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Principal Design Criteria (PDC) Development Using the Licensing Modernization Project (LMP)

Objectives

- Describe modifications to SFR-DC
- Present preliminary Natrium™ PDC
- Discuss PDC Topical Report prior to Q1 2022 submittal

Natrium Reactor Licensing Overview

- Regulatory Engagement Plan submitted 6/8/2021
- 10 CFR 50 licensing process will be followed
  - Construction Permit Application 8/2023
  - Operating License Application 3/2026
- Numerous pre-application interactions are planned to reduce regulatory uncertainty and facilitate the NRC's understanding of Natrium technology and its safety case
- LMP (NEI 18-04), as endorsed by Regulatory Guide 1.233, will support this application

Natrium Reactor Licensing Overview

- Each pre-application interaction will build upon risk insights from prior interactions to demonstrate the Natrium reactor's safety case.

- Upcoming meetings and presentations include:
  - Source Term Methodology
  - Core Flow Blockage
  - Testing Plan and Methodology
  - Functional Containment Strategy
  - Codes and Standards Review

Advanced Reactor Demonstration Program

- Demonstrate the ability to design, license, construct, startup and operate the Natrium reactor within the Congressionally mandated seven-year timeframe
- Include improvements in safety, security, economics, and environmental impacts
- Utilize a simple, robust, reliable, and proven safety profile
- Lower emissions by initiating the deployment of a fleet of Natrium reactors – Demonstrate that the plants can be built economically and that they will be attractive for future owner/operators

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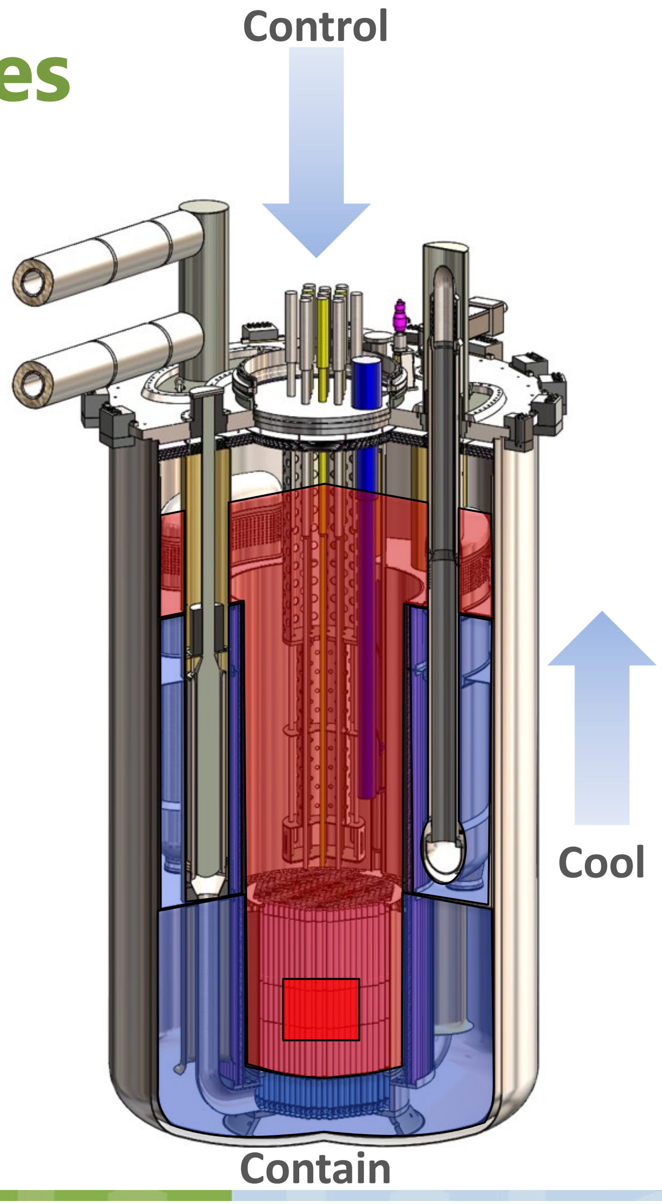
Natrium Safety Features

- 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 radionuclide retention boundaries

