January 24, 2023

TP-LIC-LET-0052
Project Number 99902100
10 CFR 50.34(a)(3)(i)

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

Subject: Submittal of TerraPower Topical Report, "Principal Design Criteria for the Natrium Advanced Reactor"

This letter transmits the TerraPower, LLC (TerraPower) Topical Report, NATD-LIC-RPRT-0002 Revision 0, Principal Design Criteria for the Natrium Advanced Reactor, to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. The report contains the result of TerraPower's process to develop Principal Design Criteria following guidance from Regulatory Guide 1.232 Revision 0, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors."

TerraPower requests the NRC's review and approval of these Principal Design Criteria for use by future applicants. TerraPower additionally requests review and approval of the rationale provided in this Topical Report for meeting the intent of Natrium Principal Design Criteria 26. This letter requests that a nominal review duration of one year be considered.

The enclosed topical report contains proprietary information. It is requested that Enclosure 2, which contains proprietary information, be withheld from public disclosure in accordance with 10 CFR 2.390(a)(4). An affidavit certifying the basis for the request to withhold Enclosure 2 from public disclosure is included as Enclosure 1. Proprietary materials have been redacted from the topical report provided in Enclosure 2 and is identified by using [[ ]]$^{(a)(4)}$. Enclosure 3 contains a non-proprietary public version of the report.

This letter and its enclosure 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

Enclosure:
1. TerraPower, LLC Affidavit and Request for Withholding from Public Disclosure (10 CFR 2.390(a)(4))
2. TerraPower, LLC Topical Report, NATD-LIC-RPRT-0002, Principal Design Criteria for the Natrium Advanced Reactor – Proprietary (Non-Public)
3. TerraPower, LLC Topical Report, NATD-LIC-RPRT-0002, Principal Design Criteria for the Natrium Advanced Reactor – Non-Proprietary (Public)

cc: Mallecia Sutton, NRC
William Jessup, 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 2, 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 2 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 2 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: January 24, 2023

George Wilson Jr

George Wilson
Vice President, Regulatory Affairs
TerraPower, LLC

ENCLOSURE 3

TerraPower, LLC
Topical Report, NATD-LIC-RPRT-0002
Principal Design Criteria for the Natrium Advanced Reactor
Non-Proprietary (Public)

Controlled Document - Verify Current Revision

TOPICAL REPORT

Document Number:	NATD-LIC-RPRT-0002		Revision:	0
Document Title:	Principal Design Criteria for the Natrium Advanced Reactor			
Functional Area:	Licensing	Engineering Discipline:	Safety & Licensing	
Effective Date:	01/24/2023	Released Date:	01/24/2023	
			Page:	1 of 47

Approval

Title	Name	Signature	Date
Originator, Licensing Manager	Michael Montecalvo	Electronically Signed in Agile	01/23/2023
Reviewer, Licensing Manager	Ian Gifford	Electronically Signed in Agile	01/20/2023
Approver, Director of Licensing	Ryan Sprengel	Electronically Signed in Agile	01/24/2023

Export Controlled Content: **Yes** ☐ **No** ☒

QA Related: **Yes** ☐ **No** ☒ (**Note:** If QA in Yes, a QA representative needs to be on review)

QA Criterion: N/A

REVISION HISTORY

Revision No.	Effective Date	Affected Section(s)	Description of Change(s)
0	01/24/2023	All	Initial issuance.

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EXECUTIVE SUMMARY

This topical report describes the development of Principal Design Criteria (PDC) for the Natrium™ advanced reactor[1]. PDC were developed using Regulatory Guides 1.232 (Reference 1) and 1.233 (Reference 2) based on design features specific to the Natrium advanced reactor. TerraPower, LLC is requesting United States (U.S.) Nuclear Regulatory Commission (NRC) review and approval of these PDC for use by future applicants.

[1] Natrium is a TerraPower and GE-Hitachi technology.

ACRONYMS

Table 1 - List of Acronyms

Acronym	Definition
AOO	Anticipated Operational Occurrence
ARDC	Advanced Reactor Design Criteria
ARDP	Advanced Reactor Demonstration Program
CFR	Code of Federal Regulations
DOE	Department of Energy
EAB	Exclusion Area Boundary
FC	Frequency and Consequence
GDC	General Design Criteria
LBE	Licensing Basis Event
LOCA	Loss-of-Coolant Accident
LPZ	Low Population Zone
LWR	Light Water Reactor
MHTGR	Modular High Temperature Gas-Cooled Reactor
MHTGR-DC	Modular High Temperature Gas-Cooled Reactor Design Criteria
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PDC	Principal Design Criteria
PRA	Probabilistic Risk Assessment
RG	Regulatory Guide
RIPB	Risk-Informed, Performance-Based
SARRDL	Specified Acceptable System Radionuclide Release Design Limit
SFR	Sodium-Cooled Fast Reactor
SFR-DC	Sodium-Cooled Fast Reactor Design Criteria
SRM	Staff Requirements Memoranda
SSC	Structure, System, and Component
U.S.	United States

1 NATRIUM ADVANCED REACTOR DESIGN FEATURES

The Natrium advanced reactor is a metal fueled, pool-type sodium-cooled fast reactor (SFR) that takes advantage of a simple and robust safety profile to reduce the complexities associated with nuclear design and construction. Safety functions are made integral to the reactor vessel and support equipment is moved to separate structures, resulting in a simplified reactor building. The heat transfer characteristics of sodium and operation at low pressures permit the use of compact and lightweight equipment, unlike in other reactor types cooled with pressurized water or gas.

