	*Yes*	2	Fuel damage w/ limited release	Layer 3	$f_d p_0 p_1$	d_{low}
			p_2 *No*	3	Fuel Damage w/ un-mitigated release	Layers 4 and 5	$f_d p_0 p_1 p_2$	d_{high}

[1] See Figure 2-4 for definition of defense-in-depth layers



SSC	LBEs	Function	SSC Performance Attribute for Special Treatment
Plant	N/A	Prevent initiating event	Reliability of plant features preventing initiating event
SSC$_1$	1	Mitigate initiating event	Capability to prevent fuel damage
	2	Prevent fuel damage	Reliability of mitigation function
	3	Help prevent large release	Reliability of mitigation function
SSC$_2$	2	Mitigate fuel damage	Capability to limit release from fuel damage
	3	Prevent unmitigated release	Reliability of mitigation function

Example Evaluation of Prevention and Mitigation Balance for MHTGR LBEs



Scenario Logic for Evaluating Prevention and Mitigation (MHTGR example)

Standard Elements of Accident Sequence	Design Features Contributing to Prevention	Design Features Contributing to Mitigation
Initiating event occurrence	Reliability of SSCs supporting power generation reduces the initiating event frequencies	Plant Capabilities of normally operating systems to continue operating during disturbances to prevent initiating events
Response of active SSCs supporting safety functions:	Reliability and availability of active SSCs reduce sequence frequency; successful operation of these SSCs prevents the sequence	Capabilities of active successful SSCs including design margins reduce the impacts of the initiating events and challenges to barrier integrity.
Response of passive features supporting safety functions	Reliability and availability of passive SSCs reduce sequence frequency; successful operation of these SSCs prevents the sequence	Capabilities of passive successful SSCs including design margins reduce the impacts of the initiating events and challenges to barrier integrity.
Fuel release fraction	None	Inherent and passive capabilities of the limit the release from the fuel.
Coolant boundary release fraction	None	Inherent and passive capabilities of the pressure boundary and the capabilities of the fuel limit the release from the pressure boundary.
Reactor building release fraction	None	Inherent and passive capabilities of the reactor building barrier and the capabilities of the fuel and coolant pressure boundary limit the release from the reactor building barrier.
Time to implement emergency plan protective actions.	None	Inherent and passive features and capabilities of the fuel, coolant pressure boundary, and reactor building barrier including design margins dictate the time available for emergency response.

Qualitative Plant Capability Evaluation Supported by Attribute Topical Questions

The evaluation of LBEs by the IDP will focus on the following questions:

- Is the selection of initiating events and event sequences reflected in the LBEs sufficiently complete? Are the uncertainties in the estimation of LBE frequency, plant response to events, mechanistic source terms, and dose well characterized?
- Are there sources of uncertainty not adequately addressed?
- Have all risk significant LBEs and SSCs been identified?
- Has the PRA evaluation provided an adequate assessment of "cliff edge effects"?
- Is the technical basis for identifying the required safety functions adequate?
- Is the selection of the safety related SSCs to perform the require safety functions appropriate?
- Have protective measures to manage the risks of multi-module and multi-radiological source accidents been adequately defined
- Have protective measures to manage the risks of all risk significant LBEs been identified, especially those with relatively high consequences.
- Have protective measures to manage the risks for all risk significant common cause initiating events such as support system faults, internal plant hazards such as fires and floods, and external hazards been identified?
- Is the risk benefit of all assigned protective measures well characterized, e.g. via sensitivity analyses?

Programmatic DID Attributes

Attribute	Evaluation Focus
Quality / Reliability	Performance targets for SSC reliability and capability
	Design, manufacturing, construction, operations and maintenance features or special treatment sufficient to meet performance targets
Compensation for Uncertainties	Compensate for Human Errors
	Compensate for Mechanical Errors
	Compensate for Unknowns
Off-Site Response	Emergency Response Capability

Guidelines for Programmatic DID Adequacy

The adequacy of Programmatic DID is based on meeting the following objectives:

- Assuring adequate margins exist between the assessed LBE risks relative to the F-C Target including quantified uncertainties
- Assuring adequate margins exist between the assessed total plant risks relative to the Cumulative Risk Targets
- Assuring appropriate targets for SSC reliability and performance capability are reflected in design and operational programs for each LBE
- Providing adequate assurance that the risk, reliability, and performance targets will be met and maintained throughout the life of the plant with adequate consideration of sources of significant uncertainties

Qualitative Programmatic Capability Evaluation Supported by Attribute Topical Questions (Example)

Attribute	Evaluation Focus	Implementation Strategies	Evaluation Considerations
Compensation for Uncertainties	Compensation for Unknowns (Performance Variability)	Operational Technical Specifications In-Service Monitoring Programs	1. Are the Technical Specification for risk-significant SSCs consistent with achieving the necessary safety function outcomes for the risk significant LBEs? 2. Are the in-service monitoring programs aligned with the risk-significant SSC identified through the RIPB SSC Classification process?
	Compensation for Unknowns (Knowledge Uncertainty)	Site Selection PIRT/ Technical Readiness Levels Integral Systems Tests / Separate Effects Tests	1. Have the uncertainties identified in PIRT or similar evaluation processes been satisfactorily addressed with respect to their impact on plant capability and associated safety analyses? 2. Has physical testing been done to confirm risk significant SSC performance within the assumed bounds of the risk and safety assessments? 3. Have plant siting requirements been conservatively established based on the risk from severe accidents identified in the PRA? 4. Has the PRA been peer reviewed in accordance with applicable industry standards and regulatory guidance? 5. Are hazards not included in the PRA low risk to the public based on bounding deterministic analysis?

Risk-Informed and Performance-Based Decision Attributes

Attribute	Evaluation Focus
Use of Risk Triplet Beyond PRA	What can go wrong?
	How likely is it?
	What are the consequences?
Knowledge Level	Plant Simulation and Modeling of LBEs
	State of Knowledge
	Margin to PB Limits
Uncertainty Management	Magnitude and Sources of Uncertainties
Action Refinement	Implementation Practicality and Effectiveness
	Cost/Risk/Benefit Considerations

Results of Integrated Decision Panel Evaluation

- Plant Capability DID is deemed to be adequate
  - Plant Capability DID guidelines are satisfied
  - Review of LBEs is completed with satisfactory results
    - Risk margins against F-C Target and Cumulative Risk Targets are sufficient
    - Role of SSCs in the prevention and mitigation at each layer of defense challenged are understood
    - Prevention/mitigation balance sufficient
    - Classification of SSCs into SR, NSRST, and NST is appropriate
    - Risk significance classification of LBEs and SSCs are appropriate
    - Independence among design features at each layer of defense is sufficient
- Programmatic DID is deemed to be adequate
  - Programmatic DID guidelines are satisfied
  - Performance targets for SSC reliability and capability are established
  - Source of uncertainty in selection and evaluation of LBE risks are identified
    - Completeness in selection of initiating events and event sequences is sufficient
    - Uncertainties in the estimation of LBE frequencies are evaluated
    - Uncertainties in the plant response to events are evaluated
    - Uncertainties in the estimation of mechanistic source terms are evaluated
  - Special Treatment for all SR and NSRST SSCs is sufficient

Plant Capability Baseline Summary

LBE IE Series Name	Functional			Physical	
	Margin Adequacy	Multiple Protective Measures	Prevention and Mitigation Balance	Functional Reliability	No Single Feature Relied Upon
Normal Operation	√			√	
AOOs	√			√	
DBEs	√	√	√	√	√
BDBEs	√	√	√	√	√
DBAs	√	√	√	√	√

NEI
NUCLEAR ENERGY INSTITUTE
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Programmatic Baseline Summary*

LBE IE Series Name	Quality/Reliability Design, Manufacturing, Construction, Operations and Maintenance	Compensation for Uncertainties			Offsite Response Emergency Response Capability
		Human Errors	Mechanical Failures	Unknowns	
Normal Operation	√	√	√	√	
AOOs	√	√	√	√	
DBEs	√	√	√	√	√
BDBEs	√	√	√	√	√
DBAs	√	√	√	√	√

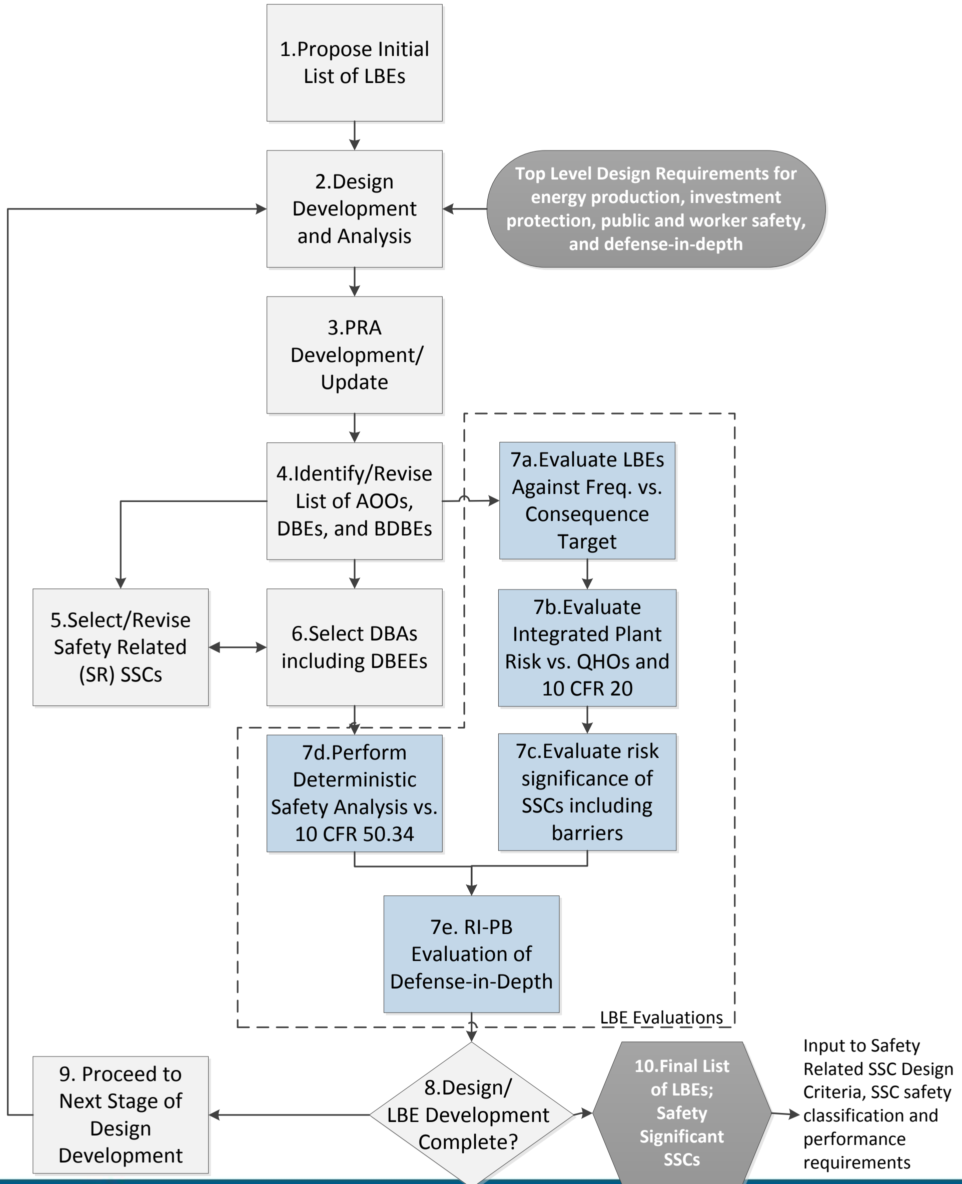
*Note – the qualitative nature of the programmatic DID evaluation results in a collective judgment on the degree of satisfaction of various DID attributes. Statements of considerations are included in the guidance to aid arriving at the final integrated decision of DID adequacy. The baseline documentation then supports future change management evaluations of the impacts on DID.

Summary

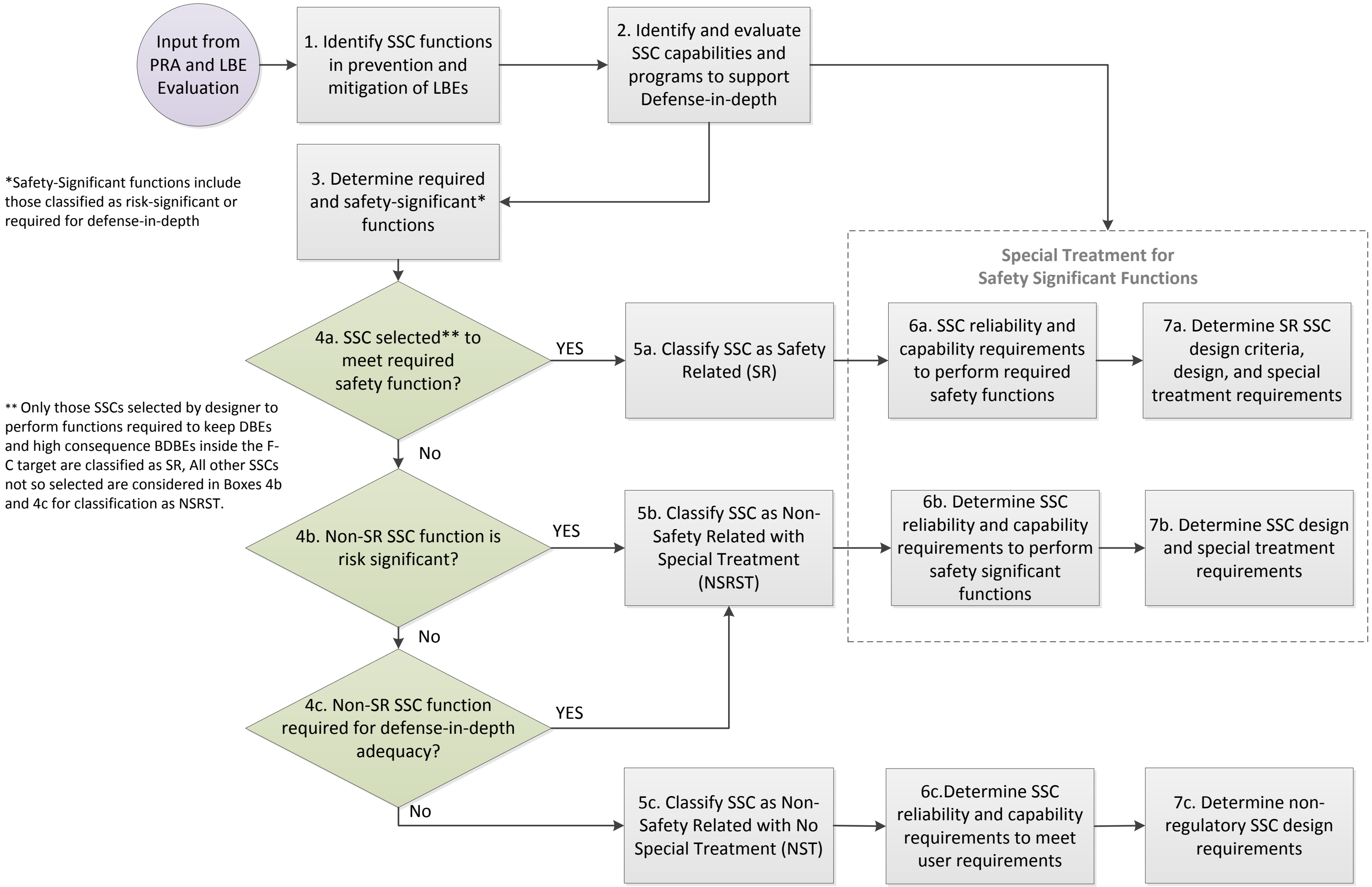
- DID paper objective is to describe DID attributes based on the DID philosophy and history
- Develop a RIPB process for establishing and evaluating DID adequacy building on the NGNP approach
- DID framework comprised of plant capability DID, programmatic DID and risk informed evaluation of DID
- For advanced reactors, the objective is to build DID into the design systematically and apply integrated decision making to the quantitative and qualitative information derived from early application of PRA to achieve a RIPB design
- Document the DID baseline to provide an referenced means of evaluating plant changes and state of knowledge changes throughout plant lifecycles

