TerraPower

NATRIUM

Regulatory Gap Analysis

a TerraPower & GE-Hitachi technology

NATD-LIC-PRSNT-0025

SUBJECT TO DOE COOPERATIVE AGREEMENT NO. DE-NE0009054

Presentation Outline

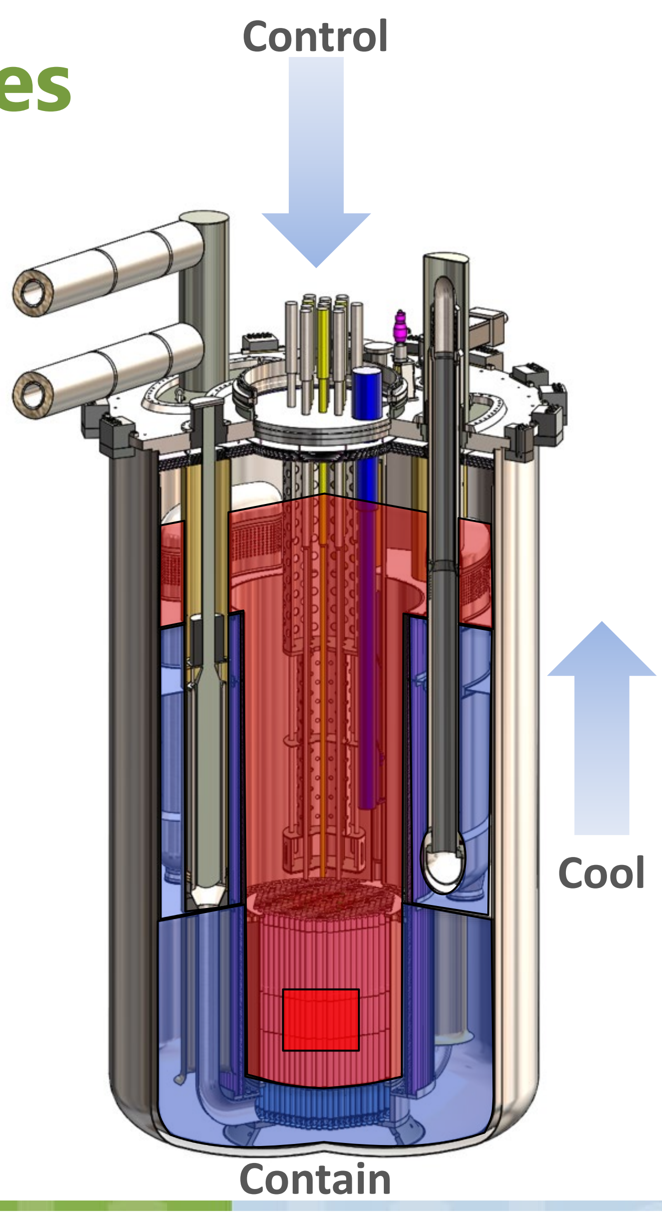
- Natrium™ Reactor Overview
- Analysis Methodology
- Analysis Results
- Potential Design or Technology Related Exemptions
- Potential Non-Design Related Exemptions
- Regulations Under Consideration
- Questions and Discussion

Natrium Reactor Overview

- Regulatory Engagement Plan was submitted 6/8/2021
- Construction Permit Application submittal planned for 8/2023
- Pre-application interactions are ongoing, intended to reduce regulatory uncertainty and facilitate the NRC's understanding of the Natrium advanced reactor and its safety case
- TerraPower is demonstrating the ability to design, license, construct, startup and operate the Natrium plant within a seven-year timeframe