The Natrium advanced reactor is coupled to an integrated energy system, a molten salt loop that transports heat from the nuclear island to the energy island. This minimizes the amount of equipment and structures on the nuclear site and enables load following, coupling to thermal energy storage, and industrial heat applications. For heat transport, energy storage, and electricity generation, the Natrium advanced reactor uses molten salt technology which reduces cost, enables thermal energy storage, and eliminates the difficulties of managing the sodium-to-steam heat transfer interface.

The higher operating temperature and constant thermal output make an ideal match for thermal energy storage using molten salt, a technology that is commercially deployed in the solar energy industry. Hot sodium from the reactor cycles through the primary heat transport system and transfers its heat to an intermediate sodium loop. The sodium in the intermediate heat transport system passes through sodium-salt heat exchangers to heat the molten salt loop, which carries heat off the nuclear island to the energy island where it can be stored, converted into electricity, or used for industrial process heating. This decoupled architecture minimizes the size of the nuclear island and allows the reactor to operate at constant power, while the energy island meets variable energy demands. The rating for the Natrium advanced reactor will be 840 MW thermal. The energy island will have the capability to produce 338 MW electric at steady state conditions and up to 500 MW electric for limited durations to accommodate grid demand.

2 INTRODUCTION

In October 2020, the Department of Energy (DOE) selected the Natrium advanced reactor design for the Advanced Reactor Demonstration Program (ARDP). TerraPower, LLC, in partnership with GE-Hitachi Nuclear Energy Americas, LLC, is using the regulatory framework outlined in Title 10 of the *Code of Federal Regulations* Part 50 (10 CFR 50) to design, license, construct, and operate the Natrium advanced reactor. TerraPower is implementing the technology-inclusive, risk-informed, and performance-based methodology outlined in Regulatory Guide (RG) 1.233 (Reference 2).

10 CFR 50.34(a)(3)(i) requires that applicants for a construction permit include the Principal Design Criteria (PDC) for a facility.

10 CFR 50, Appendix A, provides General Design Criteria (GDC) that establish the minimum requirements for PDC for light water reactors (LWRs). The GDC are generally applicable to other types of reactor technologies and provide guidance to establish the PDC for other non-LWR reactor designs.

The Natrium PDC were developed based on the Sodium-Cooled Fast Reactor Design Criteria (SFR-DC) in RG 1.232 as informed by design features specific to the Natrium advanced reactor. Section 4 of this report provides the process by which TerraPower developed PDC for the Natrium advanced reactor.

TerraPower requests NRC review and approval of these PDC to be used by future applicants. This topical report will serve as a means, via reference, for using Natrium PDC in the development of license application content for Natrium advanced reactors.

3 REGULATORY EVALUATION

Facilities licensed under 10 CFR 50 are required to describe the PDC in their preliminary safety analysis report supporting a construction permit application as described in 10 CFR 50.34(a)(3)(i).

10 CFR 50, Appendix A, contains the GDC that establish the minimum requirements for the development of PDC for LWRs. GDC are generally applicable to non-LWRs and are intended to provide guidance in establishing the PDC for non-LWRs.

RG 1.232 provides a set of advanced reactor design criteria (ARDC) which serve the same purpose for non-LWRs as the GDC serve for LWRs. RG 1.232 also provides two sets of technology-specific, non-LWR design criteria for SFRs and modular high temperature gas-cooled reactors (MHTGRs).

RG 1.233 provides guidance on using a technology-inclusive, risk-informed, and performance-based methodology to inform the licensing basis and content of applications for non-LWRs.

4 PDC DEVELOPMENT METHODOLOGY

Developing the Natrium PDC began with a review of NRC guidance. It was identified that RG 1.232 contained SFR-DC which were adapted from the GDC described in 10 CFR 50, Appendix A. The following statements were noted in RG 1.232:

> *"Applicants may use this RG to develop all or part of the PDC and are free to choose among the [Advanced Reactor Design Criterion] ARDC, SFR-DC, or [Modular High Temperature Gas-Cooled Reactor Design Criteria] MHTGR-DC to develop each PDC after considering the underlying safety basis for the criterion and evaluating the rationale for the adaptation described in this RG."*

> *"The non-LWR design criteria developed by the NRC staff and included in Appendices A to C of this regulatory guide are intended to provide stakeholders with insight into the staff's views on how the GDC could be interpreted to address non-LWR design features; however, these are not considered to be final or binding regarding what may eventually be required from a non-LWR applicant. It is the applicant's responsibility to develop the PDC for its facility based on the specifics of its unique design, using the GDC, non-LWR design criteria, or other design criteria as the foundation."*

> *"The NRC recognizes the future benefits to risk informing the non-LWR design criteria to the extent possible, depending on the design information and data available."*

From these three statements, a strategy for developing Natrium PDC was created.

The SFR-DC from Appendix B of RG 1.232 were reviewed for applicability against the Natrium advanced reactor design and adopted as written when possible. When a feature of the Natrium design was not well represented by the SFR-DC (e.g., use of a functional containment), the language from other RG 1.232 appendices were reviewed to determine if the criterion could be modified, or if a new PDC would be required. Figure 1 illustrates this process.