BACKUP SLIDES

LBE Selection and Evaluation Process



SSC Classification and Performance Requirements Process



Functional Containment Performance Criteria

Draft White Paper – ML17334A155

December 14, 2017

Format

- Paper (Policy)
  - Enclosure 1 – Background
  - Enclosure 2 – Risk Informed Performance Based Performance Criteria
- Background (Enclosure 1)
  - “Functional containment”
  - Performance criteria
  - Note that Enclosure 1 is under development and was not included in the draft white paper

Format

- Enclosure 2 & Paper Summary
- Need to resolve
  - Goal to better align regulatory and design/development processes
  - Increased number & diversity of advanced reactor designs
- Proposed Approach
  - Risk informed, performance based
  - Aligned to overall framework being developed (Enclosure 2)

Implementation Action Plans

- Strategy 3: Develop guidance for a flexible non-LWR regulatory review process …
- Strategy 5: Identify and resolve technology-inclusive policy issues…
- Contributing activities include:
  - Establish and document the criteria necessary to reach a safety, security, or environmental finding for non-LWR applicant submissions. The criteria and associated regulatory guidance are available to all internal and external stakeholders.
  - Determine and document appropriate non-LWR licensing bases and accident sets for highly prioritized non-LWR technologies.
  - Identify, document and resolve (or develop plan to resolve) current regulatory framework gaps for non-LWRs.
  - Analyze and resolve technology-inclusive non-LWR policy issues

Discussion

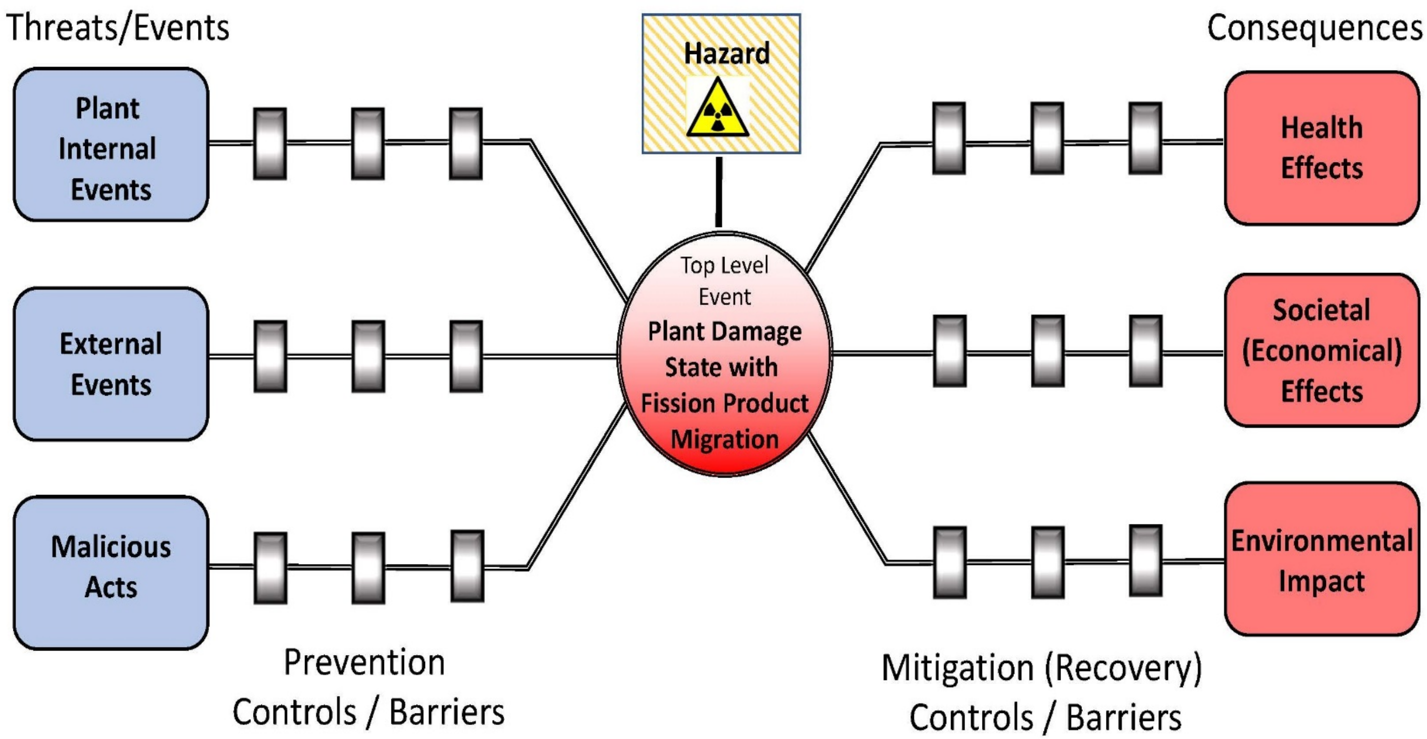
- Design features for radionuclide retention
  - *Need alignment on terminology*

- Interactions related to Next Generation Nuclear Plant (NGNP) and lessons learned from 9/11 and Fukushima

- Integrated approach to considering risks and ensuring appropriate measures to prevent or mitigate events

- Bow Tie Diagram

Bow Tie Diagram



** Need alignment on terminology / approach (e.g., top level event)*

Advanced Reactor Design Criteria

DG-1330, "Guidance for Developing Principal Design Criteria for Non-Light Water Reactors," resulted in the following design criterion and supporting rationale for "functional containment" for modular high-temperature gas-cooled reactors:

Containment design.

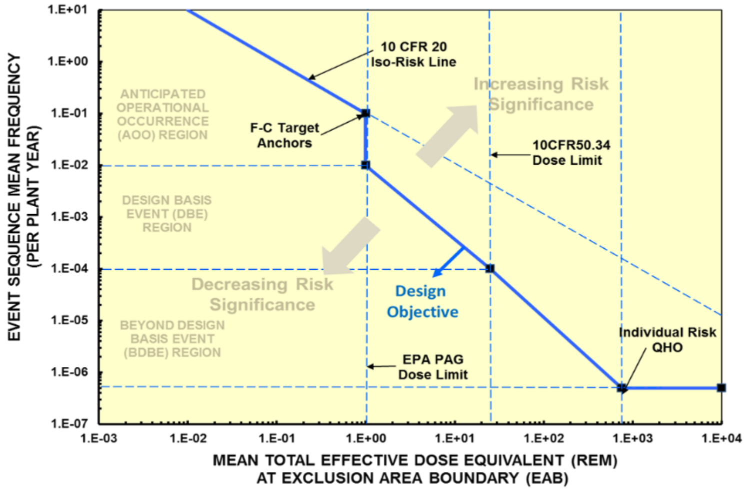
A reactor functional containment, consisting of multiple barriers internal and/or external to the reactor and its cooling system, shall be provided to control the release of radioactivity to the environment and to ensure that the functional containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Rationale

The term "functional containment" is applicable to advanced non-LWRs without a pressure retaining containment structure. A functional containment can be defined as "a barrier, or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, AOOs, and accident conditions."

Basic Framework

- The basic framework is built around the identification and categorization of licensing-basis events.



The figure is being provided to illustrate the general organization of events but the staff is not ready to request Commission-level decisions on the specifics within the figure. The staff is continuing to interact with stakeholders to reach alignment on some topics such as the demarcation of categories and ensuring consistency across the assessments of prevention and mitigation controls and barriers for various events and consequences.

Event Categories

- Normal operations define initial conditions for licensing basis events. Radiological doses resulting from normal operation are controlled by limiting routine effluent releases to below regulatory requirements (i.e., Part 20 limits)
- Anticipated Operational Occurrences (AOOs) encompass planned and anticipated events (e.g., frequencies exceed approximately 10^{-2} per plant-year). The radiological doses from AOOs are required to meet a fraction of the normal operation public dose requirements (i.e., Part 20 limits) which are established for annual dose rates due to both events and planned effluent releases. AOOs are used to set operating limits for normal operation modes and states. Design features and programmatic controls are established to limit AOO frequencies and consequences in terms of offsite doses and success of preventive controls and barriers (e.g., integrity of fuel cladding or coatings).

Event Categories

- Design Basis Events (DBEs) encompass unplanned off-normal events not expected in the plant's lifetime, but which might occur in the lifetimes of a fleet of plants (i.e., event frequencies in the range of 10^{-4} to 10^{-2} per plant-year). The radiological doses from DBEs are required to be a fraction of accident public dose requirements (e.g., 10 CFR 50.34) as shown on the sliding illustrative F-C target in Figure 2. Design features and programmatic controls are established to limit DBE frequencies and consequences in terms of offsite doses and success of preventive controls and barriers (e.g., integrity of fuel cladding or coatings). The identification and evaluation of DBEs provide input to the selection of design basis accidents (DBAs) discussed below.

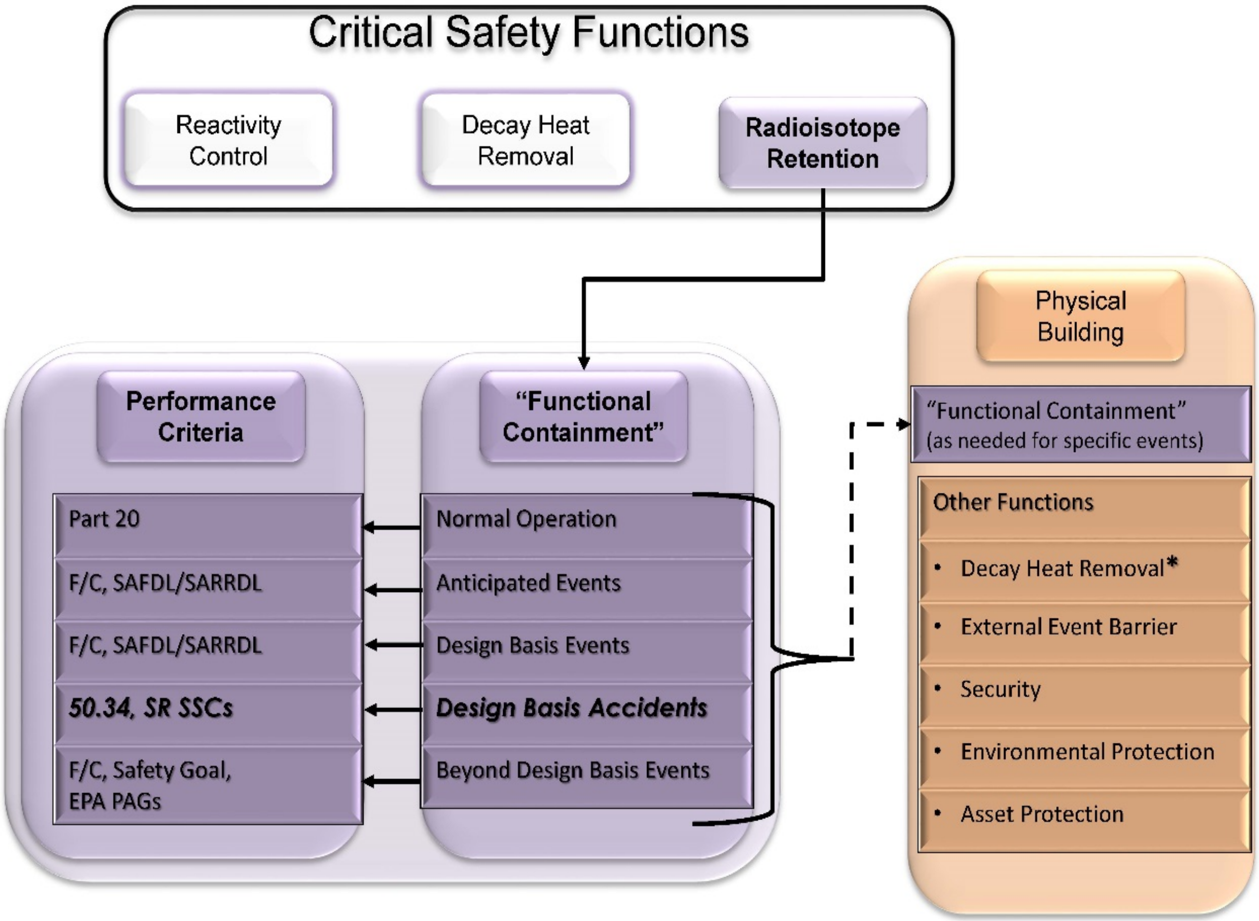
Event Categories

- Beyond Design Basis Events (BDBEs) are rare off-normal events whose frequencies range from a very low value (e.g., approximately 10^{-7} or 10^{-8} per plant-year to 10^{-4} per plant-year. BDBEs are evaluated to ensure that they do not pose an unacceptable risk to the public and to provide input to the selection of DBAs. Design features and programmatic controls are established to limit BDBE frequencies and consequences in terms of offsite doses and success of preventive barriers (e.g., integrity of fuel cladding or coatings) or mitigation barriers (e.g., severe accident design features).