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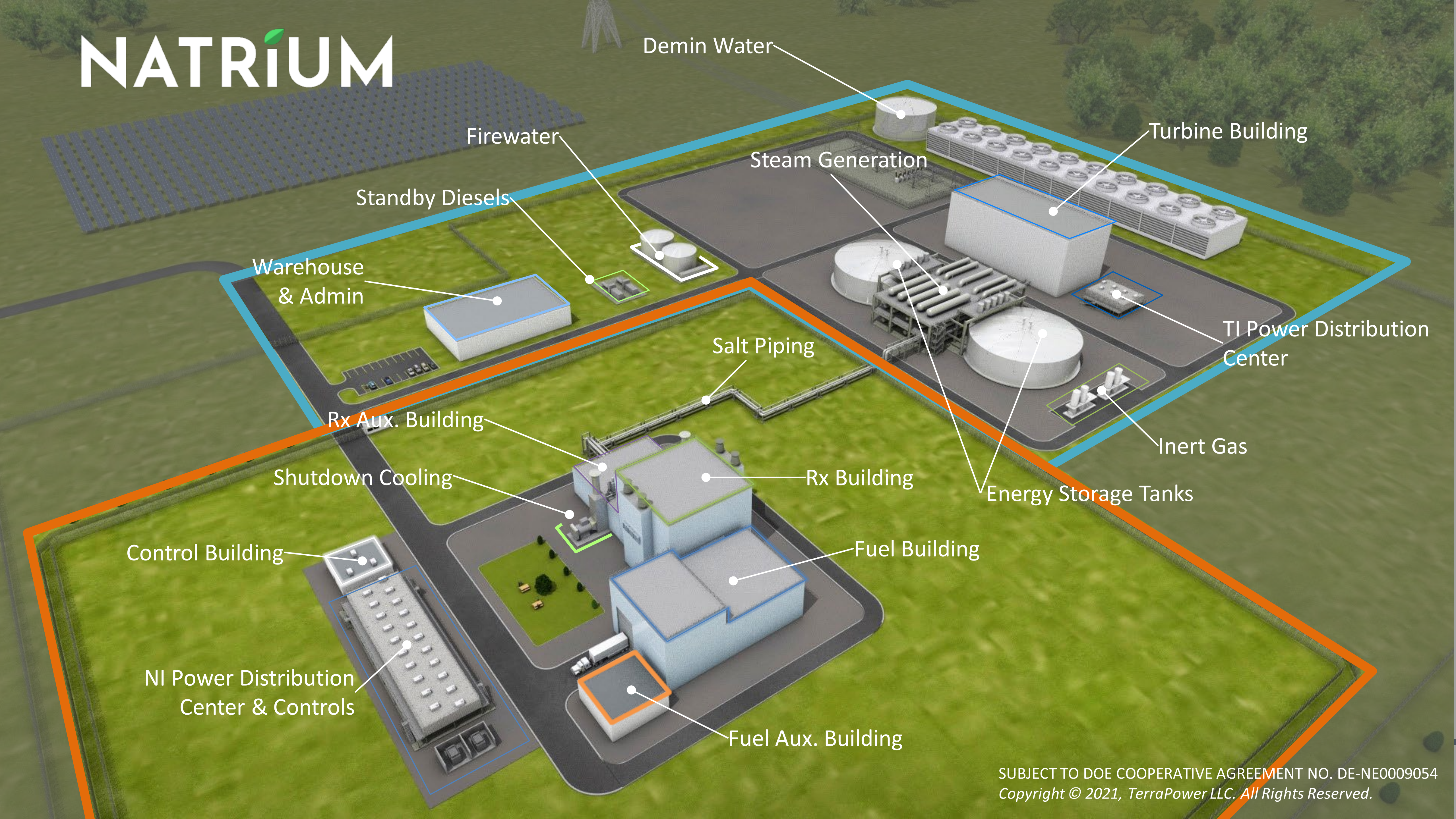

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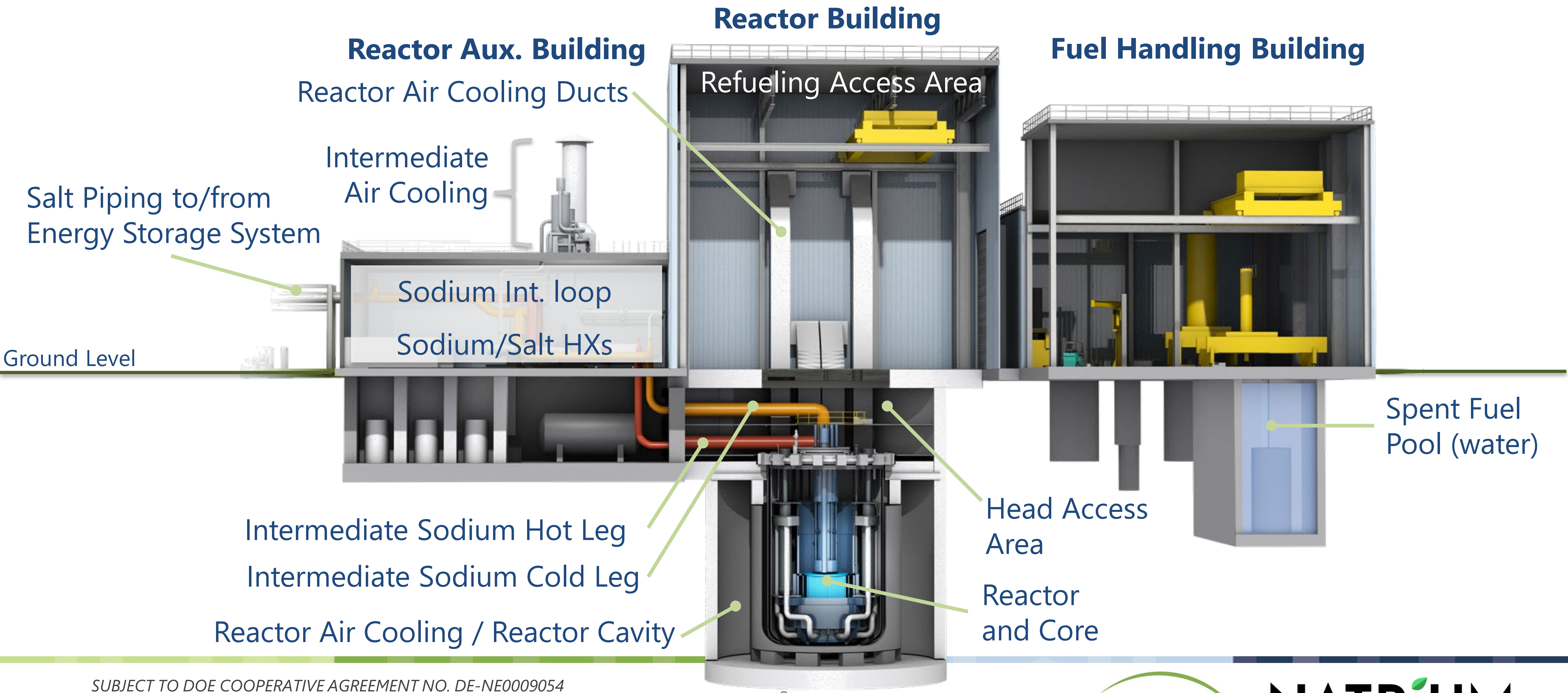
Plant Overview



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PDC Development Guidance

RG 1.232, *Guidance for Developing Principal Design Criteria for non-LWRs*

- Section B, Role of the GDC for Non-LWRs
  - "Together, [10CFR50.34 and 10CFR50 App A] recognize that different requirements may need to be adapted for non-LWR designs and that the GDC in 10CFR50 Appendix A **are not regulatory requirements** for non-LWR designs but provide guidance in establishing the PDC for non-LWR designs."