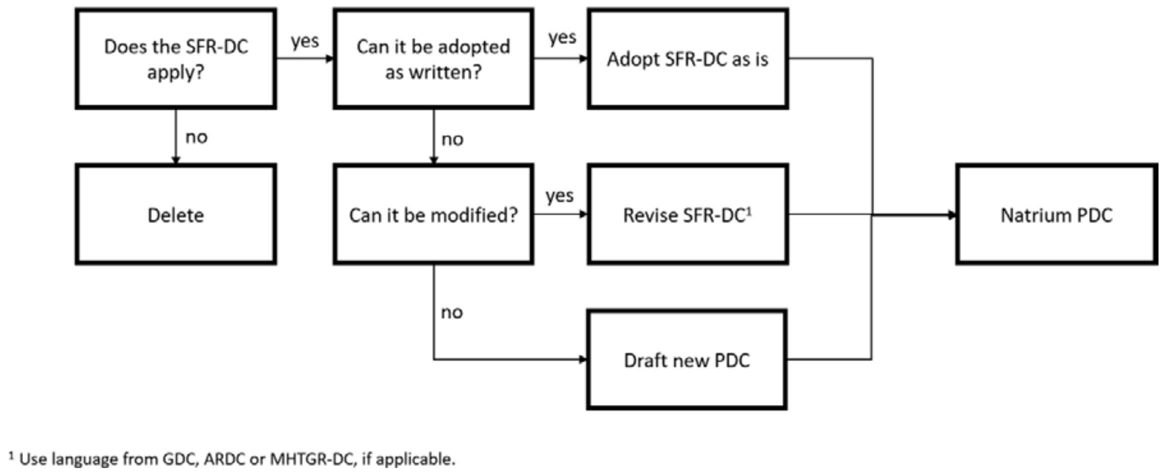

Figure 1 PDC Development Flowchart

5 SUMMARY OF CHANGES TO THE RG 1.232 DESIGN CRITERIA

A review was performed to ensure that the PDC collectively provide a comprehensive design and regulatory framework for the Natrium advanced reactor. Sections 5.1 through 5.4 provide justifications for the types of changes made to the RG 1.232 design criteria.

5.1 Use of Safety-Significant and Addition of Safety-Significant Language

Nuclear Energy Institute (NEI) Technical Report 18-04 (Reference 3) describes that a safety-significant SSC is a key contributor to a plant's overall robustness, reliability, and capability to maintain Licensing Basis Event (LBE) Frequency and Consequence (FC) within prescribed targets. To align with the approach outlined in NEI 18-04 as endorsed by the NRC in RG 1.233, "important to safety" from the GDC and design criteria in RG 1.232 is replaced with "safety-significant" in the Natrium PDC. This term is more reflective of a risk-informed, performance-based (RIPB) licensing basis and has previously been approved by the NRC (Reference 4) for use in non-LWR PDC.

5.2 Use of Specified Acceptable System Radionuclide Release Design Limit

The concept of a Specified Acceptable System Radionuclide Release Design Limit (SARRDL) was introduced out of necessity for other advanced reactor types (Reference 5). However, endorsement of a technology-inclusive RIPB licensing basis methodology (References 2 and 3) makes the SARRDL concept accessible for all non-LWR developers.

As a performance criterion, adhering to a SARRDL assures that regulatory limits for offsite doses will not be exceeded for any LBE. Compliance with a SARRDL is achieved through monitoring and controlling radionuclide inventory during normal operation so that the assumed radionuclide inventory at the onset of an LBE is not exceeded.

Similarly, the methodology endorsed by RG 1.233 provides reasonable assurance that regulatory limits for offsite dose are not exceeded as a result of LBEs. Within this methodology is an expectation for having margin to regulatory limits and assessing the magnitude of that margin when evaluating defense-in-depth.

> *"The NRC's Advanced Reactor Policy Statement includes expectations that advanced reactors should provide enhanced margins of safety. The safety margin between the design-specific PRA results and the F-C Target provides one useful and practical demonstration of how the design fulfills the NRC's expectations for enhanced safety."*

> *"A[n] example of safety margins is the evaluation of margins between the frequencies and consequences of the LBEs and the F-C Target and the margins between the cumulative risk metrics and the cumulative risk targets used for LBE evaluation. These risk margins are assessed as part of the RIPB evaluation of DID."*

SARRDL will be established from the most-limiting event sequence inclusive of a maximum assumed radionuclide inventory (i.e., the initial radionuclide inventory at the onset of an LBE). The radionuclide inventory will be monitored during normal operation to ensure that SARRDLs are not exceeded for any LBE.

5.3 RG 1.232 Design Criteria Not Applicable

Several SFR-DC were determined to be not applicable during the review. These SFR-DC fell into two groups – SFR-DC not applicable due to the use of a functional containment concept in the Natrium advanced reactor design and SFR-DC incorporated into other PDC (SFR-DC 27).

5.3.1 Functional Containment

The containment-related SFR-DC do not consider a functional containment concept; however, RG 1.232 provides a discussion on the use of a functional containment strategy in Appendix C. The functional containment concept is RIPB, and complements the mechanistic source term, use of SARRDL, and overall Licensing Modernization Project methodology endorsed in RG 1.233. Additionally, the Commission has found the functional containment concept generally acceptable as indicated in staff requirements memoranda (SRM) to SECY-93-092 (Reference 8) and SECY-03-0047 (Reference 9).

