June 29, 2023

TP-LIC-LET-0085
Project Number 99902100

U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
ATTN: Document Control Desk

Subject: Specified Acceptable System Radionuclide Release Design Limit, Functional Containment, and Major Accident Presentation Materials

This letter provides the TerraPower, LLC presentation material for the upcoming "Specified Acceptable System Radionuclide Release Design Limit (SARRDL), Functional Containment, and Major Accident" pre-application engagement meeting (Enclosures 2, 3, and 4). The presentation material contains proprietary information and as such, it is requested that Enclosure 4 be withheld from public disclosure in accordance with 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." An affidavit certifying the basis for the request to withhold Enclosure 4 from public disclosure is included as Enclosure 1. Enclosure 4 also contains export controlled information (ECI), which can be disclosed to Foreign Nationals only in accordance with the requirements of 15 CFR 730 and 10 CFR 810, as applicable. Proprietary information and ECI has been redacted from the presentation material provided in Enclosure 3; redacted information is identified using [[ ]]$^{(a)(4)}$, [[ ]]^{ECI}, or [[ ]]$^{(a)(4), ECI}$.

This letter and enclosures make no new or revised regulatory commitments.

If you have any questions regarding this submittal, please contact Ryan Sprengel at rsprengel@terrapower.com or (425) 324-2888.

Sincerely,

Ryan Sprengel
Director of Licensing, Natrium
TerraPower, LLC

Enclosures:

1. TerraPower, LLC Affidavit and Request for Withholding from Public Disclosure (10 CFR 2.390(a)(4))
2. "Specified Acceptable System Radionuclide Release Design Limit, Functional Containment, and Major Accident" Presentation Material – Open Meeting – Non-Proprietary (Public)
3. "Specified Acceptable System Radionuclide Release Design Limit, Functional Containment, and Major Accident" Presentation Material – Closed Meeting – Non-Proprietary (Public)
4. "Specified Acceptable System Radionuclide Release Design Limit, Functional Containment, and Major Accident" Presentation Material – Closed Meeting – Proprietary (Non-Public)

cc: Mallecia Sutton, NRC
William Jessup, NRC
Andrew Proffitt, NRC
Nathan Howard, DOE
Jeff Ciocco, DOE

ENCLOSURE 1

TerraPower, LLC Affidavit and Request for Withholding from Public Disclosure (10 CFR 2.390(a)(4))

Enclosure 1
TerraPower, LLC Affidavit and Request for Withholding from Public Disclosure
(10 CFR 2.390(a)(4))

I, George Wilson, hereby state:

1. I am the Vice President, Regulatory Affairs and I have been authorized by TerraPower, LLC (TerraPower) to review information sought to be withheld from public disclosure in connection with the development, testing, licensing, and deployment of the Natrium™ reactor and its associated fuel, structures, systems, and components, and to apply for its withholding from public disclosure on behalf of TerraPower.

2. The information sought to be withheld, in its entirety, is contained in Enclosure 4, which accompanies this Affidavit.

3. I am making this request for withholding, and executing this Affidavit as required by 10 CFR 2.390(b)(1).

4. I have personal knowledge of the criteria and procedures utilized by TerraPower in designating information as a trade secret, privileged, or as confidential commercial or financial information that would be protected from public disclosure under 10 CFR 2.390(a)(4).

5. The information contained in Enclosure 4 accompanying this Affidavit contains non-public details of the TerraPower regulatory and developmental strategies intended to support NRC staff review.

6. Pursuant to 10 CFR 2.390(b)(4), the following is furnished for consideration by the Commission in determining whether the information in Enclosure 4 should be withheld:

   a. The information has been held in confidence by TerraPower.

   b. The information is of a type customarily held in confidence by TerraPower and not customarily disclosed to the public. TerraPower has a rational basis for determining the types of information that it customarily holds in confidence and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application and substance of that system constitute TerraPower policy and provide the rational basis required.

   c. The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR 2.390, it is received in confidence by the Commission.

   d. This information is not available in public sources.

   e. TerraPower asserts that public disclosure of this non-public information is likely to cause substantial harm to the competitive position of TerraPower, because it would enhance the ability of competitors to provide similar products and services by reducing their expenditure of resources using similar project methods, equipment, testing approach, contractors, or licensing approaches.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on: June 28, 2023

George Wilson

George Wilson
Vice President, Regulatory Affairs
TerraPower, LLC

ENCLOSURE 2

“Specified Acceptable System Radionuclide Release Design Limit, Functional Containment, and Major Accident Presentation Materials”
Presentation Material – Open Meeting

Non-Proprietary (Public)

SARRDL, Functional Containment, and Major Accident Update

a TerraPower & GE-Hitachi technology

TP-LIC-PRSNT-0006

SUBJECT TO DOE COOPERATIVE AGREEMENT NO. DE-NE0009054

Table of Contents

- Natrium™ reactor overview
- Defining SARRDLs
- Functional Containment Strategy and Definition Update
  – Relevant PSAR sections
- Establishing Functional Containment Performance Criteria
- Major Accident Approach