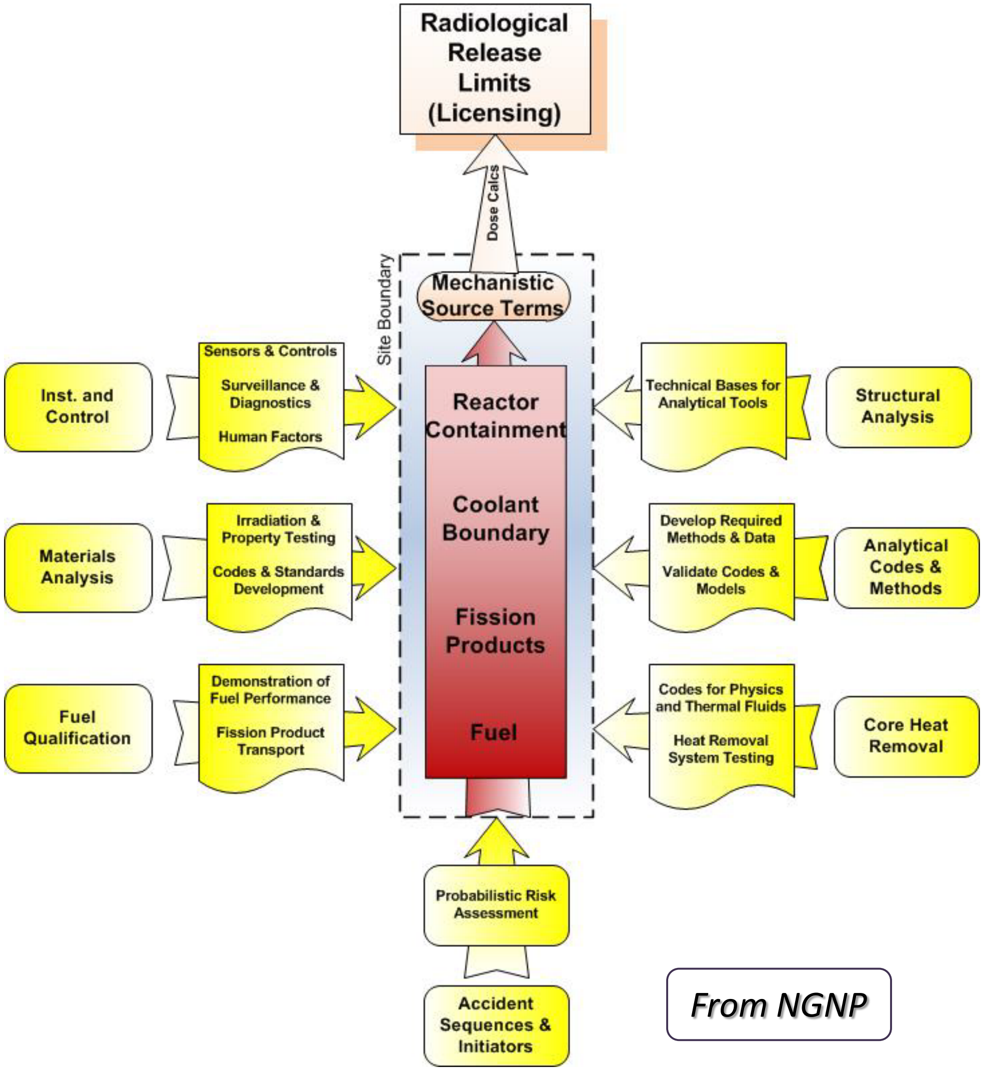
Event Categories

- Design Basis Accidents (DBAs) are the safety analysis report Chapter 15, “Accident Analyses,” which are prescriptively derived from the DBEs by assuming that only SSCs classified as safety-related are available to deal with the event. The public consequences of DBAs are conservatively calculated and assessed against 10 CFR 50.34 limits, similar to DBAs analyses for existing LWRs. DBAs have historically been used to define safety margins for SSCs and establish limiting conditions for operation.

Methodology to Identify Performance Criteria



Key Input for Licensing



From NGNP

Proposed Approach

- Performance Criteria
  - Established for event categories
  - SSC performance criteria established based on their role in meeting event category perf criteria
  - Consistent with current use of “design basis” for SSCs

For each event category, performance criteria would define specific functions to be performed by a structure, system, or component (SSC) of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. The design of each SSC would be determined based on the aggregation of performance requirements for each event category and critical safety function as well as other potential roles that a designer may choose for that SSC.

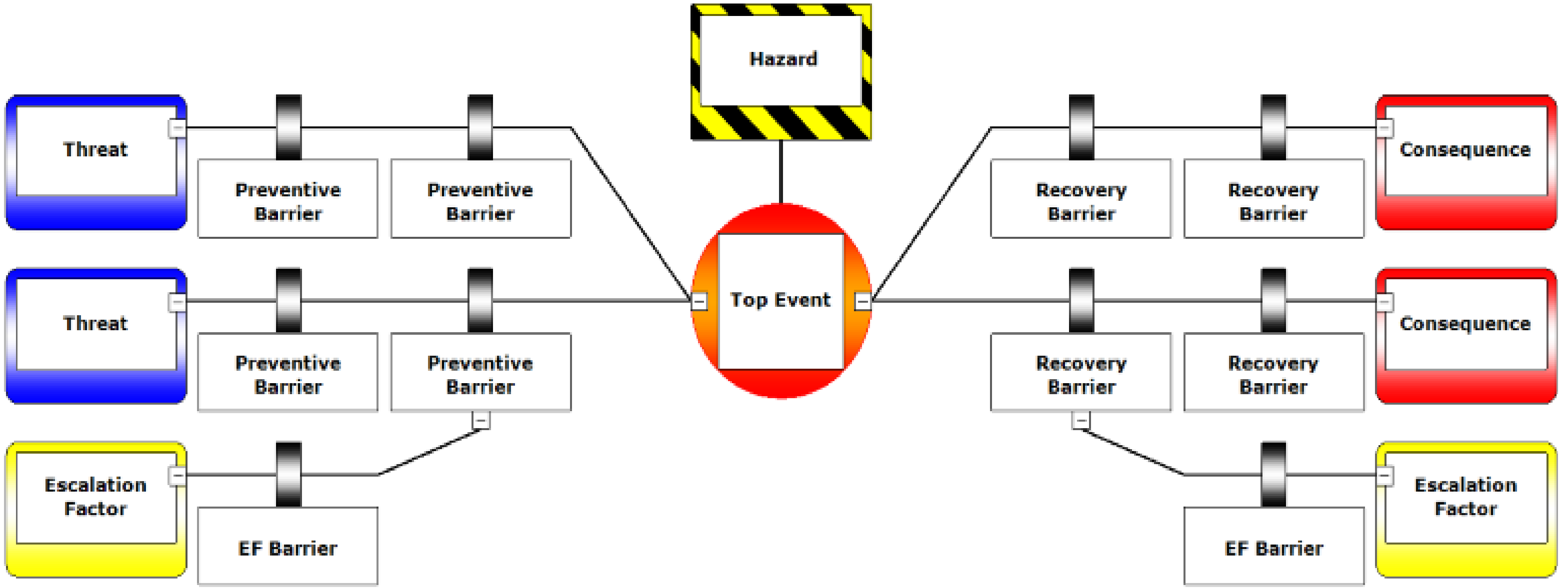
Physical Structure Functions

- Possible functions other than radionuclide retention
  - Structural support to primary cooling systems;
  - Supporting the decay heat removal critical safety function via structural support for and housing of backup or emergency cooling such as reactor cavity cooling systems;
  - Prevention barrier against external events such as flooding and wind loadings;
  - Design feature credited in aircraft impact assessments;
  - Physical security design feature credited in preventing or delaying adversaries; and
  - Design feature credited during environmental assessments of severe accident mitigation design alternatives.

Path Forward

- Discussion and next steps
  - Spring 2018 – Commission Paper
  - April 2018 – ACRS Full Committee
  - February 23, 2018 – ACRS Subcommittee
  - February 1, 2018 – Stakeholder Meeting
  - Conference Call(s)
  - December 14, 2017 – Stakeholder Meeting

Backup Slide



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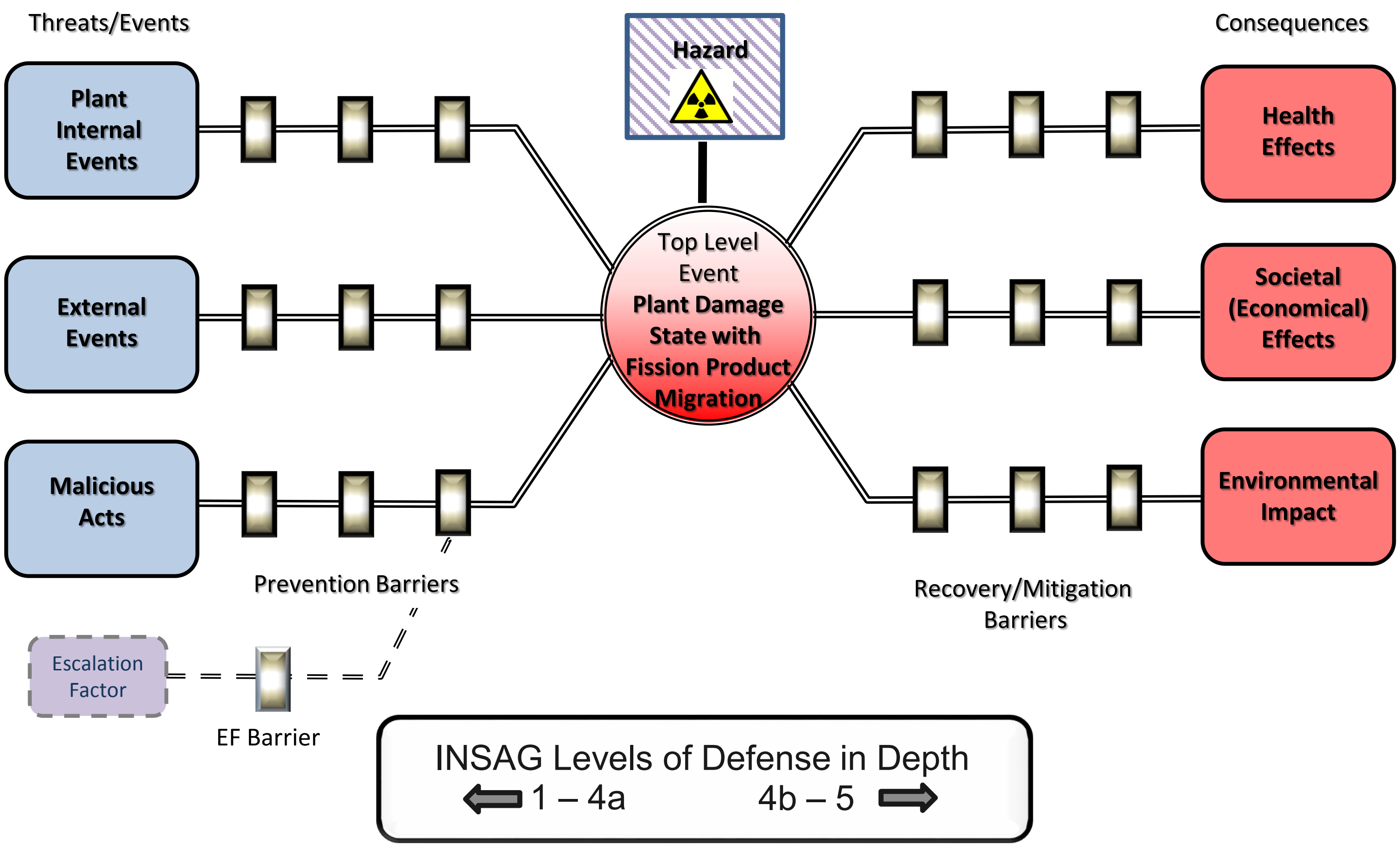


Current NRC Perspective on the Endorsement of ASME Section III, Division 5, High Temperature Reactors

Advanced Reactor Stakeholder Meeting

12/14/17

Matthew Mitchell, NRO

Recognize the importance of ASME Section III, Division 5

- NRC staff participate in multiple ASME Code Committees involved in the development of ASME Section III, Division 5
- Staff appreciate the work done by industry, DOE, and other representatives in developing ASME Section III, Division 5 to date

ASME Section III, Division 5 Endorsement Concerns

- Endorsement objective is to achieve regulatory efficiency and effectiveness
- However, endorsement will be resource intensive
  - Hundreds of pages of Code to review
- Review could impact the availability of resources to address other advanced reactor framework development topics
- Staff expect to review as an “acceptable method of meeting the regulations” rather than as a requirement and identify conditions on its use

Industry Input

- NRC needs industry input in order to make a recommendation about initiating the endorsement process
- NRC Questions:
  - Does ASME Section III, Division 5, with minimal need for exceptions, support the designs being developed for potential NRC review?
  - What additional improvements could be made to ASME Section III, Division 5 to support prospective vendors?