Design requirements for a traditional pressure-retaining containment are primarily based on the presumption of a large Loss-of-Coolant Accident (LOCA). The Natrium advanced reactor is a pool-type design which has a primary sodium coolant loop that is internal to the reactor. Unlike an LWR, the normal operating pressure will be near atmospheric, and the primary sodium coolant will operate well below its boiling point. The reactor enclosure system will utilize a head and double vessel configuration (i.e., reactor and guard vessel) that satisfies ASME Section III, Division 5, Class A (Reference 6) requirements for safety-related pressure boundary components. Concurrent leakage of both vessels is not a postulated LBE, and there are no penetrations through either the reactor vessel or the guard vessel. The primary sodium coolant will be blanketed by an inert cover gas that provides a separation distance to the reactor head. Postulated LBEs that assume leakage from reactor head penetrations are mitigated by cover gas makeup capability, isolation, and other inherent design features. For the specific case of postulated intermediate sodium coolant leaks, differences in operating and static fluid pressures preclude a loss of primary sodium coolant. Due to this configuration, a LOCA is not a postulated LBE for the Natrium advanced reactor.

The Natrium advanced reactor will contain necessary design features to prevent, or mitigate, postulated intermediate sodium coolant leaks and the consequential effects of a sodium-air reaction, including any postulated sodium spray fires. These scenarios were noted in the NRC staff rationale for SFR-DC 16 and will be considered to ensure that the functional containment performs, as necessary, to maintain offsite dose within limits for all LBEs.

Functional containment uses diverse barriers to prevent the release of radioactivity, providing additional defense-in-depth, and is more conducive to an RIPB licensing basis than a traditional containment. Functional containment has previously been approved by the NRC (Reference 4) for use in non-LWR PDC. Figure 2 illustrates an example of the overall barriers used in the Natrium advanced reactor functional containment strategy.

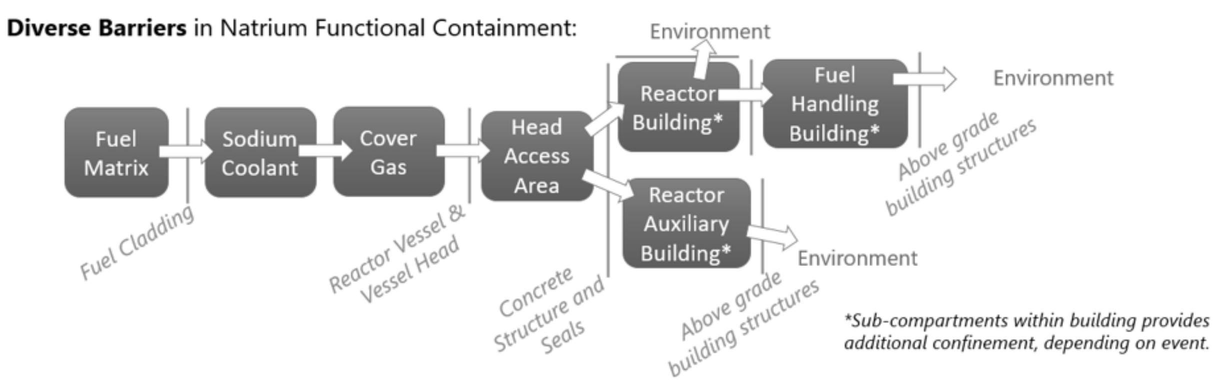

Figure 2 Natrium Functional Containment Barriers

5.3.2 Independent and Diverse

Natrium PDC 26 requires a minimum of two systems or means of reactivity control which are independent and diverse.

NRC staff rationale for adaptation of ARDC 26 in RG 1.232 states that the term “independent and diverse” indicates no shared systems or components and a design which is different enough such that no common failure modes exist between the system or means credited in ARDC 26 (2) and safety-related systems credited in ARDC 26 (1) and (3).

As discussed during a public meeting held November 10, 2022 (Reference 10), [[

]][(a)(4)] While this approach differs from the NRC staff rationale for adaptation in RG 1.232, it meets the intent of the GDC and ARDC requirements for reactivity control.

5.4 Natrium-Specific Language

Additional changes to PDC language are covered in detail under the “Basis for Change” column in Table 2.

One common set of changes is the removal of language in the MHTGR-DC which were found to be generally applicable to the Natrium design and adopted, but where a specific element is not present. For example, MHTGR-DC 71 and 72 were found to be applicable, but a postulated release of helium from the reactor building is not applicable for the Natrium design.

Another common example is incorporation of "safe shutdown" language where appropriate, based on the need to maintain a reactor coolant melting temperature which is significantly above ambient temperatures, and following the guidance available in SECY-94-084 (Reference 7).

6 NATRIUM PDC

Table 2 - Natrium Principal Design Criteria

	Natrium PDC	Source Language	Basis for Change
I.	Overall Requirements		
1	*Quality standards and records.* Safety-significant structures, systems, and components ~~important to safety~~ shall be designed, fabricated, erected, and tested to quality standards commensurate with the ~~importance of the~~ safety significance of the 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 safety-significant 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.	SFR-DC 1	The SFR-DC is modified to replace the term “important to safety” with “safety significant.” The NRC staff defines “important to safety” generally in 10 CFR 50, Appendix A, as, “structures, systems, and components [SSCs] that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.” RG 1.232 provides guidance on development of PDC in lieu of Appendix A GDC. To clarify the scope of SSCs that perform risk-significant functions (i.e., safety-related or non-safety-related with special treatment), including those that perform functions that are necessary to meet defense-in-depth criteria, NEI 18-04 (Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development) explicitly defines the term “safety significant.” The use of this term is endorsed in RG 1.233, and its use herein is judged to describe more consistently the SSCs that

	Natrium PDC	Source Language	Basis for Change
			provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. Accordingly, "safety significant" is used in defining Natrium PDC in lieu of "important to safety" as used in RG 1.232.
2	*Design bases for protection against natural phenomena.* Safety-significant structures, systems, and components ~~important to safety~~ 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 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 ~~importance of the safety~~ safety significance of the functions to be performed.	SFR-DC 2	See basis for safety-significant change in Natrium PDC 1.