- Section C, Intended use of this Regulatory Guide
  - "…the non-LWR design criteria…are intended to provide stakeholders with insight into the staff's views on how the GDC could be interpreted…however, these are **not considered to be final or binding**…"
  - "Applicants…are free to choose among the ARDC, SFR-DC, or MHTGR-DC to **develop each PDC** after considering the underlying safety basis for the criterion and evaluating the [NRC] rationale for the adaptation…"
  - "Since the GDC in 10CFR50 Appendix A are not regulatory requirements for non-LWR designs…applicants **would not need to request an exemption** from the GDC…"

LMP as endorsed by RG 1.233

Goals for Natrium PDC Development

- Maximize adherence to RG 1.232, App B
  - Demonstrate agreement with the NRC staff's technical rationale for SFR-DC
  - Minimal adaption supports an efficient development and review process
  - Some SFR-DC language, and some SFR-DCs in their entirety, are not applicable

- Effectively capture RIPB licensing basis concepts
  - Fundamental elements of the Natrium design
  - Use of NRC-approved methodology supports an efficient development and review process

- Forward-looking and accommodate future guidance
  - Consider impact of PDC language on design/license changes in the operating phase
  - NEI 21-07
  - Potential industry initiatives (e.g., US NIC, NEI ARRTF)
  - Draft NRC guidance

Modifications to SFR-DC

- Use of "safety-significant"
  - Most analogous to GDC concept of "important to safety"
    - Ensures appropriate SSCs (SR and NSRST) are within scope of PDC

- The cohesive relationship of SARRDL and FC Target
  - SAFDL is not risk-informed which is inconsistent with LMP
    - SARRDL is an inherent concept of LMP

- SR SSC responses to AOOs
  - Need to ensure PDC interpretation does not result in a deviation from LMP
    - Clearly define the role of DL2 within the PDC Topical Report

Modifications to SFR-DC

- RAC replaces multiple LWR systems
  - Must select the correct PDC title, language, and structure
    - Ensure appropriate allocation of required safety functions
    - Most non-LWRs propose a single system for RHR/ECCS

- Functional containment
  - Section V SFR-DC are similar to current GDC, which is not conducive to LMP or the Natrium design

- Additional PDC Considerations

Use of “Safety-Significant”

Use of "Safety-Significant"

Only change what is needed to support use of "safety-significant"

– Example, SFR-DC 1, *Quality Standards and Records*

Criterion	SFR-DC Title and Content
1	*Quality standards and records.* Same as GDC Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Proposed Natrium PDC 1
Structures, systems, and components ~~important to safety~~ that are safety-significant shall be designed, fabricated, erected, and tested to quality standards commensurate with the ~~importance~~ safety significance of the ~~safety~~ functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the ~~required~~ safety significance of the function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety-significant functions. Appropriate records of the design, fabrication, erection, and testing of these structures, systems, and components ~~important to safety~~ shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

DRAFT

Black text is retained from RG 1.232, Appendix B, SFR-DC
Red text is removed from RG 1.232, Appendix B, SFR-DC
Blue text is inserted to adapt the PDC to the Natrium design

Use of "Safety-Significant"

Proposed Natrium PDC 1

Structures, systems, and components ~~important to safety~~ that are safety-significant shall be designed, fabricated, erected, and tested to quality standards commensurate with the ~~importance~~ safety significance of the ~~safety~~ functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the ~~required~~ safety significance of the function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety-significant functions. Appropriate records of the design, fabrication, erection, and testing of these structures, systems, and components ~~important to safety~~ shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

- PDC is applicable to safety-significant SSCs, defined in NEI 18-04, with the definition of safety-related supplemented by 10 CFR 50.2
- Rewording is necessary to logically tie functions to SSCs within scope, and allow clear delineation of SSC treatment within NEI 18-04

Special Treatment Category	Applicability[1] SR SSC	NSRST SSC	NST SSC	Available Guidance[4]
10 CFR 50 Appendix B Quality Assurance Program	√			QA requirements consistent with 10 CFR 50 Appendix B should be risk-informed and performance-based and not compliance-based; guidance in SRP 17.5 Quality Assurance for safety-related SSCs, 10 CFR 50.69, SRP 1.201
User provided Quality Assurance (QA) Program for non-safety SSCs		√		QA requirements consistent with SRP 17.4 (Reliability Assurance Program) for non-safety-related, safety significant SSCs should be risk-informed and performance-based and not compliance based; guidance in SRP 17.5 Quality Assurance for non-safety-related SSCs, 10 CFR 50.69, SRP 1.201

- Removed *required* and created *safety-significance/significant* to avoid confusion and unintentional exclusion of other SSCs
  - Required Safety Function is a term, defined in NEI 18-04 and 21-07, that pertains only to safety-related SSCs
  - PDC scope includes NSRST SSCs; these are "non-safety" SSCs
- Removed "important to safety"; use "these" to continue restrictive logic from the previous sentence and PDC scope.

Use of "Safety-Significant"

- Similar changes are applied to develop PDC 2-5

Proposed Natrium PDC 2	Proposed Natrium PDC 3	Proposed Natrium PDC 4	Proposed Natrium PDC 5
Structures, systems, and components ~~important to safety~~ that are safety-significant shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety-significant functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the safety significance ~~importance~~ of the ~~safety~~ functions to be performed.	Structures, systems, and components ~~important to safety~~ that are safety-significant shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and fire-resistant materials shall be used wherever practical throughout the unit, particularly in locations with these structures, systems, or components ~~important to safety~~. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on these structures, systems, and components ~~important to safety~~. Firefighting systems shall be designed to ensure that their rupture or inadvertent operation does not significantly impair the safety-significant functions ~~capability~~ of these structures, systems, and components.	Structures, systems, and components ~~important to safety~~ that are safety-significant shall be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences, and postulated accidents, including the effects of liquid sodium and its aerosols and oxidation products. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate…[]	Structures, systems, and components ~~important to safety~~ that are safety-significant shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety-significant functions, including, in the event of an accident in one unit, the ability to achieve ~~an orderly shutdown and cooldown~~ and maintain safe shutdown of the remaining units.