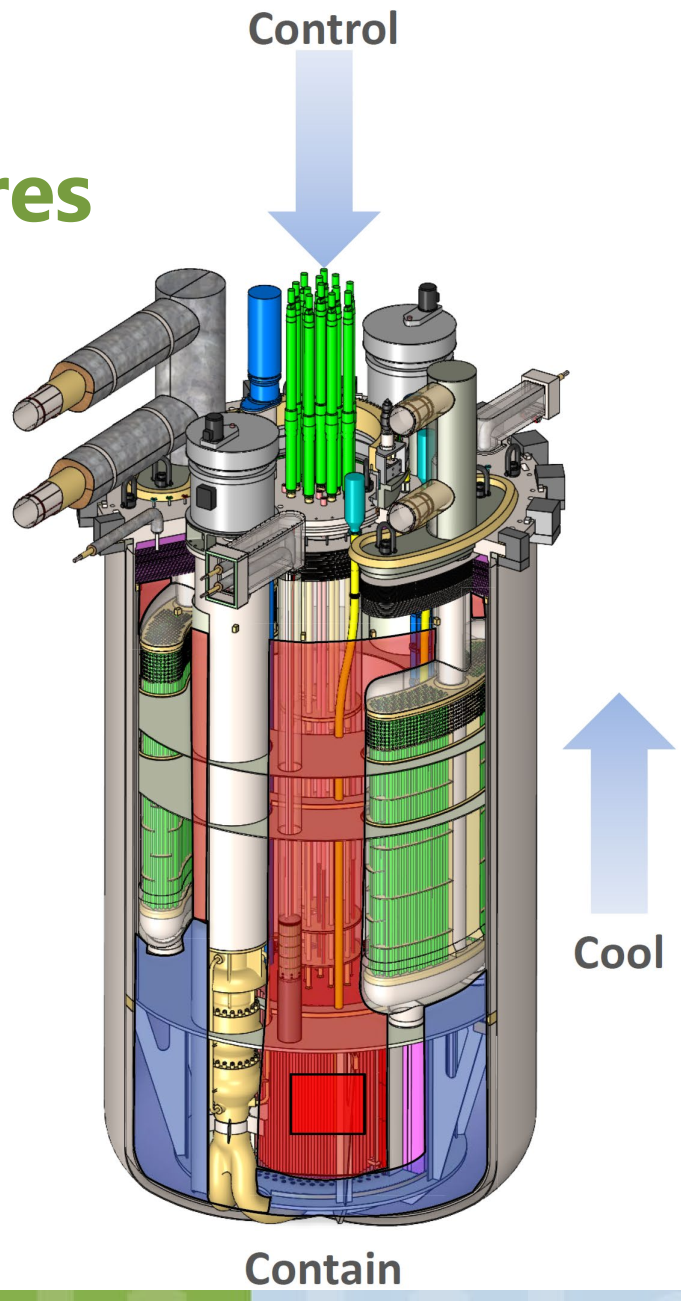
Natrium Reactor Overview

- The Natrium project is demonstrating the ability to design, license, construct, startup and operate a Natrium reactor.

- Pre-application interactions are intended to reduce regulatory uncertainty and facilitate the NRC's understanding of the Natrium design and its safety case.

Natrium Safety Features

- Pool-type Metal Fuel SFR with Molten Salt Energy Island
  - Metallic fuel and sodium have high compatibility
  - No sodium-water reaction in steam generator
  - Large thermal inertia enables simplified response to abnormal events
- Simplified Response to Abnormal Events
  - Reliable reactor shutdown
  - Transition to coolant natural circulation
  - Indefinite passive emergency decay heat removal
  - Low pressure functional containment
  - No reliance on Energy Island for safety functions
- No Safety-Related Operator Actions or AC power
- Technology Based on U.S. SFR Experience
  - EBR-I, EBR-II, FFTF, TREAT
  - SFR inherent safety characteristics demonstrated through testing in EBR-II and FFTF



Control

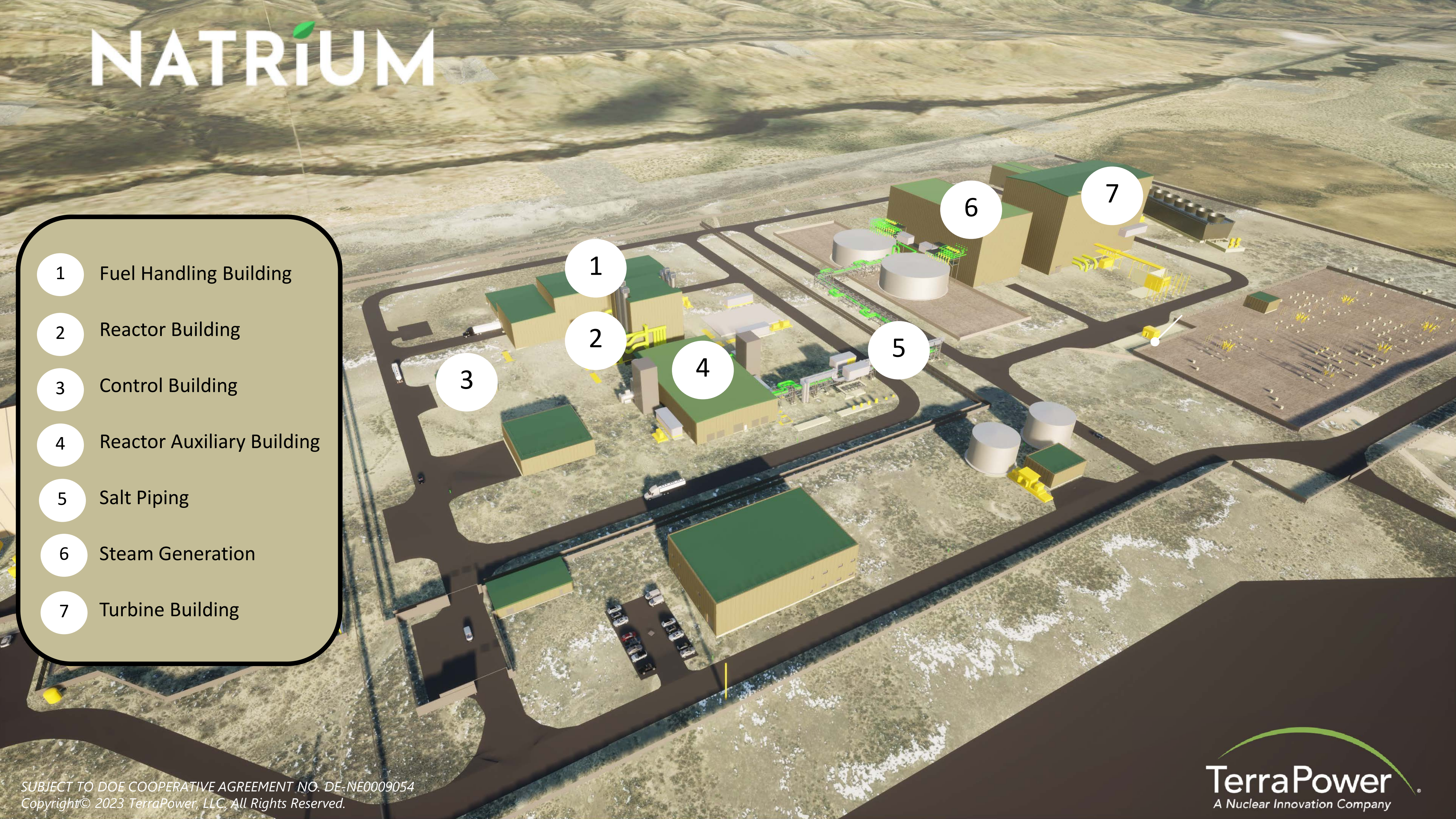
- Motor-driven control rod runback and scram follow
- Gravity-driven control rod scram
- Inherently stable with increased power or temperature