	Natrium PDC	Source Language	Basis for Change
3	*Fire protection.* Safety-significant structures, systems, and components ~~important to safety~~ 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 safety-significant 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 safety-significant 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 capability of these structures, systems, and components.	SFR-DC 3	See basis for safety-significant change in Natrium PDC 1.

	Natrium PDC	Source Language	Basis for Change
4	*Environmental and dynamic effects design bases.* Safety-significant structures, systems, and components ~~important to safety~~ 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 that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. Chemical consequences of accidents, such as sodium leakage, shall be appropriately considered for the design of safety-significant structures, systems, and components ~~important to safety~~, which must be protected.	SFR-DC 4	See basis for safety-significant change in Natrium PDC 1.

	Natrium PDC	Source Language	Basis for Change
5	*Sharing of structures, systems, and components.* Safety-significant structures, systems, and components ~~important to safety~~ 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 functions, including, in the event of an accident in one unit, the ability to achieve and maintain safe ~~an orderly~~ shutdown ~~and cooldown~~ of the remaining units.	SFR-DC 5	See basis for safety-significant change Natrium PDC 1. Additional modification to the SFR-DC is made to replace the phrase "an orderly shutdown and cooldown" with "the ability to achieve and maintain safe shutdown". This modification is made to remove the implied temperature of "cooldown." The Natrium advanced reactor is a high-temperature system, with the reactor coolant melting temperature significantly above ambient temperatures. RG 1.232 ARDC 26 rationale and SECY-94-084 (Reference 7) describe a "safe shutdown" condition as: reactor subcriticality, decay heat removal, and radioactive materials containment. This modification is made to adopt this terminology.
II.	Multiple Barriers		

	Natrium PDC	Source Language	Basis for Change
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 ~~fuel~~ design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.	SFR-DC 10	SARRDLs were developed for MHTGRs to address the need for a performance limit that reflects the performance and characteristics of coated particle fuel. In the analysis of potential offsite dose consequences for certain MHTGR postulated accidents, two limits were identified that significantly influenced the offsite dose: (1) the radionuclide concentration in circulating coolant and (2) radionuclides condensed on surfaces of the reactor helium pressure boundary. This is a result of the high pressure of the MHTGR coolant which drives much or all of this activity out of the reactor coolant (helium) pressure boundary and the reactor building during postulated rapid depressurization events. SARRDLs for the Natrium advanced reactor will fulfill the same intent described in RG 1.232 and will be established so that (1) the consequences of the most limiting design basis accident does not exceed the siting regulatory dose limits criteria at the exclusion area boundary (EAB) and low-population zone (LPZ), and (2) the 10 CFR 20.1301 annualized dose limits to the public are not exceeded at the EAB for normal operation and AOOs. The normal operation radioactivity limit establishes an

	Natrium PDC	Source Language	Basis for Change
			initial condition which provides assurance that the dose limits described above are met. The other important difference in establishing SARRDLs for the Natrium advanced reactor originates from the low-pressure, single phase reactor coolant operation and the ability of the reactor coolant to retain selected fission products released from failed fuel. The consequence of reactor coolant boundary leakage will depend on the location of the leakage rather than the entire connected set of equipment that may contain coolant. This may also result in establishment of different limits on circulating activity in the coolant and circulating activity in the cover gas.
11	*Reactor inherent protection.* The reactor core and associated systems that contribute to reactivity feedback shall be designed so that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.	SFR-DC 11	No change.

	Natrium PDC	Source Language	Basis for Change
12	*Suppression of reactor power oscillations.* 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.	SFR-DC 12	See basis for SARRDL change in Natrium PDC 10.
13	*Instrumentation and control.* Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions, as appropriate, to ensure adequate safety, including those variables and systems that can affect the fission process and the integrity of the reactor core, ~~reactor helium pressure~~ safety-significant elements of the primary coolant boundary, and functional containment. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.	MHTGR-DC 13	MHTGR-DC base language was adopted to reflect the use of functional containment in the Natrium design. See basis for functional containment in Natrium PDC 16.
14	*Primary coolant boundary.* The safety-significant elements of the primary coolant boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.	SFR-DC 14	The addition of safety-significant language is based on the consequence of leakage as related to the location of leaks or breaks within the systems and components containing reactor coolant system. Certain components of the reactor coolant boundary have a greater safety significance than other components, and the quality standards and design requirements should reflect this. These

	Natrium PDC	Source Language	Basis for Change
			safety significant components of the reactor coolant boundary will be subject to leakage monitoring.
15	*Primary coolant system design.* The primary coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the safety-significant elements of the primary coolant boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.	SFR-DC 15	See basis for safety-significant addition in Natrium PDC 14.
16	*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 safety-significant design conditions ~~important to safety~~ are not exceeded for as long as postulated accident conditions require.	MHTGR-DC 16	The MHTGR-DC is adopted because the Natrium design relies on a functional containment approach instead of the typical LWR pressure-retaining containment structure. The Natrium functional containment safety design objective is to meet 10 CFR 50.34 offsite dose requirements at the plant's EAB with margin. A functional containment is defined in RG 1.232 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." The functional containment controls radionuclides at their source without requiring active design features or operator actions to ensure that the

	Natrium PDC	Source Language	Basis for Change
			dose at the site boundary as a consequence of postulated accidents meets regulatory limits. Additionally, the sodium coolant serves as an additional distinct barrier providing retention of fission products that escape the fuel. In accordance with the RG 1.232 rationale for functional containment, GDC 38, 39, 40, 41, 42, 43, 50, 51, 52, 53, 54, 55, 56, and 57 are not applicable to a design using functional containment since they address design criteria for pressure-retaining containments in the traditional LWR sense. Requirements for the performance of the reactor building are addressed by new Criterion 81 (design basis) and Criterion 82 (provisions for periodic testing and inspection). See basis for safety-significant change in Natrium PDC 1.