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

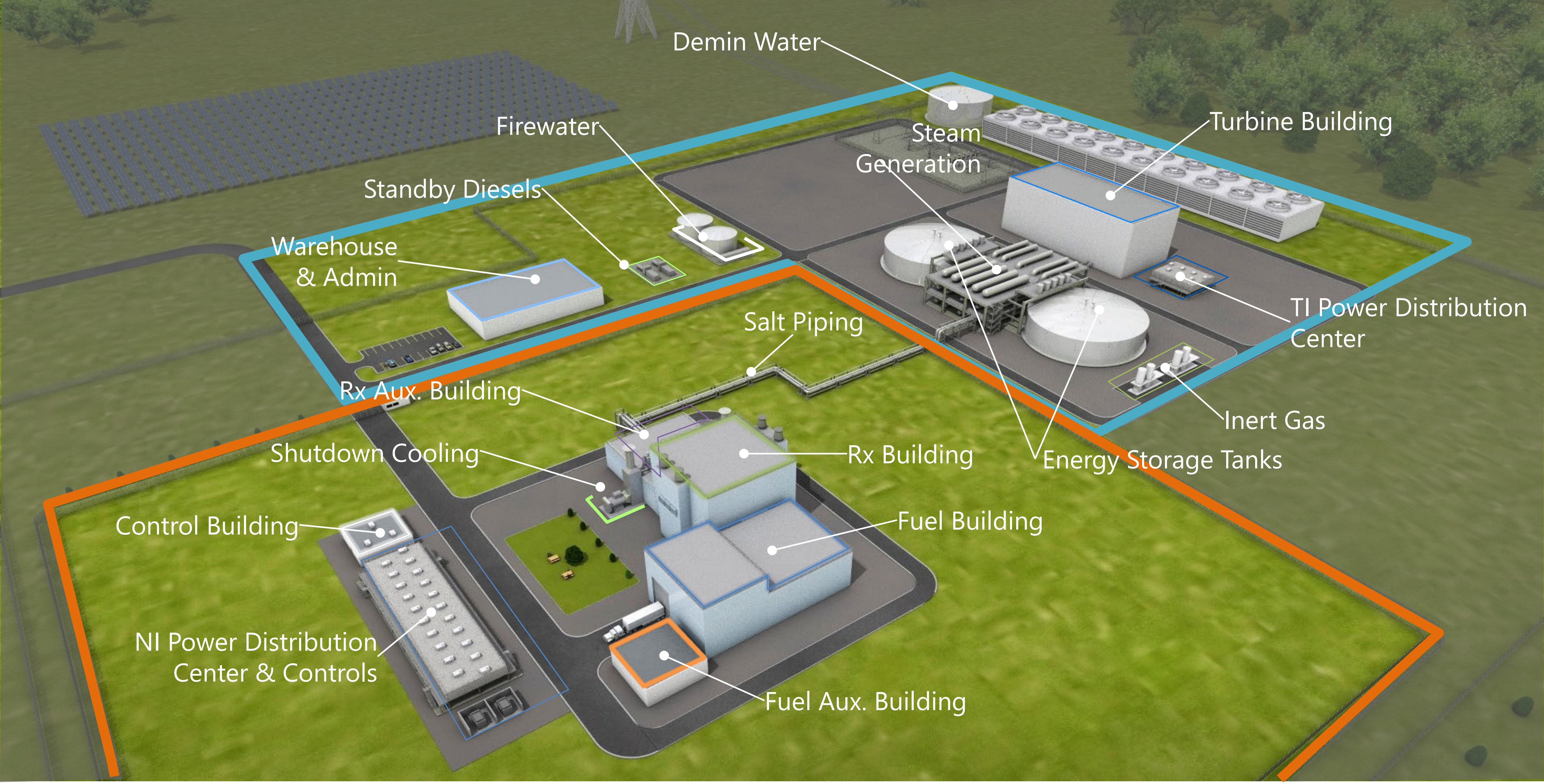
- Motor-driven control rod runback
- Gravity-driven control rod scram
- Inherently stable with increased power or temperature