Cool

- In-vessel primary sodium heat transport (limited penetrations)
- Intermediate air cooling natural draft flow
- Reactor air cooling natural draft flow – always on

Contain

- Low primary and secondary pressure
- Sodium affinity for radionuclides
- Multiple radionuclides retention boundaries



1. Fuel Handling Building
2. Reactor Building
3. Control Building
4. Reactor Auxiliary Building
5. Salt Piping
6. Steam Generation
7. Turbine Building

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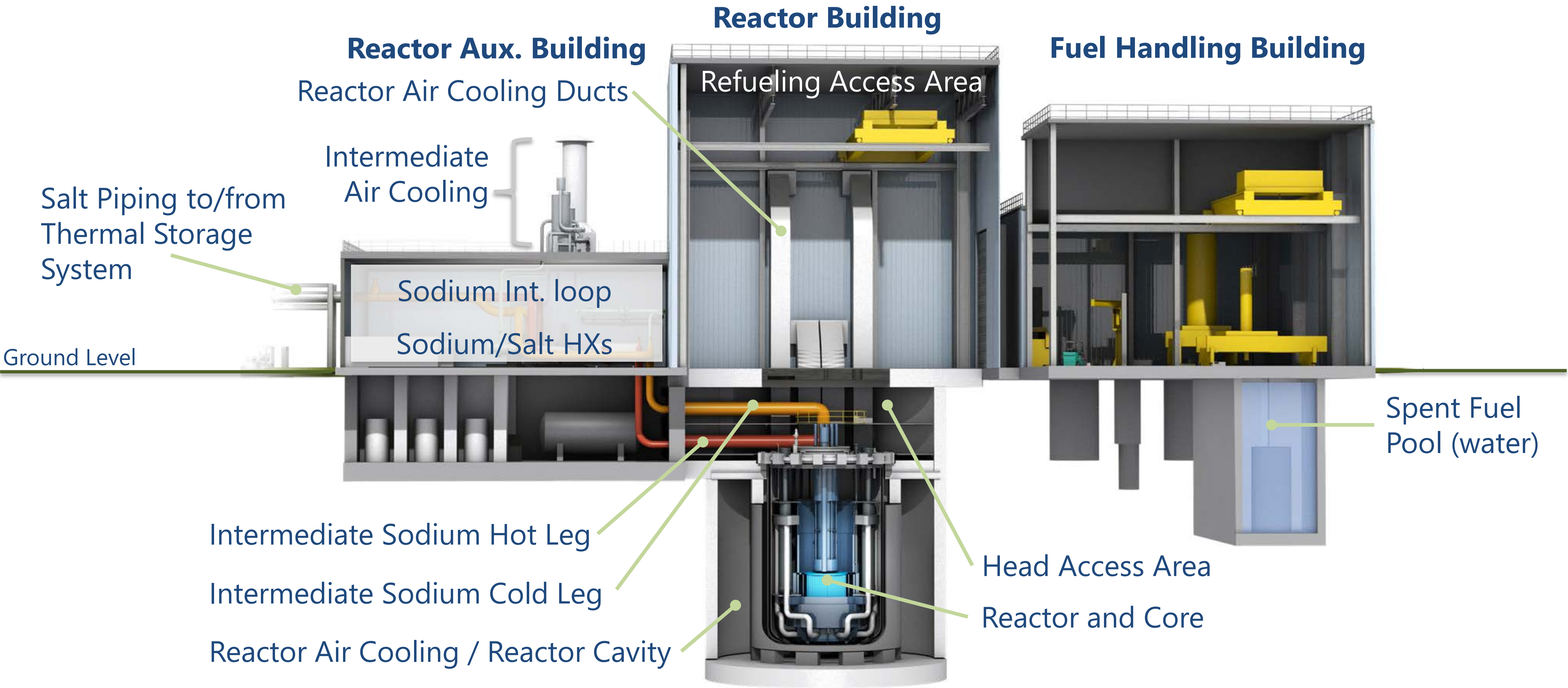


Reactor Building
Reactor Aux. Building
Fuel Handling Building
Refueling Access Area
Reactor Air Cooling Ducts
Intermediate
Air Cooling
Salt Piping to/from
Thermal Storage
System
Sodium Int. loop
Sodium/Salt HXs
Ground Level
Spent Fuel
Pool (water)
Intermediate Sodium Hot Leg
Intermediate Sodium Cold Leg
Reactor Air Cooling / Reactor Cavity
Head Access Area
Reactor and Core

SARRDL Definition and Scope

SARRDL Background and Scope

Specified **A**cceptable system **R**adionuclide **R**elease **D**esign **L**imit

- Concept first introduced during:
  - Next Generation Nuclear Plant (NGNP) pre-licensing activities (2006-2013), and
  - Review cycles of DG-1330 (later becoming Reg Guide 1.232).
- Method for monitoring integrity of TRISO Fuel Particles.
- Translates to any RN system which may contribute to annualized dose.

- Reg Guide 1.232 provides a current description of SARRDLs:

SARRDLs are established so that the most limiting license-basis event does not exceed the siting regulatory dose limits criteria at the EAB and LPZ and also that the 10 CFR 20.1301 annualized dose to the public are not exceeded at the EAB for Normal operation and AOOs.

SARRDL Background and Scope

- Relevant Regulations (cited in description)
  - 10 CFR 100.21(c)(1) (refers to requirement of Section 20.1301)

    Radiological effluent release limits associated with normal operation from the type of facility proposed to be located at the site can be met for any individual located offsite;
  - 10 CFR 20.1301 (refers to Normal Operations and AOOs)

    The TEDE dose to individual members of the public from the licensed operation does not exceed 0.1 rem (1 mSv) in a year.