	Natrium PDC	Source Language	Basis for Change
17	*Electric power systems.* Electric power systems shall be provided when required to permit functioning of structures, systems, and components. The 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) 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 safety functions, assuming a single failure. An additional power system shall have sufficient independence and testability to perform its 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.	SFR-DC 17	See basis for SARRDL change in Natrium PDC 10. See basis for safety-significant change in Natrium PDC 1.

	Natrium PDC	Source Language	Basis for Change
18	*Inspection and testing of electric power systems.* Safety-significant electric power systems ~~important to safety~~ shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among systems.	SFR-DC 18	See basis for safety-significant change in Natrium PDC 1.

	Natrium PDC	Source Language	Basis for Change
19	*Control room.* A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent, as defined in § 50.2 for the duration of the accident. Adequate habitability measures shall be provided to permit access and occupancy of the control room during normal operations and under accident conditions. Adequate protection against sodium aerosols shall be provided to permit access and occupancy of the control room under accident conditions. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt ~~hot~~ shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during ~~hot~~ shutdown, and (2) with a potential capability for subsequent ~~cold~~ safe shutdown of the reactor through the use of suitable procedures.	SFR-DC 19	See basis for change in Natrium PDC 5.

	Natrium PDC	Source Language	Basis for Change
III.	Reactivity Control		
20	*Protection system functions.* The protection system shall be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that the specified acceptable system radionuclide release design limits is not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of safety-significant systems and components ~~important to safety~~.	MHTGR-DC 20	See basis for safety-significant change in Natrium PDC 1. MHTGR-DC adopted for its use of SARRDL language. See basis in Natrium PDC 10.
21	*Protection system reliability and testability.* The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.	SFR-DC 21	No change.

	Natrium PDC	Source Language	Basis for Change
22	*Protection system independence.* The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.	SFR-DC 22	No change.
23	*Protection system failure modes.* The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis, if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, sodium and sodium reaction products, pressure, steam, water, and radiation) are experienced.	SFR-DC 23	No change.
24	*Separation of protection and control systems.* The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.	SFR-DC 24	No change.

	Natrium PDC	Source Language	Basis for Change
25	*Protection system requirements for reactivity control malfunctions.* The protection system shall be designed to ensure that specified acceptable system radionuclide release design limits are not exceeded during any anticipated operational occurrence, accounting for a single malfunction of the reactivity control systems.	MHTGR-DC 25	MHTGR-DC adopted for its use of SARRDL language. See basis in Natrium PDC 10.

	Natrium PDC	Source Language	Basis for Change
26	*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 specified acceptable system radionuclide release design limits ~~for the fission product barriers~~ 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 specified acceptable system radionuclide release design limits ~~for the fission product barriers~~ 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.	SFR-DC 26	See basis for SARRDL change in Natrium PDC 10.
27	Deleted.	SFR-DC 27	Information incorporated into SFR-DC 26 per RG 1.232.

	Natrium PDC	Source Language	Basis for Change
28	*Reactivity limits.* The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (1) result in damage to the safety-significant elements of the primary coolant boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor vessel internals to impair significantly the capability to cool the core.	SFR-DC 28	See basis for safety-significant addition in Natrium PDC 14.
29	*Protection against anticipated operational occurrences.* The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.	SFR-DC 29	No change.

	Natrium PDC	Source Language	Basis for Change
IV.	Fluid and Heat Transport Systems		
30	*Quality of primary coolant boundary.* Components that are part of the primary coolant boundary shall be designed, fabricated, erected, and tested to ~~the highest quality standards practical~~ quality standards commensurate with the safety significance of the functions to be performed. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of primary coolant leakage.	SFR-DC 30	Language revised to conform with Natrium PDC 1. See basis for safety-significant change in Natrium PDC 1.
31	*Fracture prevention of primary coolant boundary.* Safety-significant components that are part of the primary coolant boundary shall be designed with sufficient margin to ensure that, when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures, service degradation of material properties, creep, fatigue, stress rupture, and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation and coolant composition, including contaminants and reaction products, on material properties, (3) residual, steady–state, and transient stresses, and (4) size of flaws.	SFR-DC 31	See basis for safety-significant change in Natrium PDC 1. Language is also consistent with Natrium PDC 14.

	Natrium PDC	Source Language	Basis for Change
32	*Inspection of primary coolant boundary.* Safety-significant components that are part of the primary coolant boundary shall be designed to permit (1) periodic inspection and functional testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor vessel.	SFR-DC 32	See basis for safety-significant change in Natrium PDC 1. Language is also consistent with Natrium PDC 14.
33	*Primary coolant inventory maintenance.* A system to maintain primary coolant inventory for protection against small breaks in the primary coolant boundary shall be provided as necessary to ensure that specified acceptable system radionuclide release ~~fuel~~ design limits are not exceeded as a result of primary coolant inventory loss due to leakage from the primary coolant boundary and rupture of small piping or other small components that are part of the boundary. The system shall be designed to ensure that the system safety function can be accomplished using the piping, pumps, and valves used to maintain primary coolant inventory during normal reactor operation.	SFR-DC 33	See basis for SARRDL change in Natrium PDC 10.