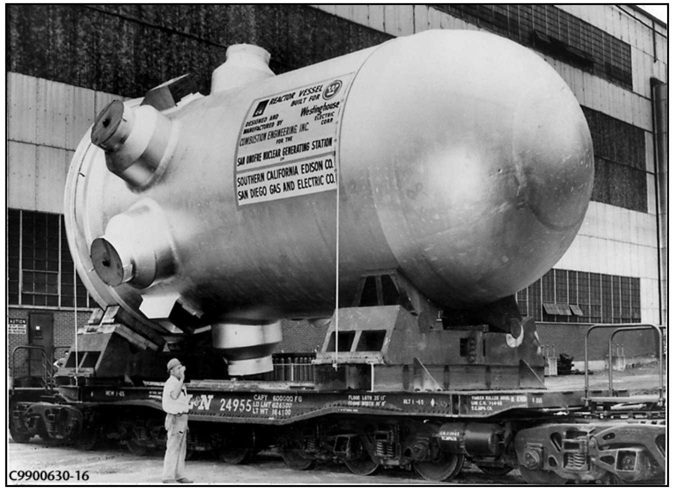
Endorsement of ASME Section III Division 5 Rules for Construction of High Temperature Nuclear Reactors: Update on Industrial Perspective

NRC Public Stakeholders Meeting
December 14, 2017

William Corwin
Office of Advanced Reactor Technologies

ASME Section III Treats Metallic Materials for Low & High Temperatures Differently

- **Allowable stresses for LWR & low-temperature advanced reactor components are not time dependent**
  - **< 700°F (371°C) for ferritic steel and < 800°F (427°C) for austenitic matls**



PWR RPV

Monju SFR IHX

- **At higher temps, materials behave inelastically. Allowable stresses are explicit functions of time & temperature**
  - **Must consider time-dependent phenomena such as creep, creep-fatigue, relaxation, etc.**
  - **ASME Sec III Division 5 provides rules for construction of high temperature reactor components**

U.S. DEPARTMENT OF ENERGY

ASME Sec III Div 5 Contains Construction & Design Rules for High-Temperature Reactors

- Includes gas, liquid metal & molten salt-cooled reactors
- First Issued in Nov 2011, revised in 2013, 2015 & 2017
- Covers high-temperature metallic components explicitly
- Includes rules for graphite & ceramic composites for core supports & internals for first time in any international design code
- Covers low temperature metallic components, largely by reference to other portions of Sec III

Discussions Regarding Endorsement of ASME Section III Division 5 Began in 2015

- **Following multiple DOE-NRC Non-LWR Advanced Reactor Workshops and ASME meetings since 2015, ASME Task Groups have been formed to define potential pathways and schedules for NRC endorsement of Div 5**
  - **Metallic structures & components**
  - **Non-metallic support structures**
- **DOE-NE supports ASME task groups & related technical basis development to reduce technical risk and support private sector deployment of new advanced reactors**
- **NRC/NRO is actively participating in task groups, but has requested industrial input regarding value/prioritization**

DOE Contacted Three Industry Technical Working Groups Regarding Div 5

- TWGs are focused on high temperature gas cooled, fast, and molten salt cooled reactors (HTGRs, FRs & MSRs)
- Input requested on value of endorsement of Div 5 for design of advanced reactors and reduction of their anticipated risk for licensing and deployment
- Support for endorsement of Div 5 has been received from all TWGs
  – Positive verbal feedback provided by Chairs of all TWGs with expectations of supporting letters soon
  – Supporting letter from HTGR TWG sent to Chair of ASME Section III, copies to NRC/NRO, 11/13/17

ASME Will Summarize and Reinforce Industrial Support Requests to NRC/NRO

- When all industry requests have been received by ASME, the Chairs of Section III and the Board of Nuclear Codes and Standards will send joint request to NRC/NRO underscoring the value of and need for endorsement of Division 5
- Existing ASME Task Groups will continue to work towards pathways and schedules for endorsement
- DOE-NE will support R&D activities agreed upon with ASME Committees and BNCS to provide continued technical basis development to optimize the existing Division 5 (2017 edition)

High Priority ASME Code Actions Are Endorsed by BNCS and Supported by DOE R&D Activities

Topics	2019 Edition	Beyond 2019
New simplified analysis methods (EPP) that replace current methods based on linear analysis (and can be used at higher temperatures) for all Class A materials	X	
Adequacy of the definition of S values used for the design of Class B components, which is based on extrapolated properties at 100,000 hours, in light of application to 500,000 hours design	X	
Construction rules for "compact" heat exchanges		X
Incorporation of new materials such as Alloy 617 and Alloy 709 (austenitic stainless)	A617	A709
Pursuit of "all temperature code"		X
Complete the extension of Class A materials for 500,000 hr-design	304H, 316H	Grade 91, 2¼Cr-1Mo, Alloy 800H
Develop design by analysis rules for Class B components (including compact HX)		X
Add non-irradiated and irradiated graphite material properties		X
Develop rules for cladded components for molten salt reactor applications		X

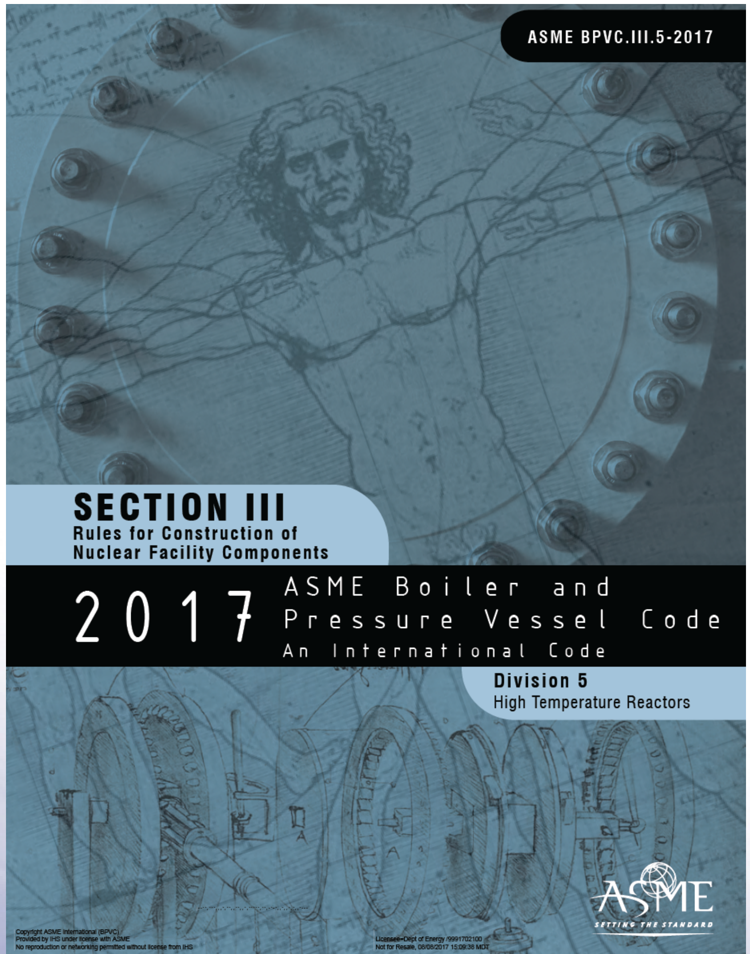
Industry Is Expressing that Endorsement of ASME Sec III Div 5 Is Valuable

- **Requests for input on Div 5 endorsement from broad range of advanced reactor vendors and suppliers is evoking positive responses**
- **ASME Section III and BNCS are also very supportive of Div 5 endorsement**
- **DOE-NE is supporting and coordinating industry and ASME support for endorsement with active R&D activities to optimize the Division 5**
- **Endorsement of Div 5 is anticipated to reduce technical risk and support private sector deployment of new advanced reactors**

Task Groups on ASME/NRC Liaison for Division 5

Sam Sham
Chair, Subgroup on Elevated Temperature Design (BPV III)
Technical Manager, Advanced Reactor Materials, Nuclear Engineering Division, Argonne National Laboratory

ASME Boiler and Pressure Vessel Code



Section III
Rules for Construction of Nuclear Facility Components

Division 5 (2017 Edition)
High Temperature Reactors

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Section III Division 5 (2017 Edition) Scope

- Division 5 rules govern the construction of vessels, storage tanks, piping, pumps, valves, supports, core support structures and nonmetallic core components for use in high temperature reactor systems and their supporting systems
  - Construction, as used here, is an all-inclusive term that includes material, design, fabrication, installation, examination, testing, overpressure protection, inspection, stamping, and certification
  - High temperature reactors include gas-cooled reactors, liquid metal reactors and molten salt reactors (liquid or solid fuel)

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SETTING THE STANDARD

Section III Division 5 (2017 Edition) Organization

Class	Subsection	Subpart	Subsection ID	Title	Scope
General Requirements *					
Class A, B, & SM	HA	A	HAA	Metallic Materials	Metallic
Class SN		B	HAB	Graphite and Composite Materials	Nonmetallic
Class A Metallic Pressure Boundary Components					
Class A	HB	A	HBA	Low Temperature Service	Metallic
Class A		B	HBB	Elevated Temperature Service	Metallic
Class B Metallic Pressure Boundary Components					
Class B	HC	A	HCA	Low Temperature Service	Metallic
Class B		B	HCB	Elevated Temperature Service	Metallic
Class A and Class B Metallic Supports					
Class A & B	HF	A	HFA	Low Temperature Service	Metallic
Class SM Metallic Core Support Structures *					
Class SM	HG	A	HGA	Low Temperature Service	Metallic
Class SM		B	HGB	Elevated Temperature Service	Metallic
Class SN Nonmetallic Core Components *					
Class SN	HH	A	HHA	Graphite Materials	Graphite
Class SN		B	HHB	Composite Materials	Composite

* Class designation being balloted

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Division 5 Rules for Metallic Components (1/2)

- The 2017 edition of Division 5 construction rules for metallic components were developed to guard against time independent and time dependent structural failure modes
- The construction rules for Division 5 Class A components are common to all qualified Class A materials
  - If additional applicable data for a specific qualified material in the 2017 edition of Division 5 are available that would permit extension to longer design lifetimes, or if a new qualified material is added, the construction rules of the 2017 edition of Division 5 would remain the same
  - Extension of design lifetimes for qualified materials or incorporation of new qualified materials is considered as “optimization” rather than affecting the “adequacy” of the 2017 edition of the Division 5 construction rules
  - Guidelines for design data needs for new materials are provided in Appendix HBB-Y of the 2017 edition of Division 5

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Division 5 Rules for Metallic Components (2/2)

- Design procedures and materials data not contained in the 2017 edition of Division 5 may be required to ensure the integrity or the continued functioning of the structural part during the specified service life
  - E.g., rules do not provide methods to evaluate deterioration that may occur in service as a result of corrosion, mass transfer phenomena, radiation effects, or other material instabilities
  - Owner/operator has the responsibility to demonstrate to the regulator that these effects are accounted for in the design of the components

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ASME Actions to Optimize 2017 Edition of Division 5 Rules

- Various actions are being taken to extend qualified lifetimes of Class A materials to support 60-year design life
- New Class A material, Alloy 617, is being incorporated into Division 5 to expand design envelope
- Elastic, perfectly plastic methods are being developed to modernize and simplify Division 5 design analyses
- Inelastic analysis methods are being developed for incorporation into Division 5 Appendix HBB-Z
- Design rules for integrally cladded components with weld overlay on Class A materials are being developed to support molten salt reactor applications
- Graphite irradiation data are being incorporated into Division 5 to support use of graphite design rules
- Ceramic composite design rules are being incorporated into Division 5

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ASME/NRC Liaison Task Groups for Division 5

- Two ASME/NRC liaison task groups for Division 5, one on metallic and the second on nonmetallic, were formed to develop roadmaps to assist NRC's internal assessment of endorsing the 2017 edition of Division 5

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Task Group Activities - Metallic

- Phase I (by August 2018)
  - White paper on the technical bases of the current rules for metallic components
    - High level exposition through reference to relevant references
  - White paper to assess the issue lists identified previously by NRC and ACRS at various times *
    - Will separate issues into two categories, one that is within the 2017 edition of the ASME Division 5 scope and the other outside the ASME space (e.g., irradiation effects)
    - Will categorize the ASME code space items into basic rules issues and issues relating to the optimization of the 2017 edition of the Division 5 rules
    - Will assess how the issues are addressed by the 2017 edition of the Division 5 rules and identify gaps, if any
  - These white papers would provide an important input for NRC's internal assessment of endorsing the 2017 edition of Division 5
- Phase II (by February 2019)
  - Roadmap on ASME actions and schedule to dress the identified gaps for metallic components

* O'Donnell, Hull and Malik, "Structural Integrity Code and Regulatory Issues in the Design of High Temperature Reactors," Proceedings of the 4th International Topical Meeting on High Temperature Reactor Technology, HTR-2008, Paper HTR2008-58061, American Society of Mechanical Engineers, New York, NY (2008)

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Task Group Activities - Nonmetallic

- NGNP high temperature materials white paper was issued in 2009
  - There were subsequent interactions between NGNP and NRC to address NRC's RAIs
  - Revision 1 including the NRC comments and DOE/vendor responses was issued August 2012*
- The information is being used to develop a roadmap on ASME actions and schedule to address any identified gaps for nonmetallic core components (February 2019)
- This would provide an important input for NRC's internal assessment of endorsing the 2017 edition of Division 5

* NGNP High Temperature Materials White Paper, INL/EXT-09-17187, Revision 1, August 2012

ASME
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U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

SITING CONSIDERATIONS RELATED TO POPULATION FOR SMALL MODULAR AND NON-LIGHT WATER REACTORS

December 14, 2017

PURPOSE

- Discuss siting requirements and guidance.
- Describe siting considerations for Small Modular Reactors (SMRs) and non-Light Water Reactors (non-LWRs).
- Obtain stakeholder feedback to support the staff’s assessment of whether a different approach, criteria, or guidance is needed.
- Discuss potential next steps.

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BACKGROUND

- SMRs and non-LWRs are anticipated to have smaller cores with simplified, inherent, passive safety features, which may result in smaller postulated accident releases.

- The staff discussed in SECY-16-0012 that use of mechanistic source term analysis methods could "…allow future COL applicants to consider reduced distances to EABs and LPZs, and potentially increased [SMR] proximity to population centers."

- Commission policy is that reactor sites should be located away from densely populated centers must also be taken into consideration.

- The draft White Paper (ML17333B158) is limited in scope to SMRs and non-LWR and focuses on the siting considerations related to population
  - Other siting considerations (safety, environmental, economic, or other factors) are not addressed

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Protecting People and the Environment

SITING REQUIREMENTS AND ASSOCIATED GUIDANCE

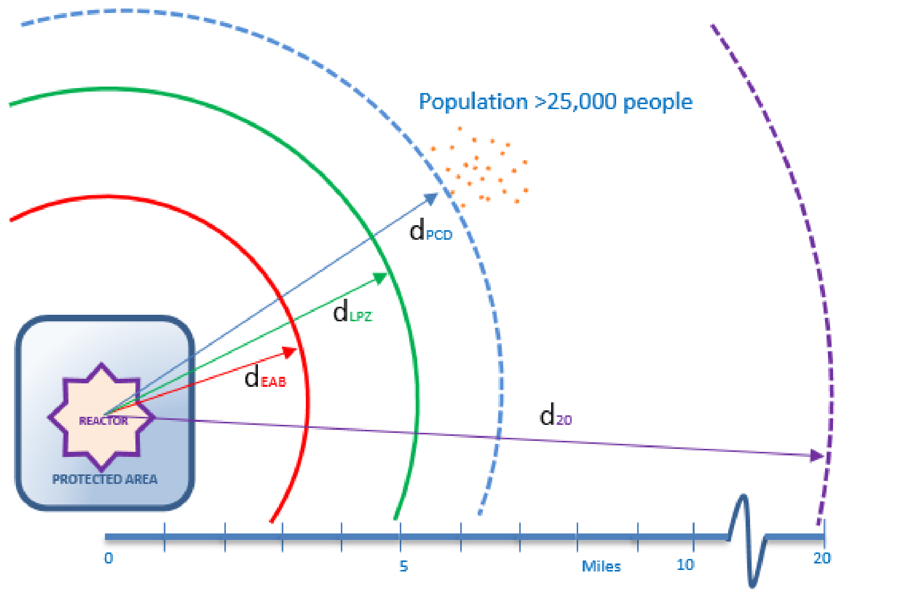
Minimum Distance to a Densely Populated Center

- **Addresses Individual Radiological Dose**

10 CFR 100.21(b)

- An individual located on the outer boundary of the LPZ would not receive a radiation dose in excess of 25 rem TEDE.