- Additional considerations
  - Occupational Dose Limits (10 CFR 20.1201) (e.g. 5 rem TEDE annual limit)
  - 10 CFR 50.34 dose criteria met by SR SSC performance and analysis

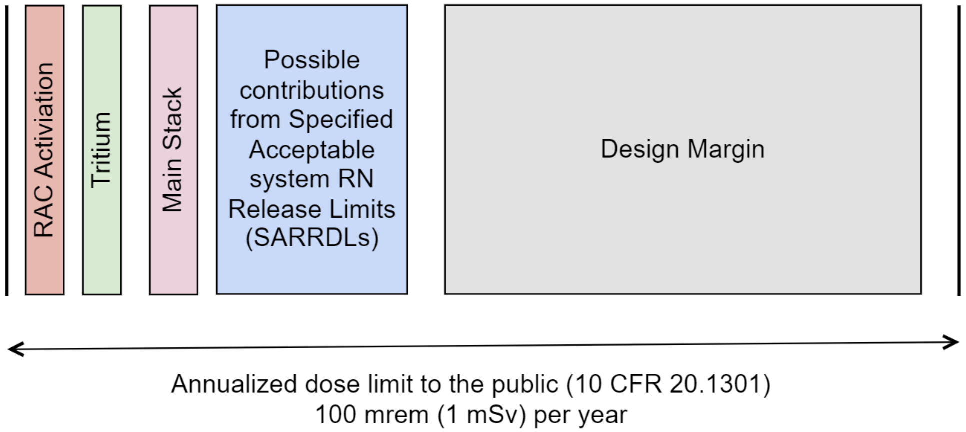
SARRDL Application

- Incorporated SARRDL into proposed PDC 10 "Reactor Design".

*The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that **specified acceptable system radionuclide release design limits are not exceeded** during any condition of normal operation, including the effects of anticipated operational occurrences.*

- SARRDL concept is integral with functional containment concept and overall Risk-Informed Performance-Based design and licensing approach.

- Combined effects of all effluents and SARRDLs adhere to 10 CFR 20.1301, with design margin.



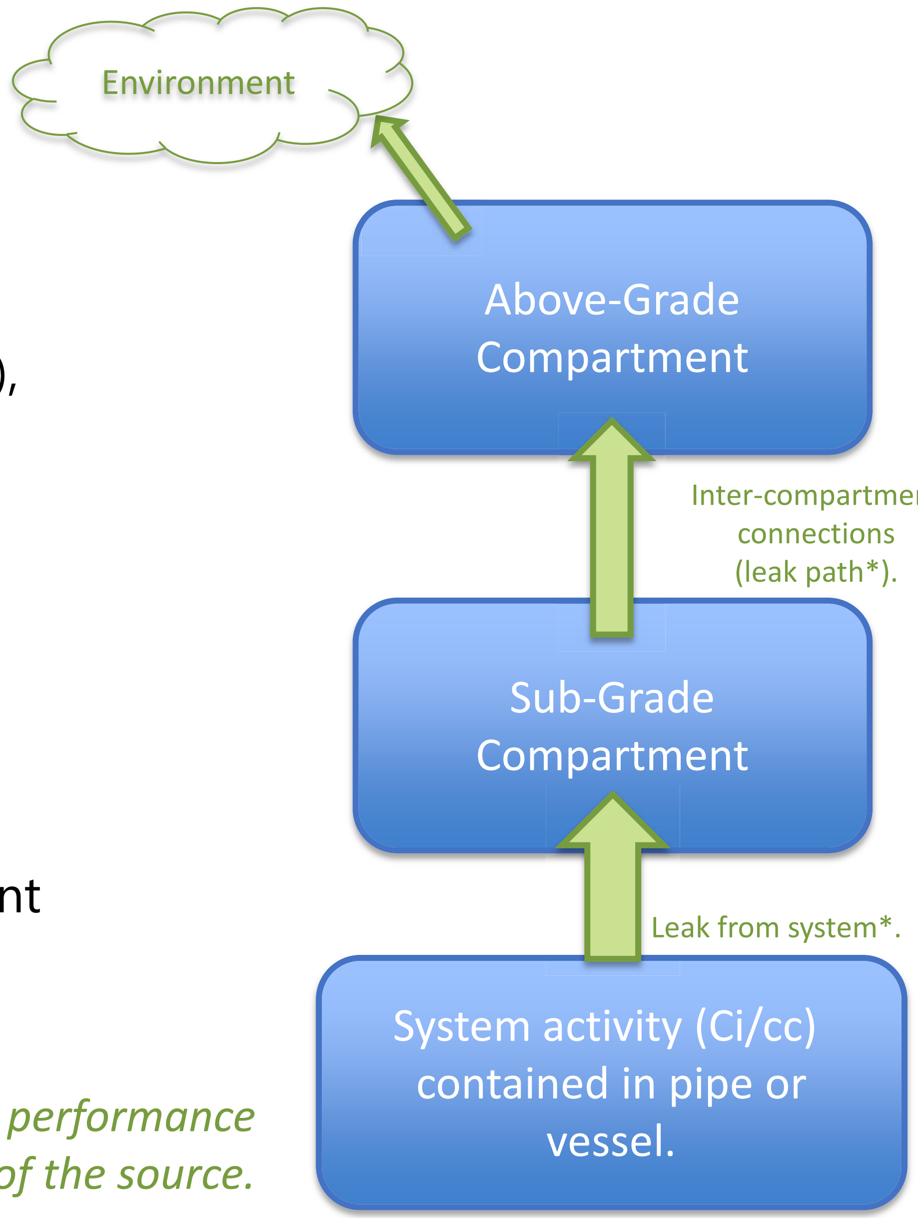
For illustration only.
Figure not to scale.

SARRDL – Example Systems

- Primary Sodium Cover Gas System
  - Example RNs: Ar-41, noble gases (possible fuel defects), Na aerosols, Cs vapors.
  - Activity continuously monitored during operation.

- Primary Sodium Processing System
  - Example RNs: Na-22, Na-24, Cs (possible fuel defects), corrosion products.
  - System fluid periodically sampled.

- Consider creditable RN retention and removal by subsequent functional containment compartments.