	Natrium PDC	Source Language	Basis for Change
34	*Residual heat removal.* A system to remove residual heat shall be provided. For normal operations and anticipated operational occurrences, the system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable system radionuclide release ~~fuel~~ design limits and the design conditions of the primary coolant boundary are not exceeded. Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities, shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.	SFR-DC 34	See basis for safety-significant change in Natrium PDC 10.
35	*Emergency core cooling.* A system to assure sufficient core cooling during postulated accidents and to remove residual heat following postulated accidents shall be provided. The system safety function shall be to transfer 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. Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.	SFR-DC 35	No change.

	Natrium PDC	Source Language	Basis for Change
36	*Inspection of emergency core cooling system.* A system that provides emergency core cooling shall be designed to permit appropriate periodic inspection of important components to ensure the integrity and capability of the system.	SFR-DC 36	No change.
37	*Testing of emergency core cooling system.* A system that provides emergency core cooling shall be designed to permit appropriate periodic functional testing to ensure (1) the structural ~~and leaktight~~ integrity of its components, (2) the operability and performance of the system components, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of any associated systems and interfaces necessary to transfer decay heat to the ultimate heat sink.	SFR-DC 37	Deleted "leaktight" in accordance with the NRC basis in RG 1.232, *"A non-leaktight system may be acceptable for some designs provided that (1) the system leakage does not impact safety functions under all conditions, and (2) defense in depth is not impacted by system leakage."*
38	Deleted.		The Natrium design does not include a pressure containing reactor containment structure. See the functional containment basis in Natrium PDC 16.
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43	Deleted.		The Natrium design does not include a pressure containing reactor containment structure. See the functional containment basis in Natrium PDC 16.
44	*Structural and equipment cooling.* A system to transfer heat from safety-significant structures, systems, and components ~~important to safety~~ to an ultimate heat sink shall be provided, as necessary, to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions. Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.	SFR-DC 44	See basis for safety-significant change in Natrium PDC 1.

	Natrium PDC	Source Language	Basis for Change
45	*Inspection of structural and equipment cooling systems.* The structural and equipment cooling systems shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to ensure the integrity and capability of the systems.	SFR-DC 45	No change.
46	*Testing of structural and equipment cooling systems.* The structural and equipment cooling systems shall be designed to permit appropriate periodic functional testing to ensure (1) the structural ~~and leaktight~~ integrity of their components, (2) the operability and performance of the system components, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequences that bring the systems into operation for reactor shutdown and postulated accidents, including the operation of associated systems.	SFR-DC 46	Deleted "leaktight" in accordance with the NRC basis in RG 1.232, *"A non-leaktight system may be acceptable for some designs provided that (1) the system leakage does not impact safety functions under all conditions, and (2) defense in depth is not impacted by system leakage."*

	Natrium PDC	Source Language	Basis for Change
V.	Reactor Containment		
50	Deleted.		The Natrium design does not include a pressure containing reactor containment structure. See the functional containment basis in Natrium PDC 16.
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	Natrium PDC	Source Language	Basis for Change
56	Deleted.		The Natrium design does not include a pressure containing reactor containment structure. See the functional containment basis in Natrium PDC 16.
57	Deleted.		The Natrium design does not include a pressure containing reactor containment structure. See the functional containment basis in Natrium PDC 16.

	Natrium PDC	Source Language	Basis for Change
VI.	Fuel and Reactivity Control		
60	*Control of releases of radioactive materials to the environment.* The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.	SFR-DC 60	No change.
61	*Fuel storage and handling and radioactivity control.* The fuel storage and handling, radioactive waste, and other systems that may contain radioactivity shall be designed to ensure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of safety-significant components ~~important to safety~~, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the safety significance ~~importance to safety~~ of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage cooling under accident conditions.	SFR-DC 61	See basis for safety-significant change in Natrium PDC 1.

	Natrium PDC	Source Language	Basis for Change
62	*Prevention of criticality in fuel storage and handling.* Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.	SFR-DC 62	No change.
63	*Monitoring fuel and waste storage.* Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.	SFR-DC 63	No change.
64	*Monitoring radioactivity releases.* Means shall be provided for monitoring the reactor building ~~containment~~ atmosphere, spaces containing components for primary system sodium and cover gas cleanup and processing, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.	SFR-DC 64	Change incorporates language from MHTGR-DC 64 to reflect functional containment approach. See basis in Natrium PDC 16. Base language from SFR-DC 64 was used to include SFR-specific reference to primary sodium and cover gas cleanup and processing.

	Natrium PDC	Source Language	Basis for Change
VII.	Additional PDC		
70	*Intermediate coolant system.* If an intermediate cooling system is provided, then the intermediate coolant system shall be designed with sufficient margin to assure that (1) the design conditions of the intermediate coolant boundary are not exceeded during normal operations, including anticipated occupational occurrences, and (2) the integrity of the primary coolant boundary is maintained during postulated accidents.	SFR-DC 70	No change.
71	*Primary coolant and cover gas purity control.* Systems shall be provided as necessary to maintain the purity of primary coolant sodium and cover gas within specified design limits. These limits shall be based on consideration of (1) chemical attack, (2) fouling and plugging of passages, and (3) radionuclide concentrations, and (4) air or moisture ingress as a result of a leak of cover gas.	SFR-DC 71	No change.