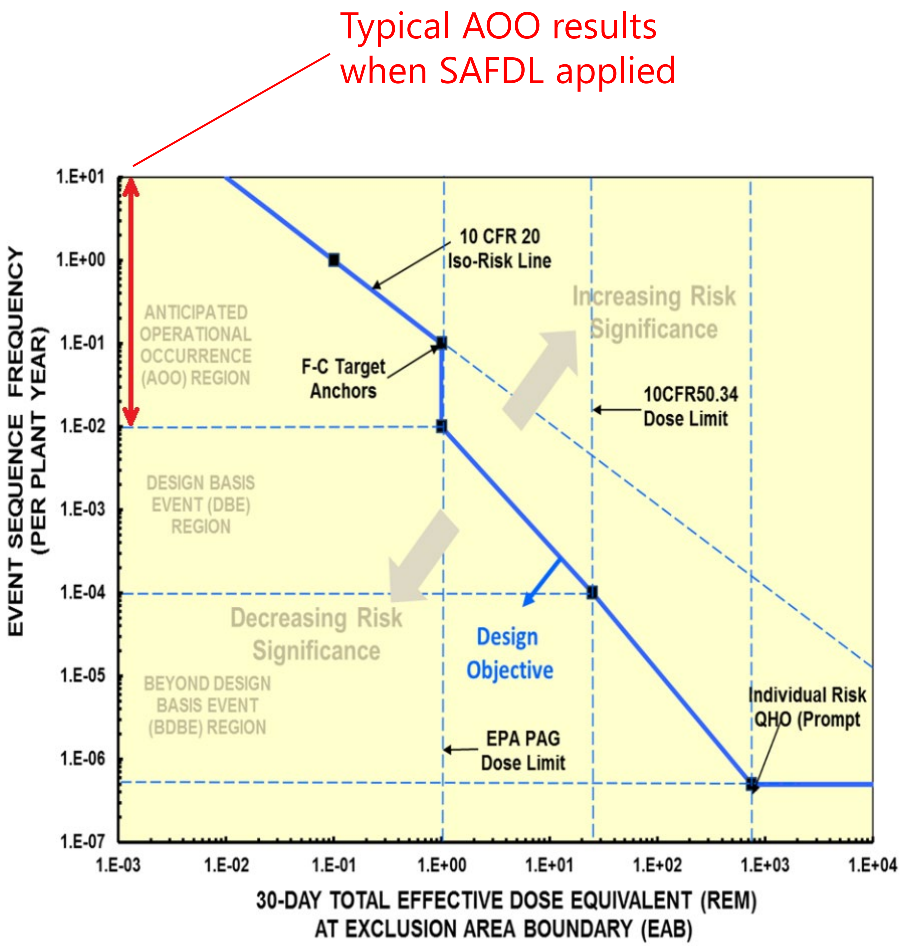
The Cohesive Relationship of SARRDL and FC Target

The Cohesive Relationship of SARRDL and FC Target

SAFDL: Specified Acceptable Fuel Design Limit

- Deterministic concept which is not conducive to LMP
  - Prior applicant made similar observation; *ML20167A174*
  - Would prevent practical usage of the AOO dose range

- Performance criteria for Normal Operations/AOOs only
  - RIPB licensing basis requires margin for <u>all</u> FC regions
  - Historically, TS and the ODCM limit operation with failed fuel
  - SAFDL goal is satisfied through RIPB licensing basis which has improved specificity for all LBEs.



Adapted from NEI 18-04

The Cohesive Relationship of SARRDL and FC Target

Technical rationale for acceptance criteria listed in NUREG-0800 Section 4.3, *Nuclear Design*, Rev 3:

- Compliance with **GDC 10** **significantly reduces the likelihood** of fuel failures occurring during normal operations, including anticipated operational occurrences, thereby **minimizing the possible release of fission products** to the environment.

- **GDC 20** requires automatic initiation of the reactivity control systems to assure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences…The automatic initiation of control systems during a reactor transient prevents damage to the nuclear fuel and, in the early stages of a reactor accident, will **minimize the extent of damage to the fuel, thus reducing the release of fission products** to the reactor coolant system and possibly the environment.

- Meeting the requirements of **GDC 25** **significantly reduces the possibility** that a malfunction in the RCS would result in nuclear fuel damage.

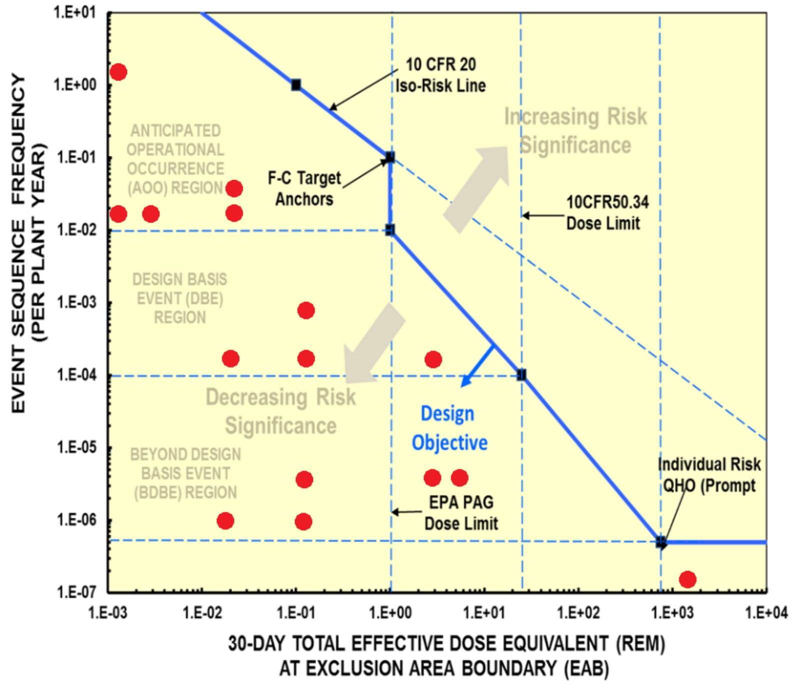
- Compliance with **GDC 26** provides assurance that core reactivity can be safely controlled…**minimizing the likelihood of fuel damage and the subsequent release of fission products.**