- Nearest boundary of a densely populated center (> 25,000 residents) be at least one and one-third times the distance from the reactor to the outer boundary of the LPZ.



d_{EAB} – exclusion area boundary (EAB) radial distance

d_{LPZ} – low population zone (LPZ) radial distance

d_{PCD} – population center distance (PCD) to the nearest boundary of a densely populated center

d_{20} – 20 mile outward radial distance (population density of <500 persons per square mile - RG-4.7)

SITING REQUIREMENTS AND ASSOCIATED GUIDANCE (Continued)

Population Density

- **Addresses Societal Consequences (per SOC for Part 100 rule)**

10 CFR 100.21(h)

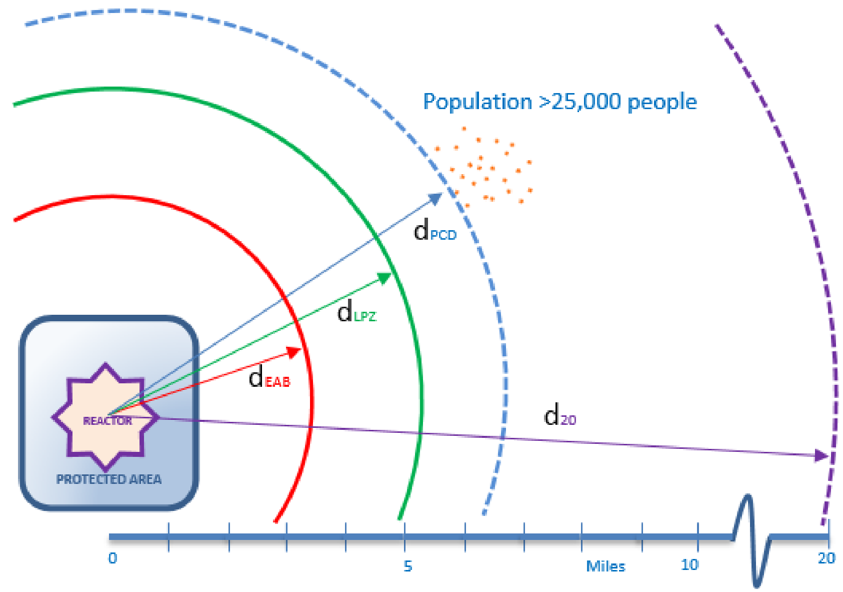
- Reactor sites should be located away from very densely populated centers and areas of low population are generally preferred.

NRC Policy (61 FR 65157)

- “a long standing policy of siting reactors away from densely populated centers”

Regulatory Guide 4.7

- 20-mile Radial Distance
- 500 persons per mile^2



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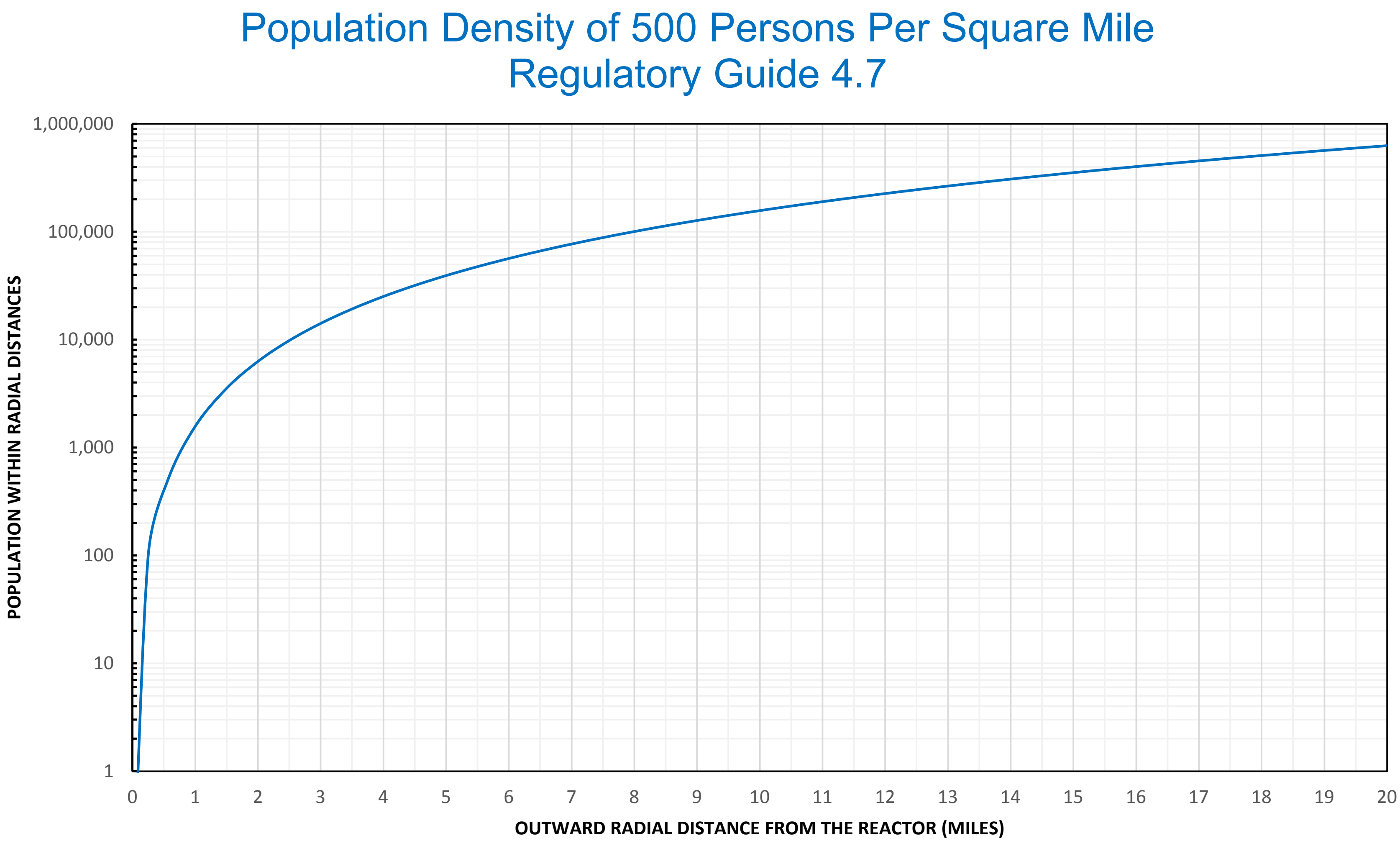
REGULATORY GUIDE 4.7



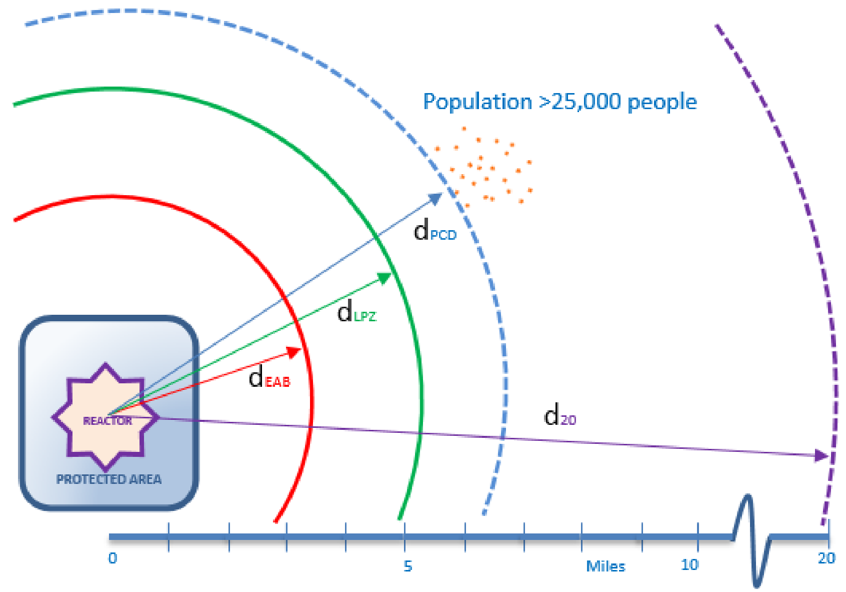
SITING CONSIDERATIONS FOR SMRS AND NON-LWRS

- Based solely on dose-based siting requirements, an applicant could potentially propose a site location for an SMR or non-LWR closer to a population center than large LWRs. Perhaps as close as 400 meters to one mile.
- However, applying the current guidance in RG 4.7 would generally preclude siting SMRs and non-LWRs in close proximity to densely populated (urban) areas.
- The staff estimates that a densely populated center would be need to be at least 4 miles away to meet the guidance.
- ORNL study - 8% of the land in the U.S. would be screened out based on population density alone. Many desirable sites (coal replacement) could be impacted.

DRAFT WHITE PAPER QUESTIONS FOR DISCUSSIONS

1.

- Should the NRC consider a different approach be used to determine the minimum distance to the population center for SMRs and non-LWRs?
- If so, how should this minimum distance be established?



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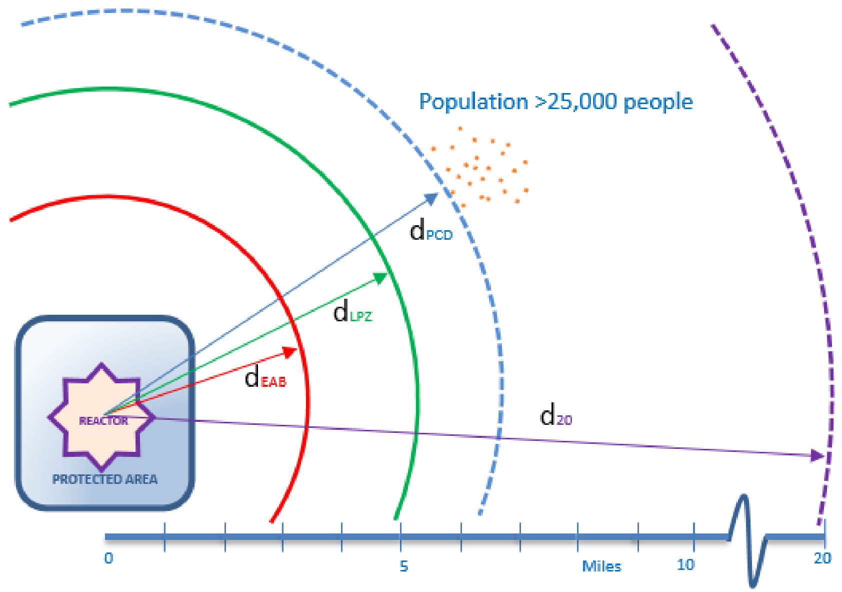
DRAFT WHITE PAPER QUESTIONS FOR DISCUSSIONS (Cont'd)

2.

- Should the NRC consider different criteria or guidance be developed for SMRs and non-LWRs to make the determination that the site is "away from densely populated centers"?
- If so, how should defense-in-depth be considered in developing the criteria?
- For example, should more complete, more realistic, and site-specific radiological release, transport, and dispersion models be used to evaluate societal risk to inform siting decisions?



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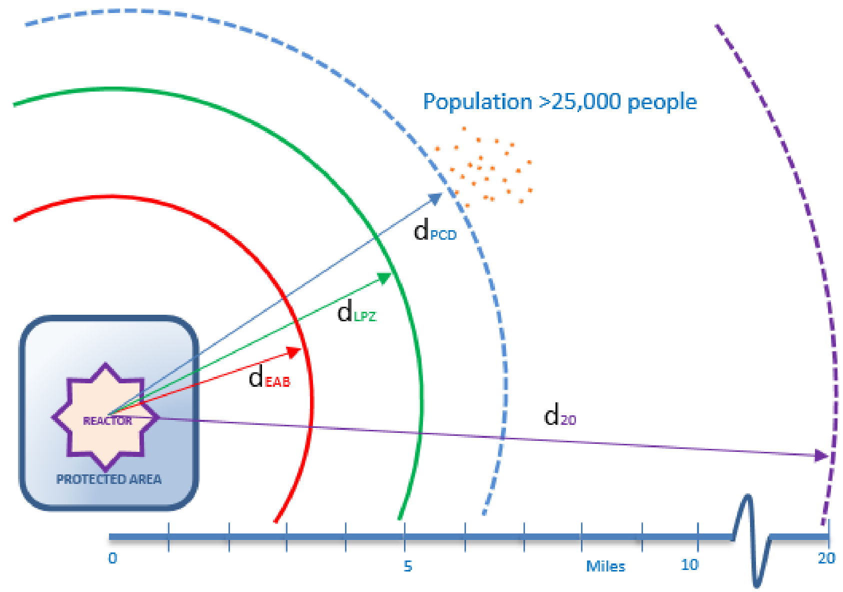
DRAFT WHITE PAPER QUESTIONS FOR DISCUSSIONS (Cont'd)

3. Additionally, the staff is seeking feedback from SMR and non-LWR designers and potential applicants regarding their plans for proposing to site reactors closer to densely populated centers, or in areas with a population density in excess of 500 persons per square mile out to 20 miles.



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NEXT STEPS

- Consider insights obtained from stakeholders and determine whether clarifications to RG 4.7 or other actions would be beneficial to address siting criteria for SMRs and non-LWRs.