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



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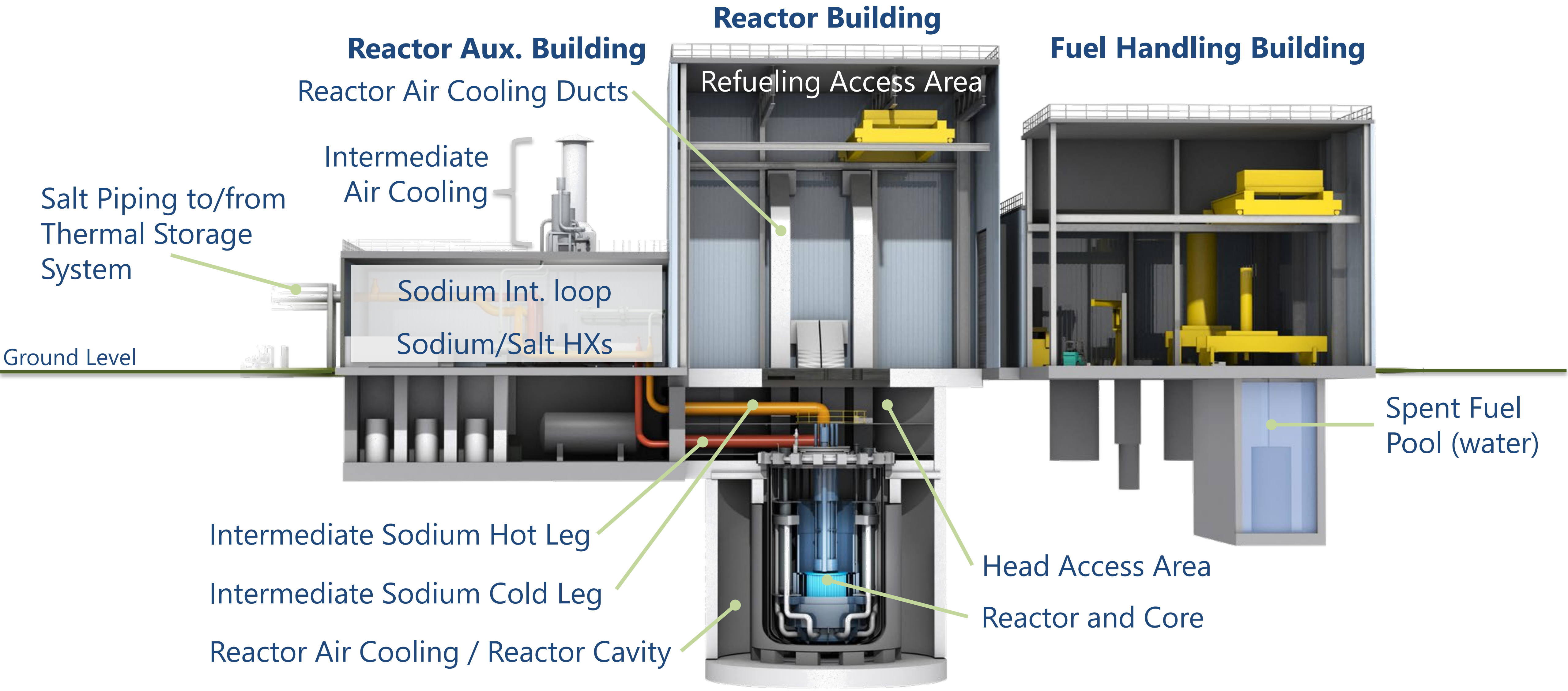

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Analysis Methodology

- Scope: 10 CFR Parts 1-199, except those that are reserved
- Four initial categories
  - Applicable with design implications
  - Applicable with entry condition (e.g., SSC specific)
  - Applicable with no design implications (e.g., programmatic)
  - Not applicable (e.g., date driven, technology specific, application pathway)
- Informed by Updated NRC Staff Draft White Paper "Analysis of Applicability of NRC Regulations for Non-Light Water Reactors" (ML21175A287)

Analysis Methodology

- Additional review of applicable regulations to identify potential exemptions
- Additional review of not applicable regulations to identify underlying bases necessary to consider for a comprehensive licensing evaluation
- Three final categories
  - Applicable Design Related
  - Applicable Non-Design Related
  - Not Applicable

Analysis Results

- Tables of categorized regulations best presented in White Paper submittal planned for August 2022
- Differences with draft guidance
  - 10 CFR 50 Appendix F: only applies to fuel reprocessing plants
  - 10 CFR 81: process for licensing NRC patented inventions for use
  - 10 CFR 50.34(f) TMI requirements: all are regulations with not-met entry conditions

Analysis Results

- 50.34(f)(1)(iii)- Reactor coolant pump seal damage
- 50.34(f)(2)(vi)- High point venting of reactor coolant system
- 50.34(f)(2)(x)- Relief and safety valves

Analysis Results

- 50.34(f)(2)(xiv)- Containment isolation
- 50.34(f)(2)(xv)- Containment purging
- 50.34(f)(2)(xvii)- Control room instrumentation for containment functions
- 50.34(f)(3)(iv)- Dedicated containment penetrations
- 50.34(f)(3)(vi)- Containment

Potential Design or Technology Related Exemptions

Potential Non-Design Related Exemptions

Regulations Under Consideration

HALEU Fuel

- 10 CFR 50.68(b) *Each licensee shall comply with the following requirements in lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24: […]*
  - *(7) The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight.*

- Exemption request will change the maximum enrichment of Uranium-235 to less than 20%.
- The Natrium design includes the use of high-assay low enriched uranium (HALEU) fuel with uranium enrichment that is higher than that specified in 10 CFR 50.68(b)(7).