The Cohesive Relationship of SARRDL and FC Target

SARRDL: Specified Acceptable System Radionuclide Release Design Limit

- Originated prior to[1], and not mentioned within, NEI 18-04 but has essentially the same objective
  - Limit radionuclide inventory of systems to preserve margin dose assumptions resulting from LBEs
  - SARRDL is derived from margin to FC Target (*see example image to the right*); both are an improvement to SAFDL "goals" highlighted in previous slide
- SARRDL is envisioned as being controlled via TS
  - Ideal would be akin to setpoint control program; allowing adherence to the RIPB methodology
  - Reduces NRC and licensee burden involved with frequent revisions to a TS Safety Limit

For illustrative purposes only



Hypothetical event sequences with release from a specific system
For illustrative purposes only

Adapted from NEI 18-04

1. Developed as part of INL-EXT-14-31179-R1; *July 2013 – December 2014*

The Cohesive Relationship of SARRDL and FC Target

Proposed Natrium PDC 10	Proposed Natrium PDC 12	Proposed Natrium PDC 20
The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable system radionuclide release ~~fuel~~ design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.	The reactor core; associated structures; and associated coolant, control, and protection systems shall be designed to ensure that power oscillations that can result in conditions exceeding specified acceptable system radionuclide release ~~fuel~~ design limits are not possible or can be reliably and readily detected and suppressed.	The protection system shall be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable system radionuclide release ~~fuel~~ design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components that are safety-significant ~~important to safety~~.

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The Cohesive Relationship of SARRDL and FC Target

Proposed Natrium PDC 25	Proposed Natrium PDC 26
The protection system shall be designed to ensure that specified acceptable system radionuclide release ~~fuel~~ design limits are not exceeded during any anticipated operational occurrence accounting for a single malfunction of the reactivity control systems.	A minimum of two reactivity control systems or means shall provide: (1) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that ~~the design limits for the fission product barriers~~ specified acceptable system radionuclide release design limits are not exceeded and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences. (2) A means which is independent and diverse from the other(s), shall be capable of controlling the rate of reactivity changes resulting from planned, normal power changes to assure that ~~the design limits for the fission product barriers~~ specified acceptable system radionuclide release design limits are not exceeded. (3) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the capability to cool the core is maintained and a means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following a postulated accident. (4) A means for holding the reactor shutdown under conditions which allow for interventions such as fuel loading, inspection and repair shall be provided.

SR SSC Responses to AOOs

SR SSC Responses to AOOs

Addressing AOOs in PDC using LMP

"Anticipated operational occurrences mean those conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit and include, but are not limited to, loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power."

-10 CFR 50, Appendix A, *Definitions and Explanations*

This states that the subsequent events are AOOs.

These are not expected to be AOOs.

"The applicant has historically been expected to classify the off-normal events considered within the design basis as either AOO or DBA based on a list of historically considered events for light water reactors (LWRs) and with subjective assessment of the expected frequency of occurrence. For advanced non-LWRs, the prescriptive lists of generic LWR events are not applicable."

-From NEI 18-04, Section 3, *Selection of Licensing Basis Events*

LMP allows applicants to select their own LBEs.

SR SSC Responses to AOOs

SR SSCs are not the preferred means for AOO mitigation but are credited in the deterministic analysis required by NEI 18-04 for evaluating DID adequacy

Recognized by the SRP, and NRC-endorsed IEEE Stds, as the system required for initiating a reactor scram and actuating engineered safety features; both are SR SSCs per 10 CFR 50.2

Criterion	SFR-DC Title and Content
20	*Protection system functions.* Same as GDC The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Table 5-2. Guidelines for Establishing the Adequacy of Overall Plant Capability Defense-in-Depth

Layer[a]	Layer Guideline		Overall Guidelines	
	Quantitative	Qualitative	Quantitative	Qualitative
1) Prevent off-normal operation and AOOs	Maintain frequency of plant transients within designed cycles; meet owner requirements for plant reliability and availability[b]		Meet F-C Target for all LBEs and cumulative risk metric targets with sufficient[d] margins	No single design or operational feature,[c] no matter how robust, is exclusively relied upon to satisfy the five layers of defense
2) Control abnormal operation, detect failures, and prevent DBEs	Maintain frequency of all DBEs < 10^{-2}/plant-year	Minimize frequency of challenges to SR SSCs		
3) Control DBEs within the analyzed design basis conditions and prevent BDBEs	Maintain frequency of all BDBEs < 10^{-4}/plant-year	No single design or operational feature[c] relied upon to meet quantitative objective for all DBEs		
4) Control severe plant conditions and mitigate consequences of BDBEs	Maintain individual risks from all LBEs < QHOs with sufficient[d] margins	No single barrier[c] or plant feature relied upon to limit releases in achieving quantitative objectives for all BDBEs		
5) Deploy adequate offsite protective actions and prevent adverse impact on public health and safety				