- Proposed actions as described in SECY-16-0012

BACK-UP SLIDES

BACKGROUND

- 10 CFR 100, Reactor Siting Criteria (1962)
- NUREG-0478, Metropolitan Siting – A Historic Perspective (1978)
- NUREG-0625, Report of the Siting Policy Task Force (1979)
- 61 FR 65157, Statements of Consideration (1996)
  - Describes augmentations to 10 CFR 100, Reactor Site Criteria - current final rule
  - Describes the Commission Policy on siting reactors away from densely populated center
  - Clarifies guidance in Reg. Guide 4.7 to site reactors away from densely populated center
- 10 CFR 100, Reactor Siting Criteria (1996)
  - Current regulations

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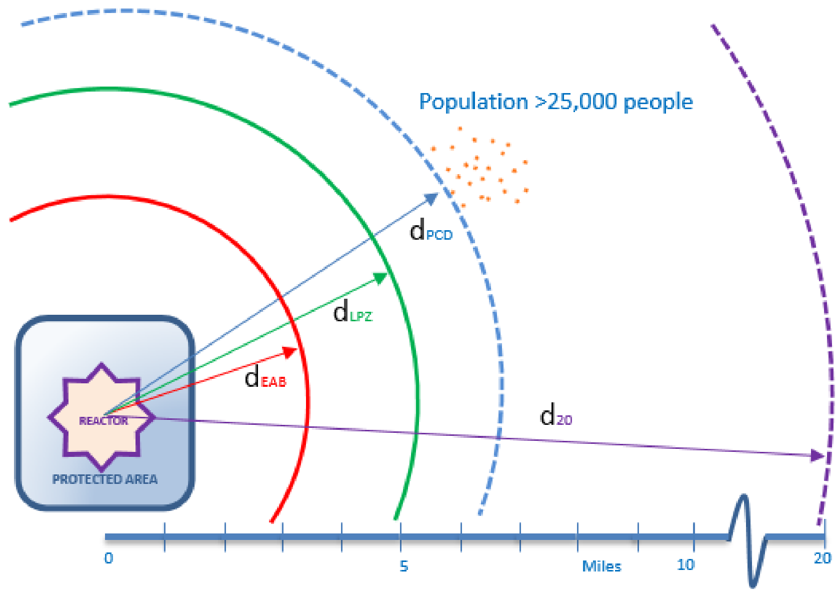
SITING DISTANCES

10 CFR 100.3 Definitions

- Exclusion Area Boundary
- Low Population Zone
- Population Center Distance
- Nearest boundary of less than 25,000 residents



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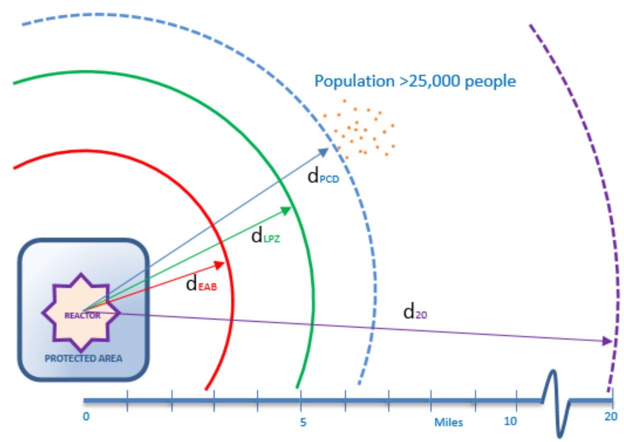
SITING DISTANCES (Cont'd)

§ 50.34(a)(1)(D)(1)(2) Radiological dose consequences

- (1) An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE)
- (2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE)



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Siting Options and Challenges for SMR Deployment in the US – Focus on Population Density

Presentation for:

Possible Regulatory Process Improvements for Advanced Reactor Designs

December 14, 2017

maysgt@ornl.gov

bellesrj@ornl.gov

ORNL Collaborators:
Randy Belles
Gary Mays
T. Jay Harrison
Femi Omitaomu
Mike Poore

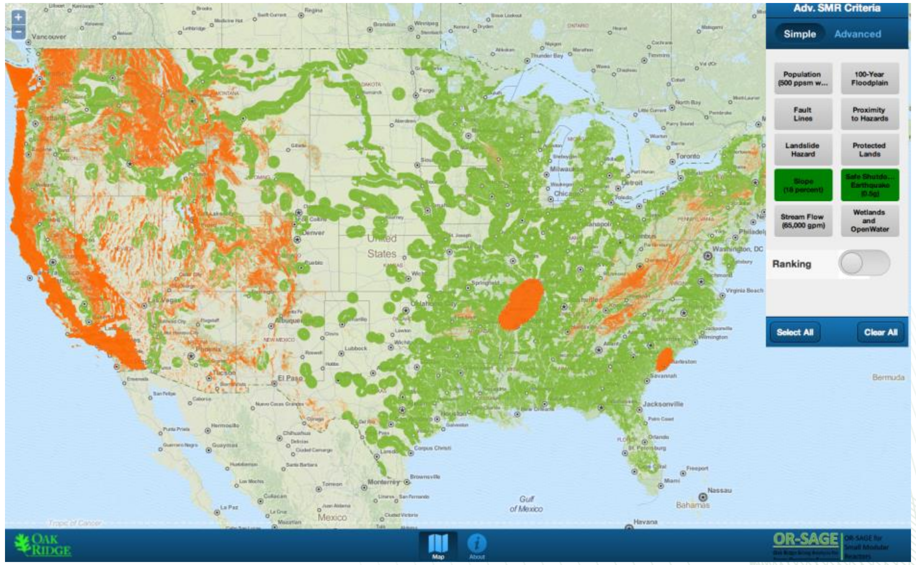
ORNL is managed by UT-Battelle
for the US Department of Energy

ORNL Siting Tool Provides Insights on Challenges and Benefits of Deploying SMRs

- Using geographical information systems (GIS) and spatial modeling techniques to gain insights into options, challenges, and benefits of SMRs
  - How much additional area is available to site SMRs vs large LWRs?
  - How do reduced cooling water requirements differ? What about dry cooling?
  - How close are potential areas for siting SMRs to electrical transmission lines? railways?
  - Are older coal plants, DoD, and DOE sites suitable for siting SMRs?
  - What about siting options for federal agencies to meet clean energy goals?

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

OR-SAGE Results Presented in ORNL Visualization Lab

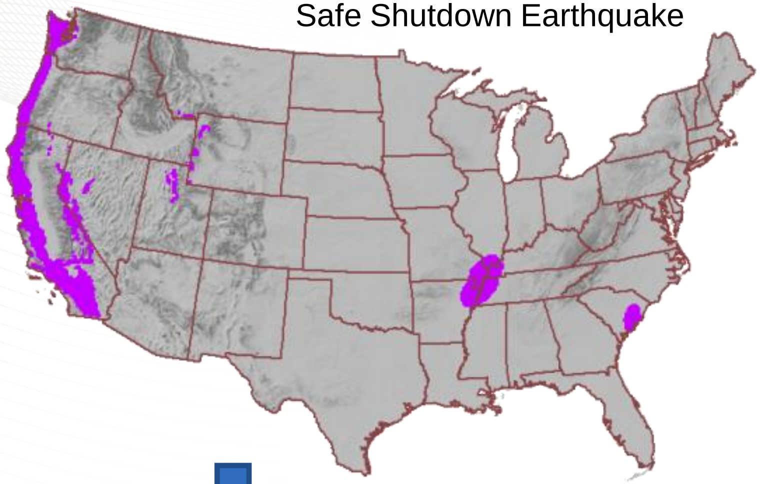


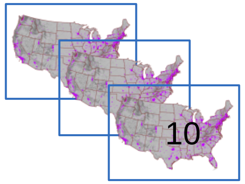
OR-SAGE GUI for Evaluating SMR Siting Options

Oak Ridge National Laboratory

Baseline Query* of Database Indicates 33% of Land Potentially Available for SMR Siting

Safe Shutdown Earthquake

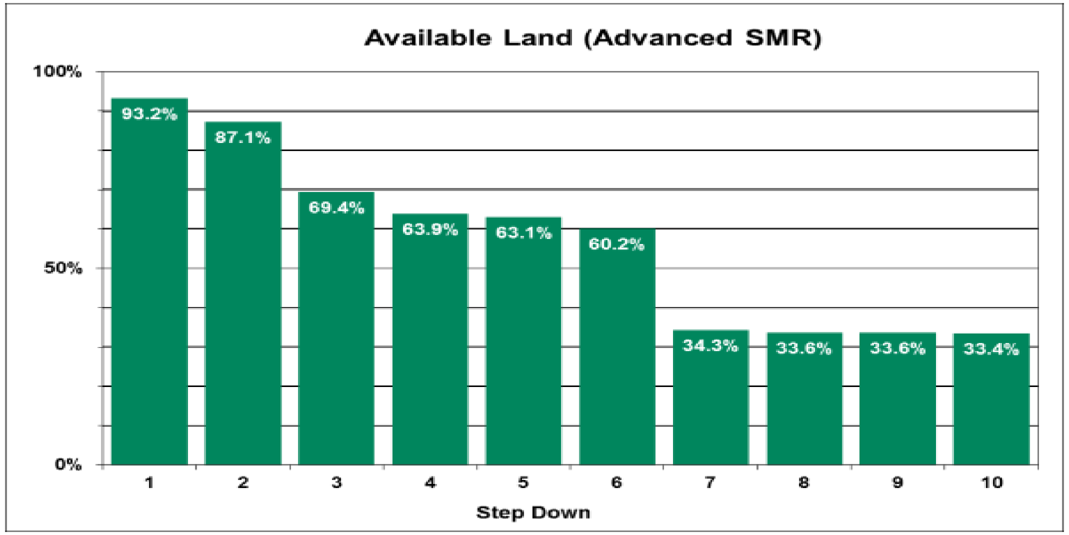




* 540 MWe @ 65,000 gpm stream flow on 50-acre site

	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

Apply Screening Factors



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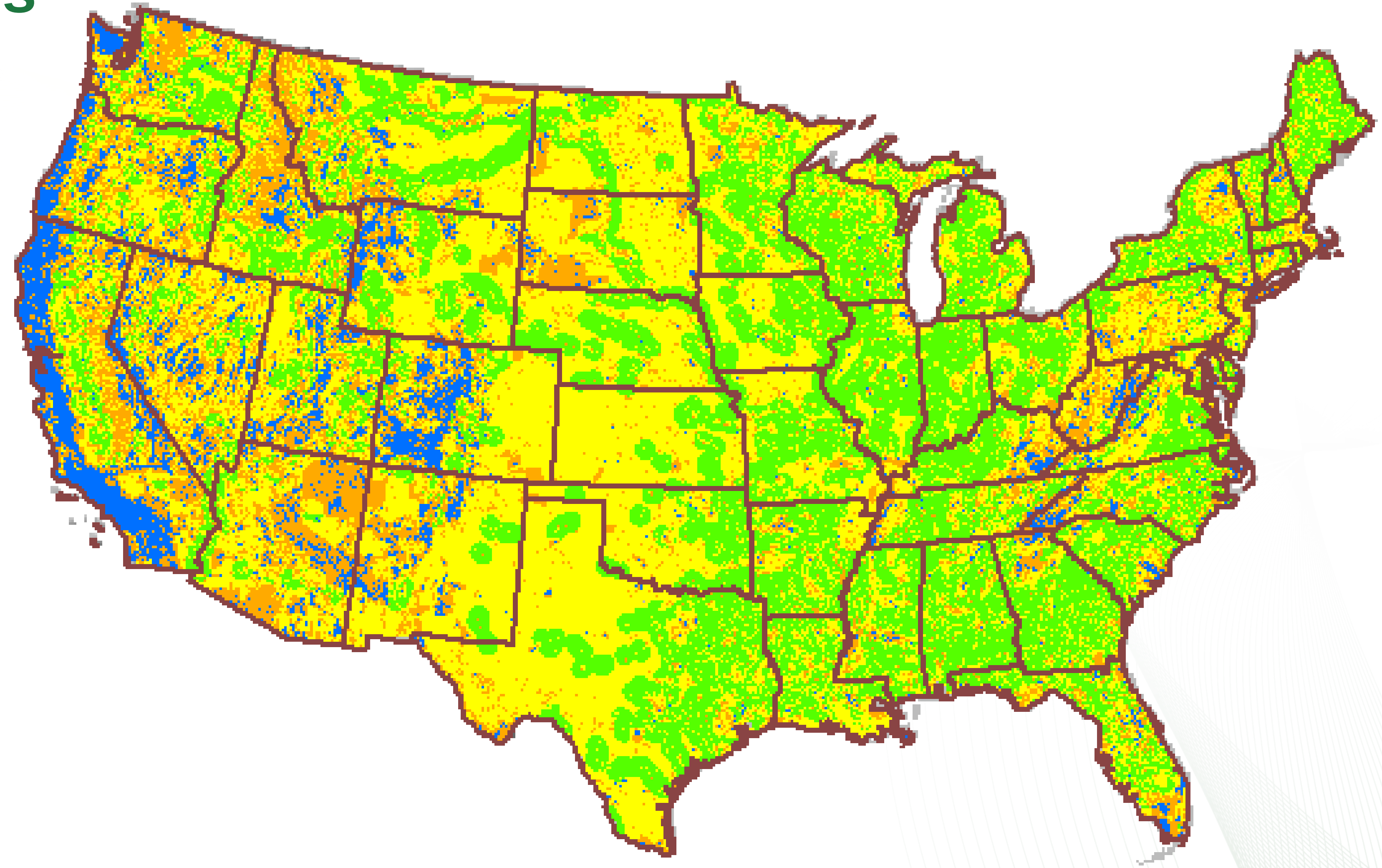
Composite screening results provides additional insight into siting possibilities



Composite Map Result

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

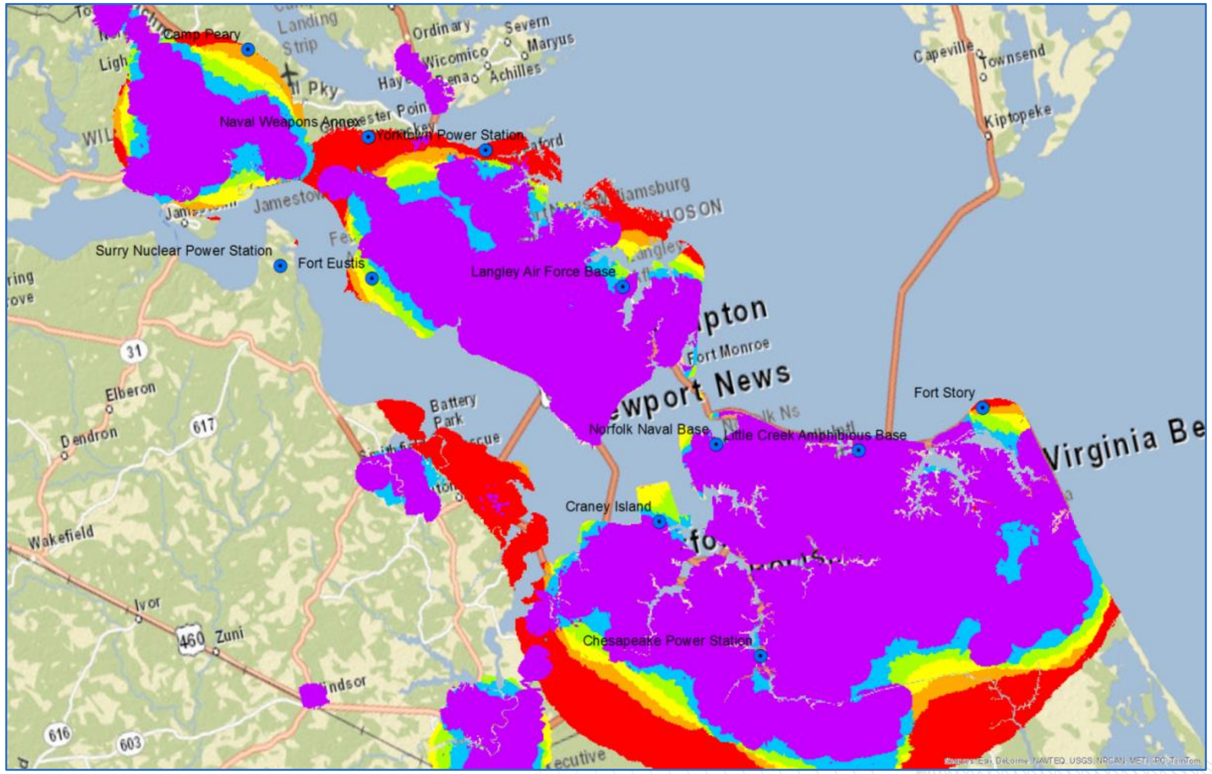
Yellow + Green = 74.7% potentially meets criteria

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

A previous project evaluated 11 Sites in Hampton Roads

Sites with excellent potential to host an SMR (including low population within 10 miles):

- Surry NPP
- Camp Peary
- Yorktown Weapons Station



Other possible sites:

Proposed site	Distance to current population density > 500 ppsm[a] Possible EPZ??
Chesapeake Energy Center	1 mile
Craney Island	1 mile
Fort Story	2-5 miles
Langley Air Force Base	2 miles
Norfolk Naval Base	3 miles
Yorktown Power Station	5 miles

[a]Assumes optimal siting at facility.