** SARRDL establishes performance criteria downstream of the source.*

Functional Containment

Functional Containment Definition

From SECY 18-0096:

A "functional containment" [is] a barrier, or a set of barriers taken together, that effectively limits the physical transport of radioactive material to the environment. (pg 2)

In terms of functional containment, a given SSC may have performance criteria associated with its role to limit effluent releases during normal operation and anticipated events as well as performance criteria associated with its role to retain radionuclides during design basis accidents or beyond-design-basis events. (pg 4)

Natrium PDC 16 – Functional Containment:

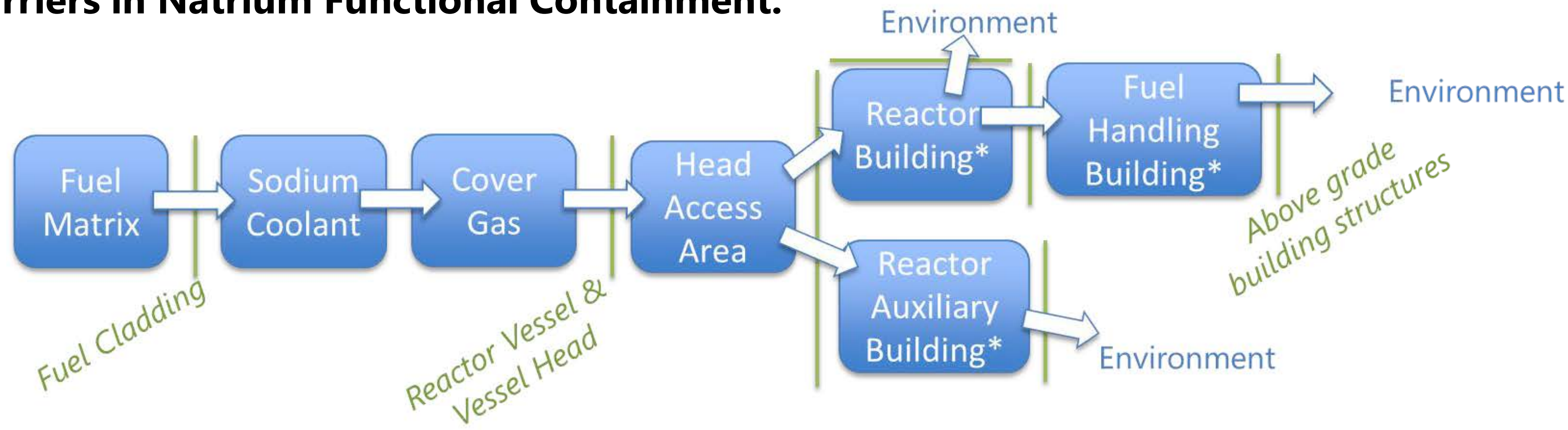
A functional containment, consisting of multiple barriers internal and/or external to the reactor and its cooling system, shall be provided to control the release of radioactivity to the environment and to ensure that the functional containment safety-significant design conditions are not exceeded for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

Application of Functional Containment

The Natrium design is well suited for functional containment:

- Low operating pressures & large margin to sodium boiling.
  -> *Eliminate high pressure releases and LOCAs.*
- High thermal conductivity coolant & passive emergency core cooling.
  -> *Effective core cooling reduces risk of fuel overheating.*

Multiple Barriers in Natrium Functional Containment:



**Sub-compartments within building provides additional confinement, depending on event.*

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Define Functional Containment Performance Criteria