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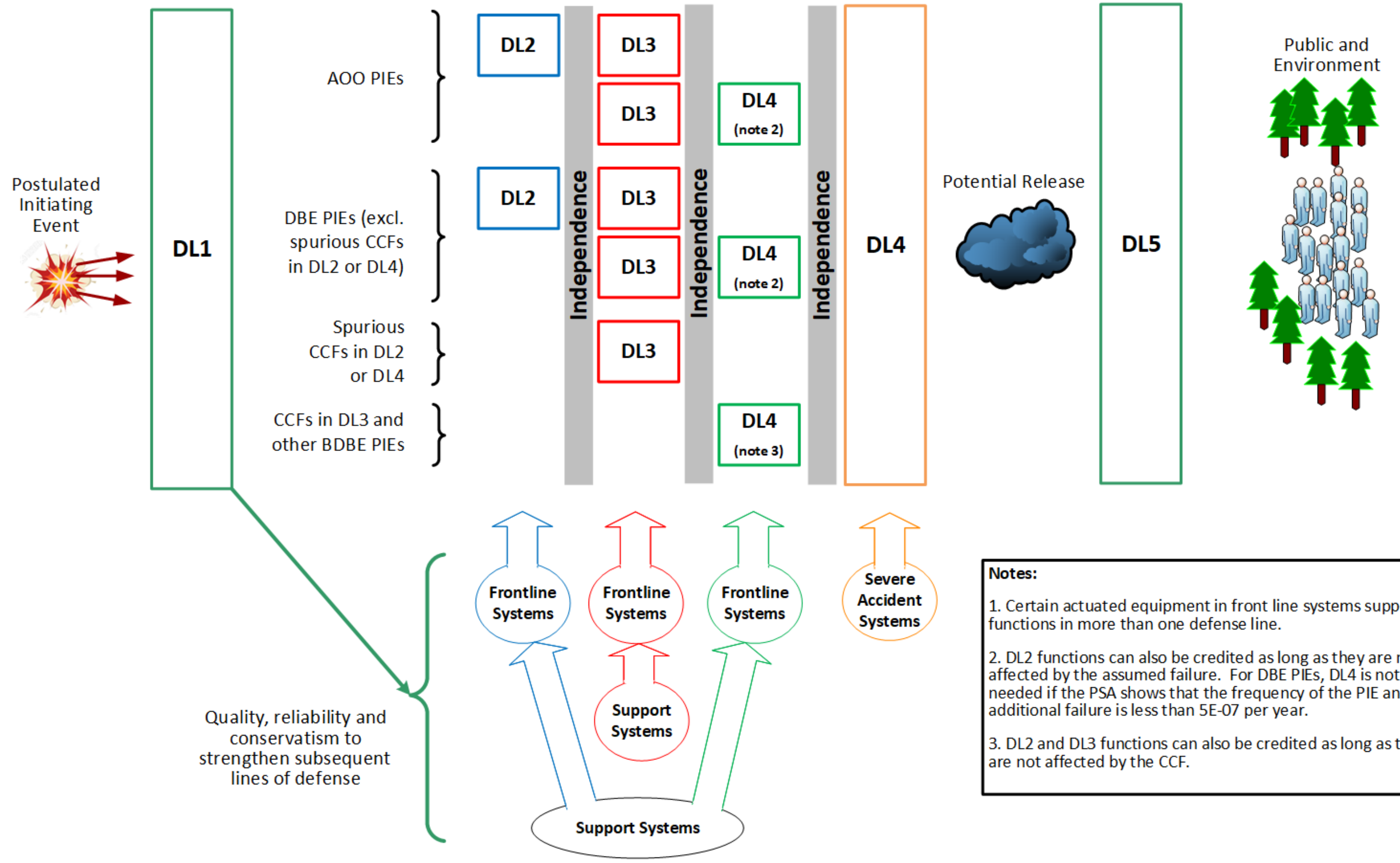
SR SSC Responses to AOOs



Notes:

1. Certain actuated equipment in front line systems supports functions in more than one defense line.

2. DL2 functions can also be credited as long as they are not affected by the assumed failure. For DBE PIEs, DL4 is not needed if the PSA shows that the frequency of the PIE and additional failure is less than 5E-07 per year.

3. DL2 and DL3 functions can also be credited as long as they are not affected by the CCF.

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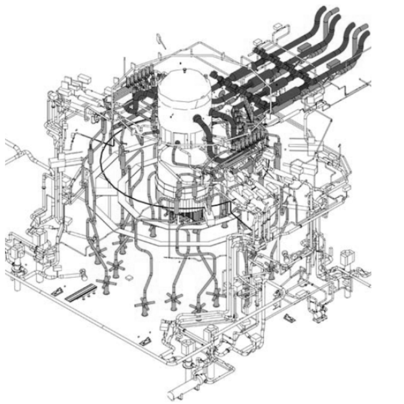
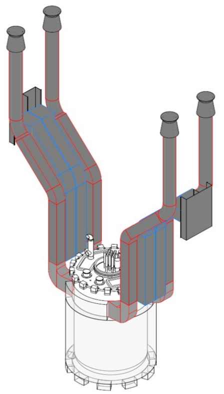
RAC Replaces Multiple LWR Systems

RAC Replaces Multiple LWR Systems

LWR ECCS

- 2600+ ASME Sect. III Pipe Welds
- High Pressure Injection (1000+ PSI)
- Large Water Inventory Requirements
- Active Valve and Pump Operation
- Multiple Trains and Sub-systems





Natrium RAC System

- Zero ASME Sect. III Pipe Welds
- Atmospheric Pressure (<1 PSI)
- Unlimited Air-Cooled Heat Sink Supply
- Fully Passive (Always in Operation)
- Singular Rugged System

	SFR-DC 33	SFR-DC 34	SFR-DC 35	SFR-DC 38	Natrium PDC
	Primary coolant make-up/inventory maintenance	RHR	ECCS	Containment Heat Removal	RAC
Phase	Normal operations with a small leak	Normal/AOO	During and following accident	Following accident	All
Goal	SAFDL	SAFDL/PCB DL	Fuel/clad damage that impairs core cooling	P/T design limit	SARRDL
Classification	SR	SR	SR	SR	SR
Suitable capabilities differences	None	Isolation	Isolation and containment	Isolation and containment	Isolation and containment

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RAC Replaces Multiple LWR Systems

Reactor Air Cooling System (RAC)

- Performs equivalent LWR safety functions of RHR, ECCS, and Containment Heat Removal
  - SFR-DC 34
  - SFR-DC 35
  - SFR-DC 38
- Reactor vessel design and operating features eliminate need for a reactor coolant make-up system
  - ~~SFR-DC 33~~

Proposed Natrium PDC ##

Reactor Air Cooling System

A system to assure sufficient core cooling ~~during postulated accidents and to remove residual heat following postulated accidents~~ shall be provided. The system required safety functions shall be to:

(1) ~~transfer~~ remove fission product decay heat, and other residual heat from the reactor core, at a rate such that specified acceptable ~~fuel~~ system radionuclide release design limits ~~and the design conditions of the primary coolant boundary~~ are not exceeded for normal operations and anticipated operational occurrences.