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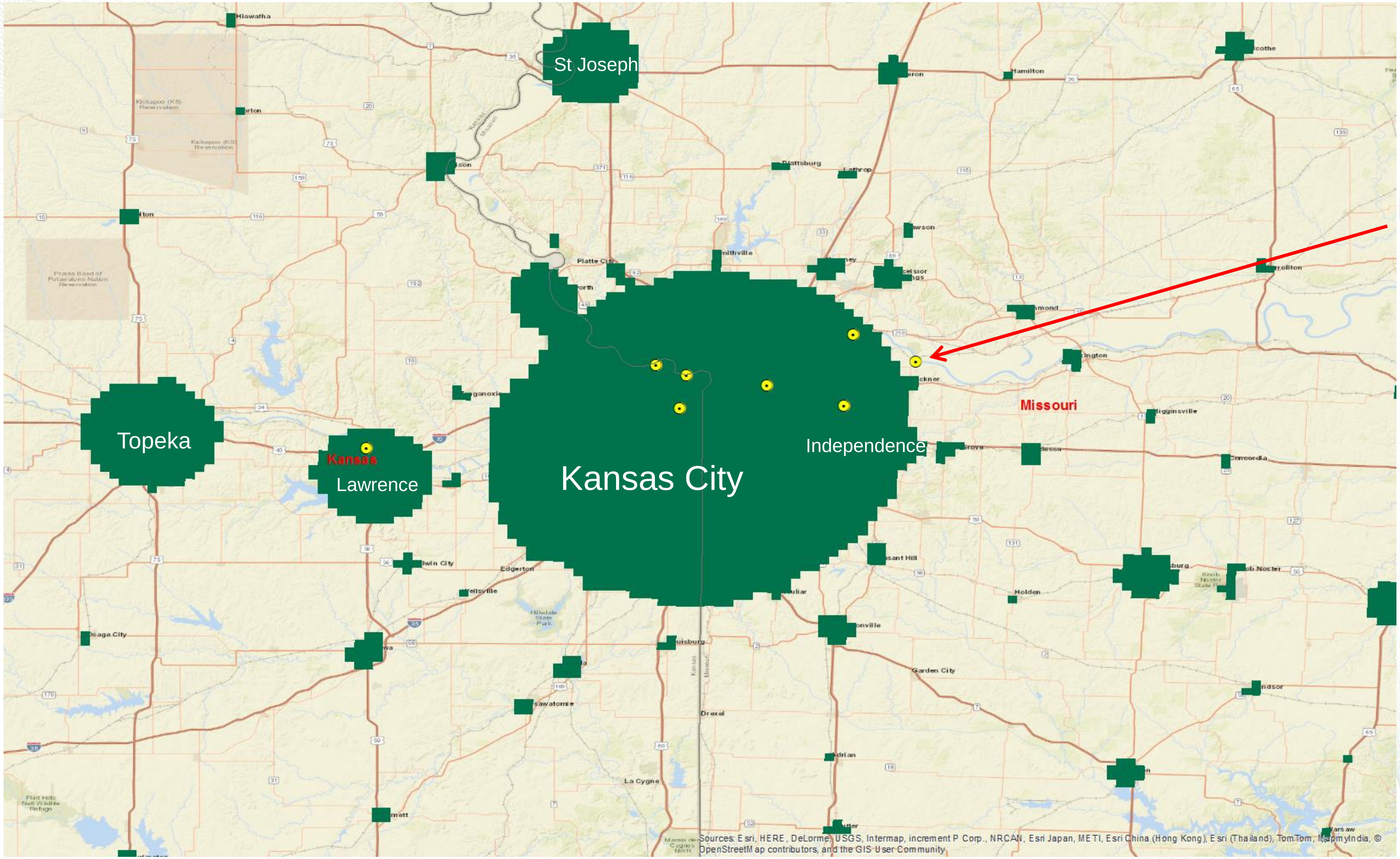
Current Population Density (500 ppsm @ 20 miles)



Population not an issue at 20 miles

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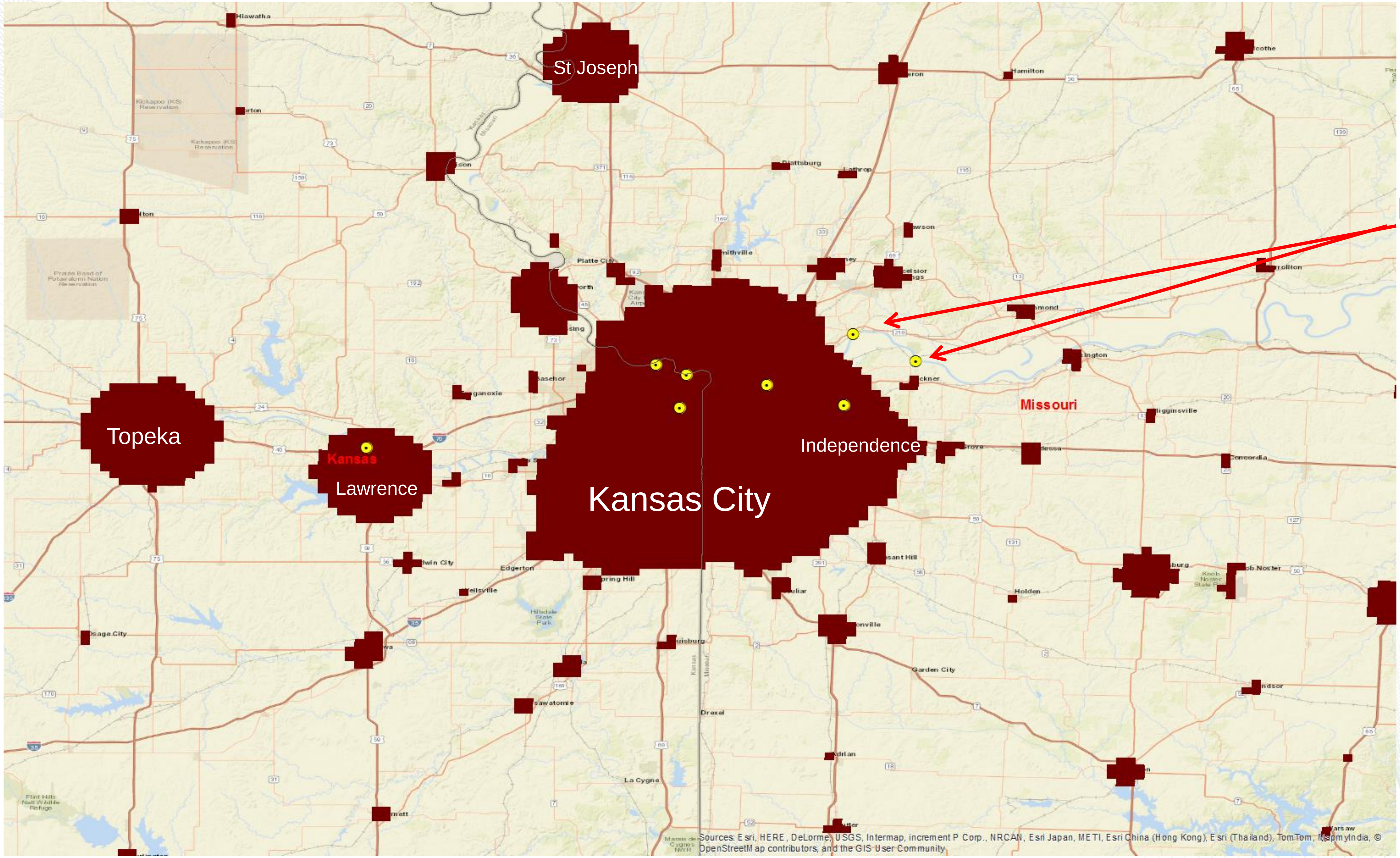
Current Population Density (500 ppsm @ 10 miles)



Population not an issue at 10 miles

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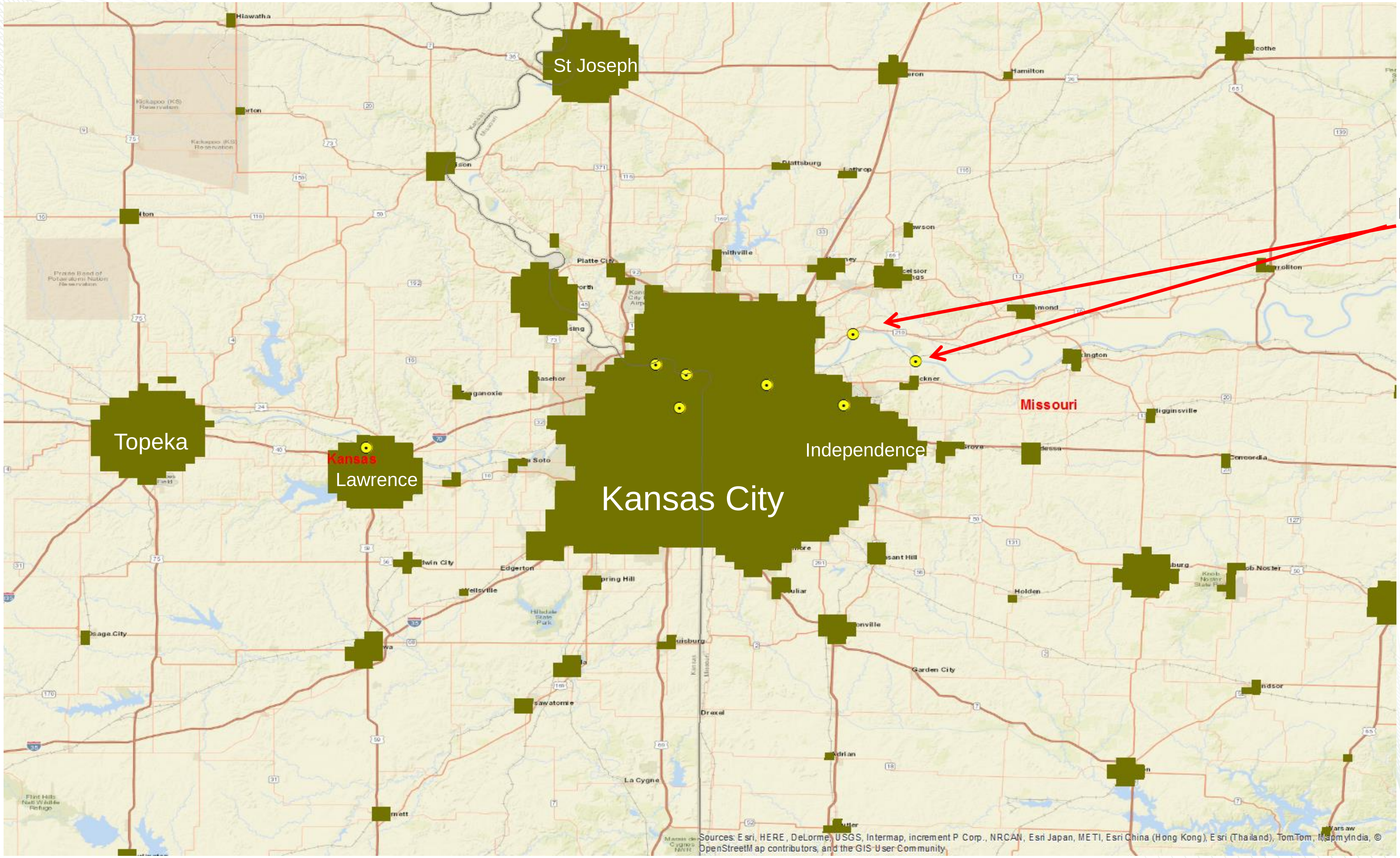
Current Population Density (500 ppsm @ 5 miles)