- Consistent with Updated NRC Staff Draft White Paper "Analysis of Applicability of NRC Regulations for Non-Light Water Reactors" (ML21175A287)

ECCS Analysis

- 10 CFR 50.34(a)(4) *[…] Analysis and evaluation of ECCS cooling performance and the need for high point vents following postulated loss-of-coolant accidents must be performed in accordance with the requirements of § 50.46 and § 50.46a of this part for facilities for which construction permits may be issued after December 28, 1974.*

- Exemption request will remove the requirement to provide an analysis and evaluation of ECCS cooling performance in accordance with the requirements of 10 CFR 50.46.
- 10 CFR 50.46 is applicable only to LWRs, therefore is not applicable to the Natrium design.
- The first sentence of 10 CFR 50.34(a)(4) will not be impacted by the exemption request, i.e.:
  - *A preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility […].*

- Consistent with Updated NRC Staff Draft White Paper "Analysis of Applicability of NRC Regulations for Non-Light Water Reactors" (ML21175A287)

ECCS Analysis

- 10 CFR 50.34(b)(4) *[...] Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of § 50.46 for facilities for which a license to operate may be issued after December 28, 1974.*

- Exemption request will remove the requirement to provide an analysis and evaluation of ECCS cooling performance in accordance with the requirements of 10 CFR 50.46.
- 10 CFR 50.46 is applicable only to LWRs, therefore is not applicable to the Natrium design.
- The first sentence of 10 CFR 50.34(b)(4) will not be impacted by the exemption request. i.e.:
  - *A final analysis and evaluation of the design and performance of structures, systems, and components [...].*

- Consistent with Updated NRC Staff Draft White Paper "Analysis of Applicability of NRC Regulations for Non-Light Water Reactors" (ML21175A287)

Fission Product Release

- 10 CFR 50.34(a)(1)(ii)(D) *[…] In performing this assessment, 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. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable site characteristics, including site meteorology, to evaluate the offsite radiological consequences. Site characteristics must comply with part 100 of this chapter. The evaluation must determine that […]*

- Exemption request will remove the requirements for an assumption of fission product release from the core into containment and the use of a demonstrable containment leak rate.
- The phrasing "from the core into the containment" and "demonstrable containment leak rate" of the regulation is not consistent with the use of a functional containment design.

- Consistent with Updated NRC Staff Draft White Paper "Analysis of Applicability of NRC Regulations for Non-Light Water Reactors" (ML21175A287)

Potential Design or Technology Related Exemptions

Potential Non-Design Related Exemptions

Regulations Under Consideration

Shift Technical Advisor

- 10 CFR 50.120(b) *The training program must be derived from a systems approach to training as defined in 10 CFR 55.4, and must provide for the training and qualification of the following categories of nuclear power plant personnel: [...]*
  - *(iii) Shift technical advisor.*

- Exemption request will remove requirement for a training program for STA personnel.
- Topics applicable to a typical STA training and qualification program (e.g., generic fundamentals, mitigating core damage, transient accident analysis) will be incorporated into the Natrium operator training program.
- Additionally, inherent safety features of the Natrium design result in lower operational complexity and provide improvement in overall plant safety including reduced reliance on operator actions, precluding the necessity to train and qualify personnel to perform specific STA roles.
- Consistent with considerations identified in SECY-21-0039, "Elimination of the Shift Technical Advisor for the NuScale Design."

Nuclear Island to Energy Island Interface

- 10 CFR 50.10(a)(1) *Activities constituting construction are the driving of piles, subsurface preparation, placement of backfill, concrete, or permanent retaining walls within an excavation, installation of foundations, or in-place assembly, erection, fabrication, or testing, which are for:*
  - *(iv) SSCs whose failure could cause a reactor scram or actuation of a safety-related system.*
- 10 CFR 50.65(b) *The scope of the monitoring program specified in paragraph (a)(1) of this section shall include safety related and nonsafety related structures, systems, and components, as follows:*
  - *(2) Nonsafety related structures, systems, or components:*
    - *(iii) Whose failure could cause a reactor scram or actuation of a safety-related system.*

- Exemption request will be submitted with the Nuclear Island Energy Island Interface Topical Report.

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Potential Design or Technology Related Exemptions

Potential Non-Design Related Exemptions

Regulations Under Consideration

Reactor Coolant Pressure Boundary

- 10 CFR 50.2 *Reactor coolant pressure boundary means all those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are[…]*

- Exemption request is under consideration to provide a definition of "reactor coolant boundary" that is applicable to the Natrium design, and to replace the term "reactor coolant pressure boundary" with the term "reactor coolant boundary" within the regulations described on the following slides.
- The 10 CFR 50.2 definition of reactor coolant pressure boundary is limited to "boiling and pressurized water-cooled nuclear power reactors," therefore the Natrium design does not include a reactor coolant pressure boundary as defined in 10 CFR 50.2.