(2) remove heat from the reactor core during and following postulated accidents, such that fuel and clad damage that could interfere with continued effective core cooling is prevented.

(3) remove heat from the ~~reactor~~ functional containment ~~shall be provided~~, as necessary, to maintain the functional containment performance criteria ~~pressure and temperature~~ within acceptable limits during and following postulated accidents.

Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to ensure that the system required safety functions can be accomplished, assuming a single failure.

Functional Containment

Functional Containment

- Mechanistic source term is an inherent assumption of NEI 18-04
  - Impractical without functional containment

- Acceptable dose is the goal
  - Common objective of LMP and SECY 18-0096

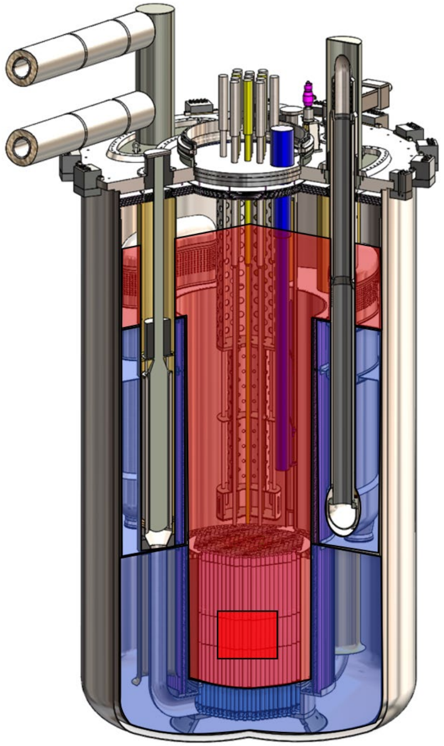
- Section V SFR-DC not that different from GDC
  - LWR design-basis, several large penetrations assumed, CIVs must be redundant and SR

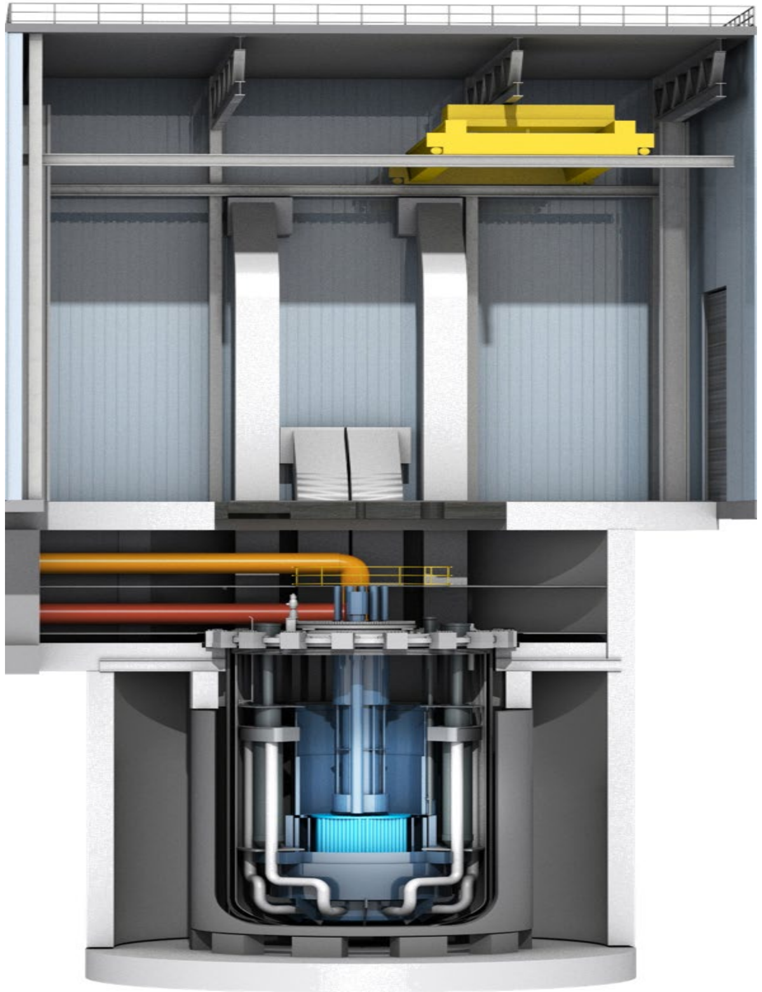
- SFR-DC 16 *Containment Design* is most conducive to the Natrium design
  - Guard vessel is pressure-retaining and encloses the reactor vessel and primary coolant loops
  - 10 CFR 50.34 postulated accident dose limit is common to LMP and SECY 18-0096

Functional Containment

Proposed Natrium PDC 16
A ~~reactor~~ functional containment, consisting of ~~a low-leakage, pressure retaining structure surrounding the reactor and its primary cooling system~~ multiple barriers internal and/or external to the reactor and its primary coolant boundary shall be provided to control the release of radioactivity to the environment. ~~and to ensure that the reactor containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.~~ ~~The containment leakage~~ Leakage through the aggregate SSCs making up the functional containment shall be restricted to be less than that needed to meet ~~the acceptable onsite and offsite dose consequence limits, as specified in~~ the functional containment performance criteria established for each event category, including 10 CFR 50.34 postulated accidents.