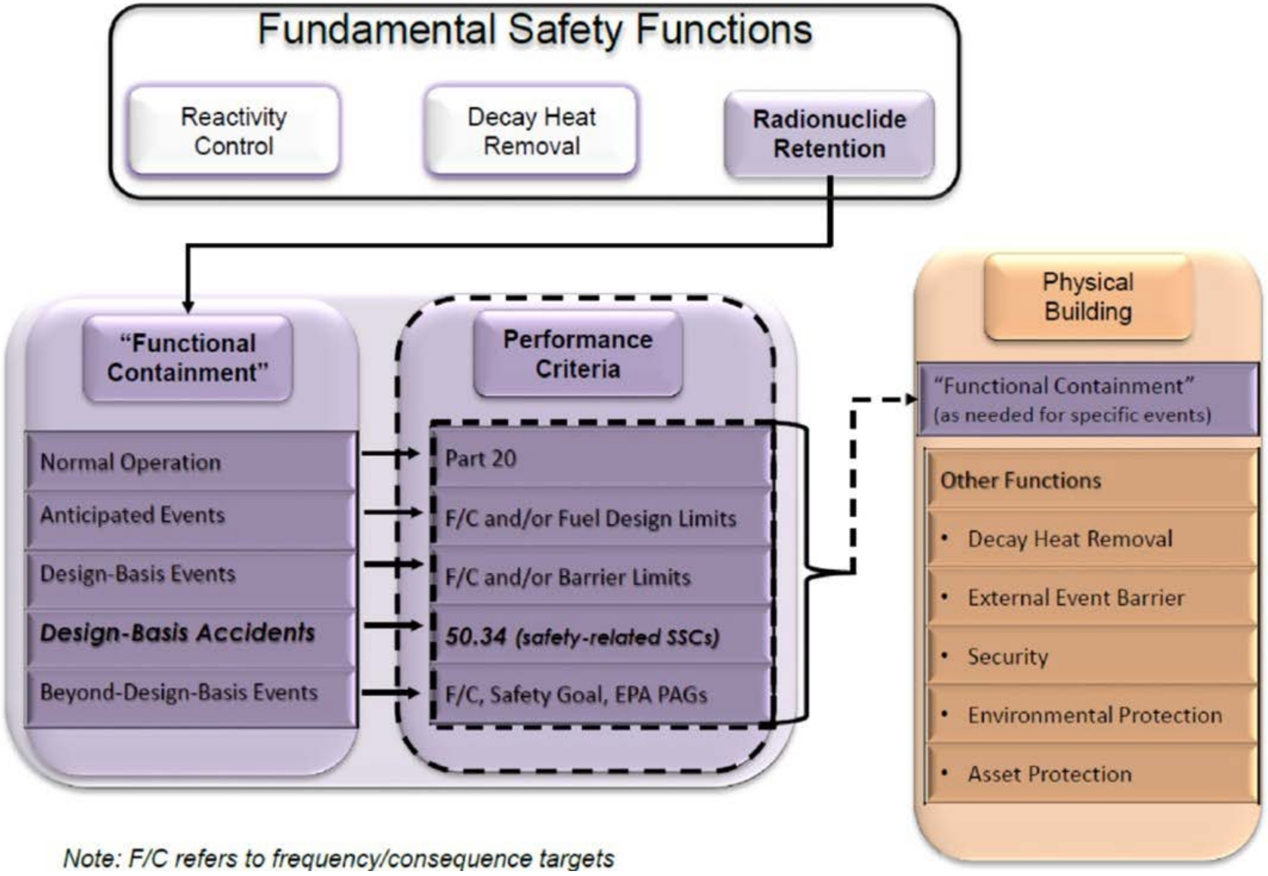

Figure 3 from SECY 18-0096 Enclosure 2

- Graded dose limit criteria according to event category from the LMP process.
- "Other Functions" considered as part of the building design requirements.
- Barrier performance is set to satisfy bounding event or source term release for a given configuration or event category.

Define Functional Containment Performance Criteria

Barrier performance criteria (e.g. leakage rates) are specified in terms of the performance needed to meet applicable dose consequences (e.g. F-C target, 10 CFR 100, effluent release limits) plus design margin.

Event Categories	Governing Dose Limit Criteria
Normal Operations	10 CFR Part 20, Subpart D (annualized dose to the public) System Criteria: SARRDLs
AOOs	F-C target: dose less than the 10 CFR 20 iso-risk line or 1.0 rem depending on event frequency System Criteria: SARRDLs
DBEs	F-C target: dose less than 1.0 rem (preferred) or 25 rem depending on event frequency
Design Basis Accidents	10 CFR 50.34 (e.g. worst 2-hr dose at EAB < 25 rem TEDE) EPZ methodology limits.
Beyond-Design Basis Events	F-C target: with margin EPZ methodology limits.

Iterative Performance Criteria Process

1. Specify bounding leakage rate for barriers.
2. Perform source term and radiological consequence scoping analyses.
3. Consult with design groups on leakage specifications/assumptions.
4. Refine 1st barrier performance with current design evolutions and align with SSC classification.
5. Re-run source term / rad con, evaluate against governing dose criteria.
   (Consider DBA event with limited barriers and removal mechanisms – confirm SR SSCs)

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Functional Containment PSAR Sections

- Discussion of functional containment will occur in relevant sections of the PSAR.
- Key PSAR sections for functional containment
  - **Section 1.3.2.1 FSF: Retaining RNs** – overview of functional containment concept, high level definition, make ties to specific plant systems
  - **Chapters 2 and 3** – describe role of functional containment in source term methodology and analysis of LBEs
  - **Chapter 5** – Safety Functions, Design Criteria, and SSC Classification – SSC classification and required safety functions.
  - **Chapter 6** – Safety Significant SSC Criteria and Capabilities – SRDC summary
  - **Chapter 7** – Descriptions of Safety-Significant SSCs that support functional containment
  - **Chapter 8** – Plant Programs – testing, inspection, and maintenance of barriers

Major Accident for Functional Containment Performance

LMP and Major Accident

- 10 CFR 50.34(a)(1)(ii)(D) requires an analysis to demonstrate containment performance:
  ...an applicant shall assume a fission product release from the core into the containment assuming that the facility is operated at the ultimate power level contemplated.

- Following NEI 18-04, plant siting is established through:
  - Evaluation of LBEs against:
    - F-C target curve (AOOs, DBEs, BDBEs),
    - 10 CFR 50.34 dose criteria (DBAs),
    - EPZ 4-day dose criteria (DBAs, BDBEs)
  - Evaluation of normal operation effluents adhering to 10 CFR 20.1201 & 20.1301.

- SR SSC performance is confirmed by evaluating DBAs defined by the LMP.

- Plan to postulate a major accident to demonstrate functional containment performance, and adherence to 10 CFR 50.34 dose criteria.

Major Accident Definition (Containment Performance)

- Typical major accidents assume substantial core damage which release large amounts of fission products to primary system.
- Event initiators and accident progressions differ widely between LWRs and SFRs.
- When evaluated by PRA, an equivalent event (with substantial fission product release) will be below the LBE frequency of 5e-7 (less frequent than BDBE).
- Natrium Major Accident Analysis
  - Select a mechanistic PRA sequence, resulting in substantial fission product release from the core to functional containment.
  - Design (demonstrable) leak rate assumed on Reactor Vessel Head (RES boundary).
  - Additional functional containment barriers and compartments available.
  - Analysis demonstrates offsite doses to be within 10 CFR 50.34 limits.

Major Accident Potential Examples

- Degraded Heat Removal and/or Failure of Pump Trip
  - Decay Heat + Pump Heat > Passive IAC + RAC
  - Gradual heating of core and primary sodium pool.
  - Eutectic / creep rupture failure of cladding (highest burnup fuel first, lowest burnup last).
  - No fuel melt expected.
  - Large fission product release (full core plus in-vessel storage).

- Unprotected Loss of Flow with Degraded Coastdown
  - Similar to degraded heat removal having gradual heat up of primary system.
  - Eutectic / creep rupture failure of cladding, no fuel melting.
  - More localized failure.
  - Passive heat removal available (minimizes extent of cladding failures).

- Unprotected Reactivity Insertion Events
  - Bulk primary system temperatures remain at normal conditions (~500°C Hot Pool Temp).
  - Fuel failure is most likely in higher burnup with some melting of the fuel matrix.
  - Fewer failures likely (localized to affected flow channels).
  - In-vessel spent fuel storage not impacted.

Questions?