Reactor Coolant Pressure Boundary

- 10 CFR 50.2 related regulations:
  - 10 CFR 50.36(c)(2)(ii)
    - 10 CFR 50.36(c)(2)(ii) *A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:*
      - *(A) Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. [...]*
  - 10 CFR 50.49(b)
    - 10 CFR 50.49(b) *Electric equipment important to safety covered by this section is:*
      - *(1) Safety-related electric equipment.*
        - » *(i) This equipment is that relied upon to remain functional during and following design basis events to ensure—*
          - *(A) The integrity of the reactor coolant pressure boundary [...]*

- Consistent with Updated NRC Staff Draft White Paper "Analysis of Applicability of NRC Regulations for Non-Light Water Reactors" (ML21175A287)

Reactor Coolant Pressure Boundary

- 10 CFR 50.2 related regulations:
  - 10 CFR 50.65(b)
    - 10 CFR 50.65(b) *The scope of the monitoring program specified in paragraph (a)(1) of this section shall include safety related and nonsafety related structures, systems, and components, as follows:*
      - *(1) Safety-related structures, systems and components that are relied upon to remain functional during and following design basis events to ensure the integrity of the reactor coolant pressure boundary, [...]*
  - 10 CFR 50 Appendix S, III. Definitions
    - 10 CFR 50 Appendix S, III. Definitions
      - *Structures, systems, and components required to withstand the effects of the safe-shutdown earthquake ground motion or surface deformation are those necessary to assure:*
        - » *(1) The integrity of the reactor coolant pressure boundary; [...]*

- Consistent with Updated NRC Staff Draft White Paper "Analysis of Applicability of NRC Regulations for Non-Light Water Reactors" (ML21175A287)

General Design Criteria

- Exemption request is under consideration to remove, from 10 CFR 50 Appendix A, "The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units."
- Principal Design Criteria have been developed consistent with Regulatory Guide 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors."

Emergency Preparedness

- Exemption requests from certain portions of the following regulations would be necessary without codification of 10 CFR 50.160.
  - 10 CFR 50.33(g), Radiological Emergency Plans of State and Local Governments
  - 10 CFR 50.47(b), Offsite Emergency Plan Standards
  - 10 CFR 50.47(c)(2), Emergency Planning Zones
  - 10 CFR 50 Appendix E, Offsite Emergency Response Considerations

- Consistent with Updated NRC Staff Draft White Paper "Analysis of Applicability of NRC Regulations for Non-Light Water Reactors" (ML21175A287)

Seismic Design Criteria

- Exemption requests from certain portions of the following regulations may be needed to incorporate the ASCE 43-19 risk-informed performance-based seismic design guidance for Natrium SSCs:
  - 10 CFR 50 Appendix S, Earthquake Engineering Criteria for Nuclear Power Plants
  - 10 CFR 100.23, Geologic and Seismic Siting Criteria

- ASCE 43-19 provides a RIPB seismic design approach.

Pressurized Thermal Shock Events

- 10 CFR 50.61(a)(2) *Pressurized Thermal Shock Event means an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel.*
- 10 CFR 50.34(b)(9) *A description of protection provided against pressurized thermal shock events, including projected values of the reference temperature for reactor vessel beltline materials as defined in § 50.61(b)(1) and (b)(2).*

- The description of pressurized thermal shock event in 10 CFR 50.61(a)(2) is not applicable to the Natrium design.
- The Natrium reactor operates at near atmospheric conditions and does not experience pressurized thermal shock events.

- Consistent with Updated NRC Staff Draft White Paper "Analysis of Applicability of NRC Regulations for Non-Light Water Reactors" (ML21175A287)

Questions?

Acronym List

AC – Alternating Current
ASCE- American Society of Civil Engineers
CFR – Code of Federal Regulations
EBR – Experimental Breeder Reactor
ECCS – Emergency Core Cooling System
FFTF – Fast Flux Test Facility
GDC – General Design Criteria
HALEU – High-Assay Low-Enriched Uranium
HX- Heat Exchanger
LWR- Light Water Reactor
RIPB- Risk-Informed Performance-Based
SSC – Structures, Systems, and Components
SFR – Sodium Fast Reactor
STA – Shift Technical Advisor
TMI- Three Mile Island
TREAT – Transient Reactor Test