- Section V SFR-DC not required to support Natrium PDC 16
  - ~~SFR-DC 50-57~~





Additional PDC Considerations

Additional PDC Considerations

Intermediate Air Cooling – passive mode

- Expected to be NSRST
  - Anticipate developing a PDC for this function

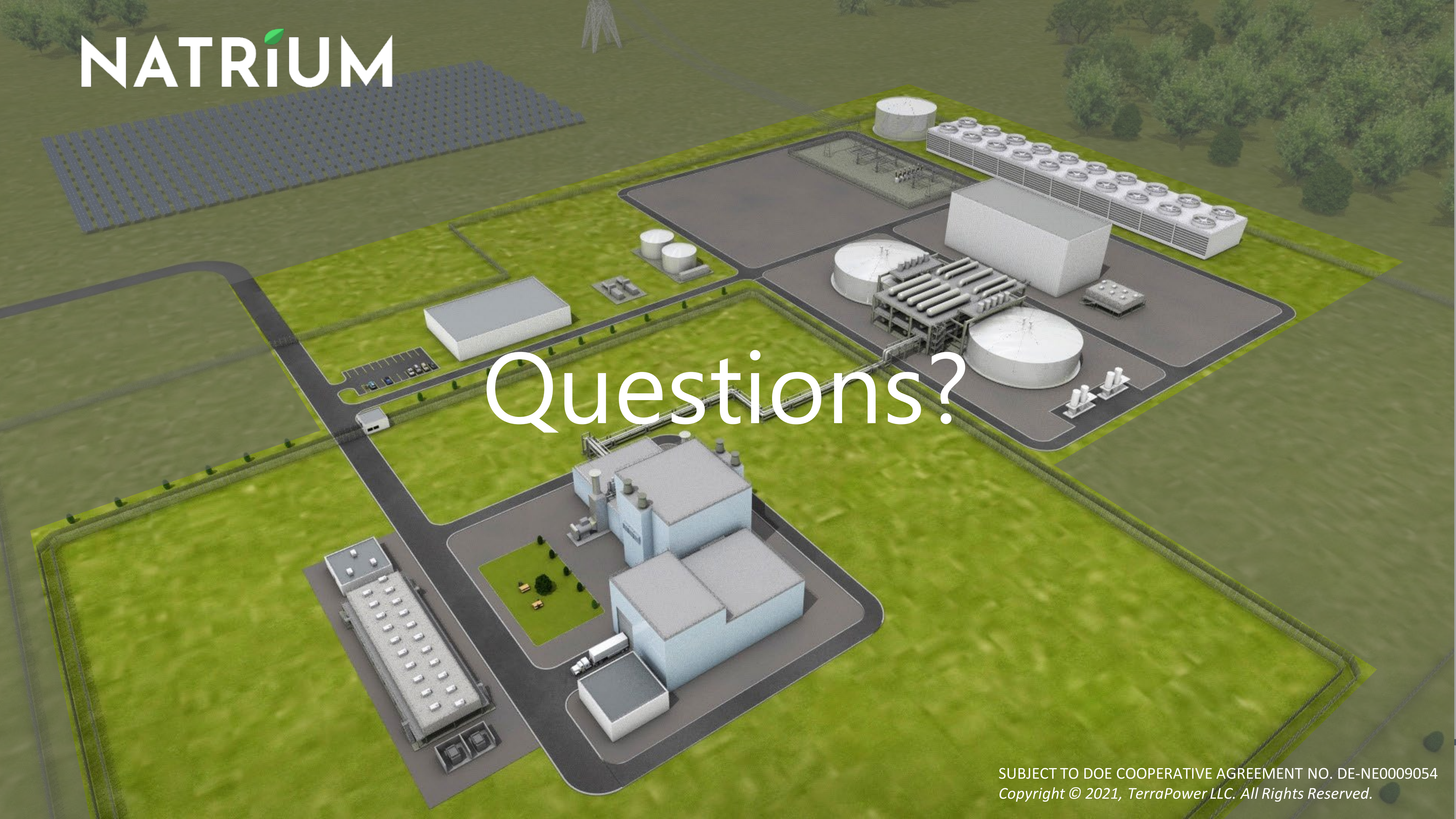
Electric Power Systems

- PDC development continues
  - May not replace "important to safety" with "safety-significant"

Intermediate Coolant System

- Assessing SFR-DC 70, 73-78

Electric power systems shall be provided when ~~required~~ needed to permit ~~functioning~~ required safety functions of structures, systems, and components. The required safety function for each power system shall be to provide sufficient capacity and capability to ensure that (1) that ~~the~~ specified acceptable system radionuclide release design limits ~~for the fission product barriers~~ are not exceeded as a result of anticipated operational occurrences and (2) required safety functions that rely on electric power are maintained in the event of postulated accidents.
The electric power systems shall include an onsite power system and an additional power system. The onsite electric power system shall have sufficient independence, redundancy, and testability to perform its required safety functions, assuming a single failure. An additional power system shall have sufficient independence and testability to perform its required safety function.
If electric power is not needed for anticipated operational occurrences or postulated accidents, the design shall demonstrate that power for ~~important to~~ safety-significant functions is provided.



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Questions?

SUBJECT TO DOE COOPERATIVE AGREEMENT NO. DE-NE0009054

Acronyms

- AOO – Anticipated Operational Occurrence
- CCF – Common Cause Failure
- DBE – Design Basis Event
- DBA – Design Basis Accident
- DID – Defense-in-Depth
- DL – Defense Layer
- ECCS – Emergency Core Cooling System
- FC – Frequency-Consequence
- GDC – General Design Criteria
- LBE – Licensing Basis Event
- LMP – Licensing Modernization Project
- LWR – Light Water Reactor
- ODCM – Offsite Dose Calculation Manual
- RCS – Reactor Core Cooling System
- PCB – Primary Coolant Boundary
- PDC – Principal Design Criteria
- PIE – Postulated Initiating Event
- RHR – Residual Heat Removal
- RIPB – Risk Informed, Performance Based
- SAFDL – Specified Acceptable Fuel Design Limit
- SARRDL – Specified Acceptable System Radionuclide Release Design Limit
- SFR – Sodium Fast Reactor
- SFR-DC – Sodium Fast Reactor Design Criteria
- SSC – Structures, Systems, and Components
- SR – Safety Related
- TS – Technical Specfications
- NSRST – Non-Safety Related, Special Treatment