Acronym List

AOO – Anticipated Operational Occurrence
CFR – Code of Federal Regulations
DBA – Design Basis Accidents
DBE – Design Basis Event
BDBE – Beyond Design Basis Event
EBR – Experimental Breeder Reactor
EAB – Exclusion Area Boundary
EPZ – Emergency Planning Zone
F-C – Frequency Consequence
FFTF – Fast Flux Test Facility
FSF – Fundamental Safety Function
IAC – Intermediate Air Cooling
LBE – Licensing Basis Event
LOCA – Loss of Coolant Accident
LMP – Licensing Modernization Project
LPZ – Low Population Zone
LWR – Light Water Reactor
NSRST - Non-Safety-Related with Special Treatment
NST – Non-Safety-Related with No Special Treatment
NGNP – Next Generation Nuclear Plant
PDC – Principal Design Criteria
PRA – Probabilistic Risk Assessment
PSAR – Preliminary Safety Analysis Report
RAC – Reactor Air Cooling
RES – Reactor Enclosure System
RN – Radionuclide
SARRDL – Specified Acceptable System Radionuclide Release Design Limit
SFR – Sodium-Cooled Fast Reactor
SR – Safety-Related
SRDC – Safety-Related Design Criteria
SSC – Structure, System, and Component
TREAT – Transient Reactor Test

ENCLOSURE 3

"Specified Acceptable System Radionuclide Release Design Limit, Functional Containment, and Major Accident Presentation Materials"
Presentation Material – Closed Meeting

Non-Proprietary (Public)

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SARRDL, Functional Containment, and Major Accident Update

a TerraPower & GE-Hitachi technology

TP-LIC-PRSNT-0007

SUBJECT TO DOE COOPERATIVE AGREEMENT NO. DE-NE0009054

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Portions of this presentation are considered proprietary and TerraPower, LLC requests it be withheld from public disclosure under the provisions of 10 CFR 2.390(a)(4).

Nonproprietary versions of this presentation indicate the redaction of such information using [[ ]]$^{(a)(4)}$, [[ ]]^{ECI}, or [[ ]]$^{(a)(4), ECI}$.

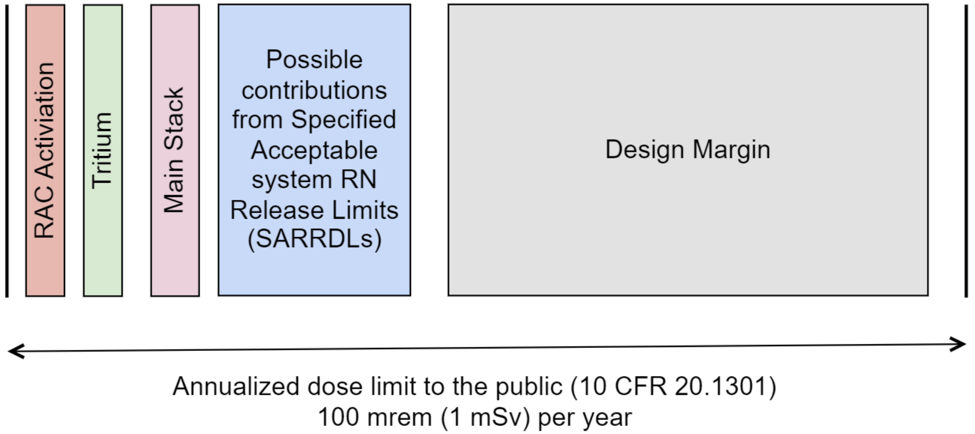
Table of Contents

SARRDL Definition and Example

SARRDL Approach

[[

- Design Limit vs. Safety Limit



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]](a)(4),ECI

- Assume that each established SARRDL simultaneously contributes to the annualized dose limit.

[[

]](a)(4),ECI

SARRDL – Scope of Systems

- Consult Radionuclide Identification and Screening
  - Identified and quantified radionuclide sources.
  - Estimated offsite dose potential for each source.
  - Develop SARRDLs for system/sources which have the potential to [[ ]](a)(4)

- Translation from RN Screening to SARRDL Definition

- Preliminary List of Systems for SARRDLs

[[

]](a)(4)

SARRDL Detection and Management

[[

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SARRDL Detection and Management (cont'd)

[[

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SARRDL Example – RCC / PHT

[[

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SARRDL Example – RCC / PHT (cont'd)

[[

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Functional Containment Definition and Performance Criteria

Functional Containment Update

- Completed plant level SSC classification. - Q1 2023
  - Identified functional containment barriers (RR) along with associated defense layer (DL), e.g., "DL3-RR1".
- Completed functional containment definition and strategy. - Q1 2023
  - Identification of key boundaries and interfacing systems.
  - Established high-level strategy for testing and inspection of barriers.
- Performed initial source term scoping runs with assumed barrier performance. - Q1 & Q2 2023
- Completed initial containment performance analysis and sensitivities for loss of cooling in the HAA. - Q2 2023
- Iterations with design groups on barrier/seal performance - on going.
- Update source term calculations with established barrier performance - on going.

Key Boundaries and Barrier Identification

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Key Boundaries and Barrier Identification (cont'd)

[[

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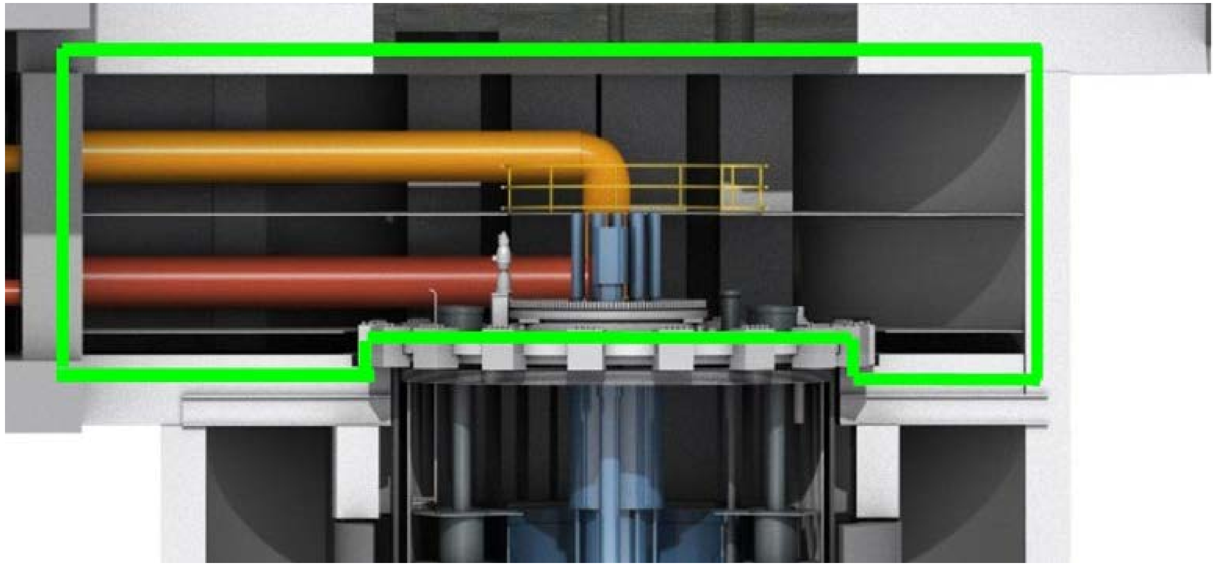
HAA Functional Containment Boundary



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HAA Seal and Well Seal

[[

- Provides critical barriers between HAA and RAC airflow path.
  - Maintains the RAC boundary for cooling flow.
  - Prevents hot RAC gases to enter HAA.