Population not an issue at 5 miles

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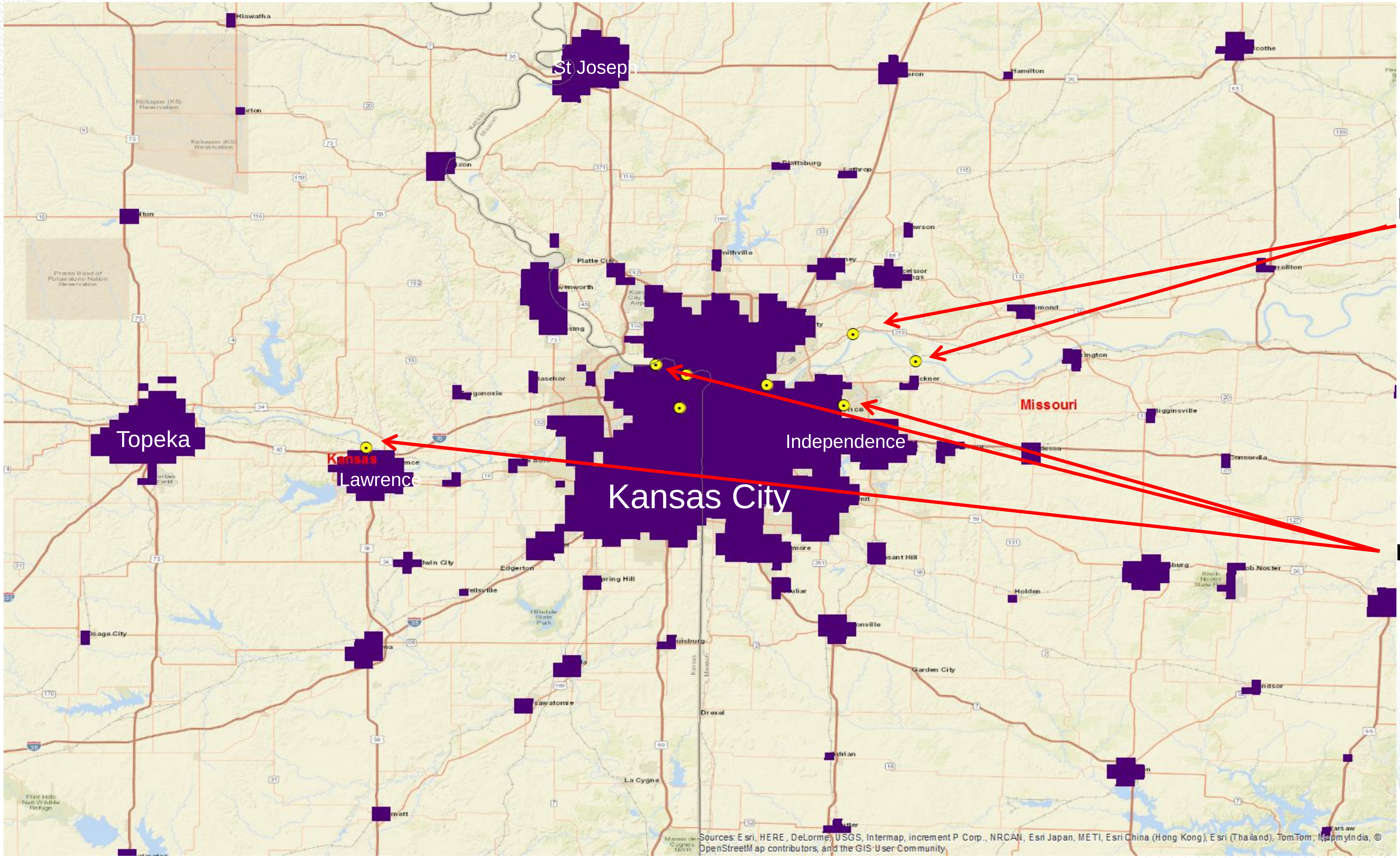
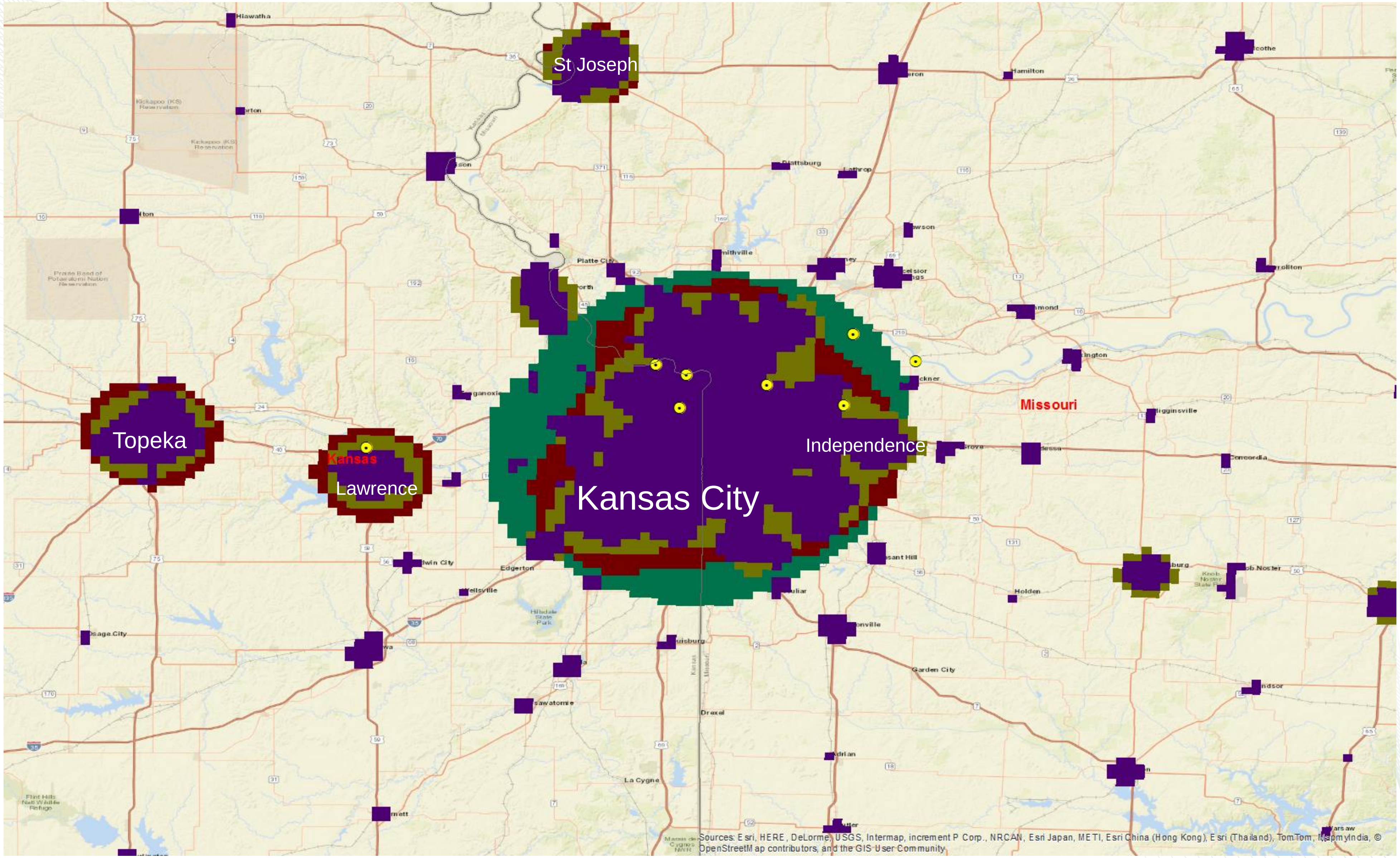
Current Population Density (500 ppsm @ 2 miles)



Population not an issue at 2 miles

Population may not be an issue at 2 miles

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



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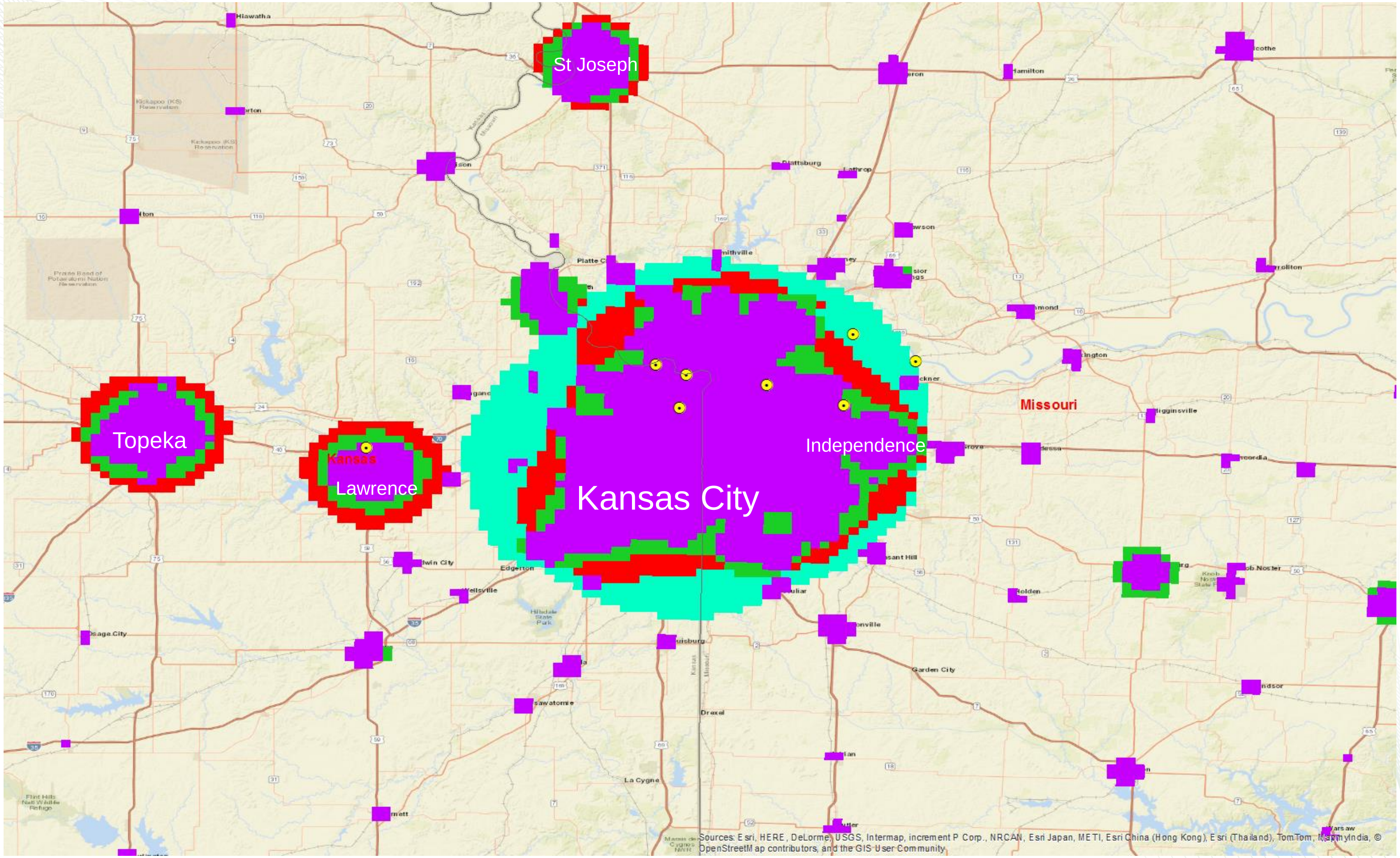
2030 Density Comparison (500 ppsm @ 2, 5, 10, & 20 mi)



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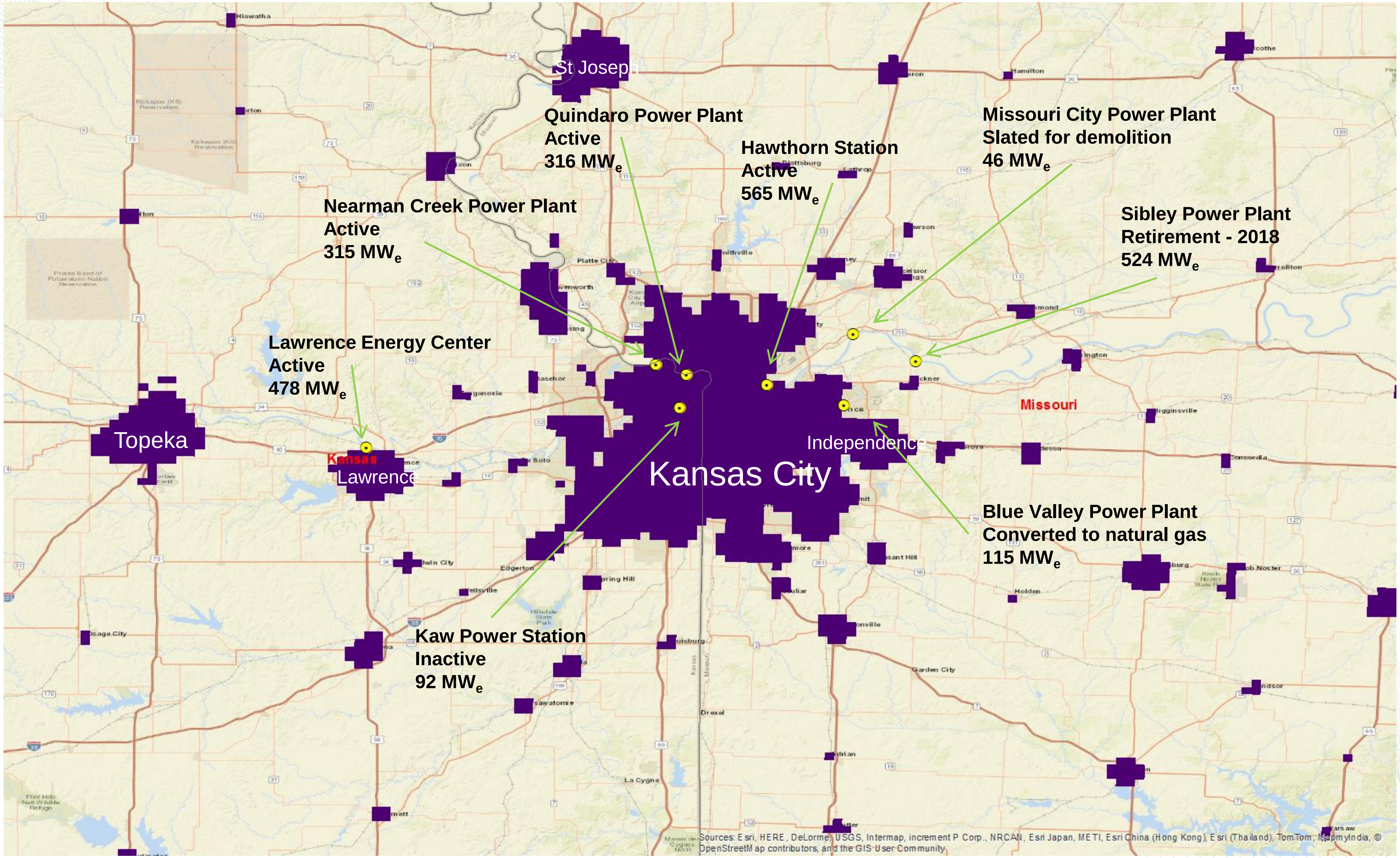
Identification of Selected Fossil Plants



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Summary

- OR-SAGE results are based on RG 4.7 means of locating reactors away from population centers by observing ambient population density average over any radial distance out to 20 miles.
  - Discussed in the NRC white paper.
- Reducing the radial distance calculation to values less than 20 miles is intended to be a visual surrogate for calculations that may indicate reduced source term and subsequent lower EAB/LPZ dose calculations for advanced reactor designs.

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