]](a)(4),ECI

- Welded seals (zero leakage).

[[

- Welds are accessible and would be inspected during outages.

]](a)(4),ECI

HAA Pressure-Driven Leakage and Mixing

[[

]](a)(4),ECI

Functional Containment – Testing and Inspection

[[

]](a)(4),ECI

Rotatable Plug Seal Performance Example

RPA SSC Classification and Performance

- Rotatable Plug Assembly (RPA) dynamically operates during refueling to assist with fuel movements of the In-Vessel Transfer Machine (IVTM).

- RPA has sealing functions both during normal operations and refueling modes.

- Normal Operation – RPA is seated on metal ledge seal and bolted.

- Refueling – RPA is lifted off metal ledge seal [[ ]][(a)(4),ECI] and allowed to rotate as needed for fuel movements.

- Inflatable seals (static & dynamic) provide sealing functions in both configurations.

Rotatable Plug Assembly – Key Seals

[[

]](a)(4),ECI

RPA Seal Performance Events

[[

]](a)(4)

RPA Seal Performance – Preliminary Analysis

[[

]](a)(4),ECI

Functional Containment Performance

Functional Containment and Major Accident

- 10 CFR 50.34(a)(1)(ii)(D) requires an evaluation of postulated "...fission product release from the core into the containment..." to demonstrate the performance of containment and removal systems.

- The Natrium™ major accident will evaluate a severe event with a substantial fission product release from the core to functional containment with design leak rates.

- This accident type is not a credible LBE nor DBA as assessed by PRA (<5x10-7/ plant year).

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Major Accident Potential Examples

- Degraded Heat Removal and/or Failure of Pump Trip
  - Decay Heat + Pump Heat > Passive IAC + RAC
  - Gradual heating of core and primary sodium pool.
  - Eutectic / creep rupture failure of cladding (highest burnup fuel first, lowest burnup last).
  - No fuel melt expected.
  - Large fission product release (full core plus in-vessel storage).

- Unprotected Loss of Flow with Degraded Coastdown
  - Similar to degraded heat removal having gradual heat up of primary system.
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  - More localized failure.
  - Passive heat removal available (minimizes extent of cladding failures).

- Unprotected Reactivity Insertion Events
  - Bulk primary system temperatures remain at normal conditions (~500°C Hot Pool Temp).
  - Fuel failure is most likely in higher burnup with some melting of the fuel matrix.
  - Fewer failures likely (localized to affected flow channels).
  - In-vessel spent fuel storage not impacted.

Major Accident Example

- Degraded Heat Removal
  - Failure to trip all sodium pumps.
  - Successful plant trip.
  - Active IAC - Failed.
  - Passive IAC and RAC – Available.
  - Heat addition exceeds heat removal.
  - Functional containment intact.

[[

]](a)(4),ECI

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DG-1404 Proposed Options

- Draft Guide DG-1404 (May 2023), Section C.3(c) - Two options for meeting 10 CFR 50.34(a)(1)(ii)(D) via use of the LMP.

- Option 1 - Use DBA dose consequence results from the LMP-based approached. However, applicant may need to include an exemption from the regulation that requires an assumed "major accident" for containment performance.

- Option 2 - Use the greater of dose consequence results from the bounding DBA and from a bounding BDBE. "This option provides an acceptable approach ... that precludes the need for an exemption ... as long as the bounding BDBE involves or bounds an event sequence meeting the description of a major accident..."


Questions?

Acronym List

AOO – Anticipated Operational Occurrence
BLTC – Bottom Loading Transfer Cask
CRDM – Control Rod Drive Mechanism
CFR – Code of Federal Regulations
DBA – Design Basis Accident
DBE – Design Basis Event
BDBE – Beyond Design Basis Event
EBR – Experimental Breeder Reactor
EAB – Exclusion Area Boundary
EPZ – Emergency Planning Zone
EVHM – Ex-Vessel Handling Machine
EVST – Ex-Vessel Storage Tank
F-C – Frequency Consequence
FFTF – Fast Flux Test Facility
FSF – Fundamental Safety Function
HAA – Head Access Area
IAC – Intermediate Air Cooling
IVTM – In-Vessel Transfer Machine
LBE – Licensing Basis Event
LOCA – Loss of Coolant Accident
LMP – Licensing Modernization Project
LPZ – Low Population Zone
LWR – Light Water Reactor
NSRST - Non-Safety-Related with Special Treatment
NST – Non-Safety-Related with No Special Treatment
NGNP – Next Generation Nuclear Plant
NHV – Nuclear Island HVAC

PHT – Primary Heat Transport
PIC – Pool Immersion Cell
PDC – Principal Design Criteria
PRA – Probabilistic Risk Assessment
PSAR – Preliminary Safety Analysis Report
RAC – Reactor Air Cooling
RCC – Reactor Core and Core Components
RES – Reactor Enclosure System
RN – Radionuclide
RPA – Rotatable Plug Assembly
RWG – Gaseous Radiological Waste
RVH – Reactor Vessel Head
RXB – Reactor Building
SARRDL – Specified Acceptable System Radionuclide Release Design Limit
SCG – Sodium Cover Gas
SPS – Sodium Processing System
SFR – Sodium-Cooled Fast Reactor
SR – Safety-Related
SRDC – Safety-Related Design Criteria
SSC – Structure, System, and Component
T-H – Thermal Hydraulic
TREAT – Transient Reactor Test