	Natrium PDC	Source Language	Basis for Change
72	*Sodium heating systems.* Heating systems shall be provided for safety-significant systems and components ~~that are important to safety, and~~ that contain or could be required to contain sodium. These heating systems and their controls shall be appropriately designed to ensure that the temperature distribution and rate of change of temperature in systems and components containing sodium are maintained within design limits assuming a single failure. If plugging of any cover gas line due to condensation or plate out of sodium aerosol or vapor could prevent accomplishing a safety function, the temperature control and the relevant corrective measures associated with that line shall be considered safety-significant ~~important to safety~~.	SFR-DC 72	See basis for safety-significant change in Natrium PDC 1.
73	*Sodium leakage detection and reaction prevention and mitigation.* Means to detect and identify sodium leakage as practical and to limit and control the extent of sodium-air and sodium-concrete reactions and to mitigate the effects of fires resulting from these sodium-air and sodium-concrete reactions shall be provided to ensure that the safety functions of safety-significant structures, systems, and components ~~important to safety~~ are maintained. Systems from which sodium leakage constitutes a significant safety hazard shall include measures for protection, such as inerted enclosures or guard vessels.	SFR-DC 73	See basis for safety-significant change in Natrium PDC 1.

	Natrium PDC	Source Language	Basis for Change
74	*Sodium/water reaction prevention/mitigation.* Structures, systems, and components containing sodium shall be designed and located to avoid contact between sodium and water and to limit the adverse effects of chemical reactions between sodium and water on the capability of any structure, system, or component to perform any of its intended safety functions. If steam-water is used for energy conversion, to prevent loss of any plant safety function, the sodium-steam generator system shall be designed to detect and contain sodium-water reactions and limit the effects of the energy and reaction products released by such reactions, including mitigation of the effects of any resulting fire involving sodium.	SFR-DC 74	No change.
75	*Quality of the intermediate coolant boundary.* Components that are part of the intermediate coolant boundary shall be designed, fabricated, erected, and tested to quality standards commensurate with the ~~importance of the~~ safety significance of the functions to be performed.	SFR-DC 75	See basis for safety-significant change in Natrium PDC 1.
76	*Fracture prevention of the intermediate coolant boundary.* Safety-significant components that are part of intermediate coolant boundary shall be designed with sufficient margin to ensure that, when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.	SFR-DC 76	See basis for safety-significant change in Natrium PDC 1. Language is also consistent with Natrium PDC 14.

	Natrium PDC	Source Language	Basis for Change
77	*Inspection of the intermediate coolant boundary*. Components that are part of the intermediate coolant boundary shall be designed to permit (1) periodic inspection and functional testing of important areas and features to assess their structural and leaktight integrity commensurate with the system's ~~importance to~~ safety significance, and (2) an appropriate material surveillance program for the intermediate coolant boundary.	SFR-DC 77	See basis for safety-significant change in Natrium PDC 1.

	Natrium PDC	Source Language	Basis for Change
78	*Primary Coolant System Interfaces.* When the primary coolant system interfaces with a structure, system, or component containing fluid that is chemically incompatible with the primary coolant, the interface location shall be designed to ensure that the primary coolant is separated from the chemically incompatible fluid by two redundant, passive barriers. When the primary coolant system interfaces with a structure, system, or component containing fluid that is chemically compatible with the primary coolant, then the interface location may be a single passive barrier provided that the following conditions are met: (1) postulated leakage at the interface location does not result in failure of the intended safety functions of safety-significant structures, systems or components ~~important to safety~~ or result in exceeding the specified acceptable system radionuclide release ~~fuel~~ design limits (2) the fluid contained in the structure, system, or component is maintained at a higher pressure than the primary coolant during normal operation, anticipated operational occurrences, shutdown, and accident conditions.	SFR-DC 78	See basis for safety-significant change in Natrium PDC 1. See basis for SARRDL change in Natrium PDC 10.
79	*Cover gas inventory maintenance.* A system to maintain cover gas inventory shall be provided as necessary to ensure that the primary coolant sodium design limits are not exceeded as a result of cover gas loss due to leakage from the primary coolant boundary and rupture of small piping or other small components that are part of the primary coolant boundary.	SFR-DC 79	No change.

	Natrium PDC	Source Language	Basis for Change
80	*Reactor vessel and reactor system structural design basis.* The design of the reactor vessel and reactor system shall be such that their integrity is maintained during postulated accidents (1) to ensure the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink and (2) to permit sufficient insertion of the neutron absorbers to provide for reactor shutdown.	MHTGR-DC 70	Adopted MHTGR-DC 70 without changes to support other Natrium PDC with MHTGR language. From RG 1.232, *"New MHTGR design-specific GDC are necessary to ensure that the reactor vessel and reactor system (including the fuel, reflector, control rods, core barrel, and structural supports) integrity is preserved for passive heat removal and for the insertion of neutron absorbers."*
81	*Reactor building design basis.* The design of the reactor building shall be such that, during postulated accidents, it structurally protects the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink ~~and provides a pathway for the release of reactor helium from the building in the event of depressurization accidents~~.	MHTGR-DC 71	Adopted MHTGR-DC 71 based on a review of RG 1.232 rationale for functional containment. Reference to helium release removed based on Natrium design. From RG 1.232, *"GDC 38, 39, 40, 41, 42, 43, 50, 51, 52, 53, 54, 55, 56, and 57 are not applicable to the MHTGR design, since they address design criteria for pressure-retaining containments in the traditional LWR sense. Requirements for the performance of the MHTGR reactor building are addressed by new Criterion 71 (design basis) and Criterion 72 (provisions for periodic testing and inspection)."* Since the Natrium design uses a functional containment approach, inclusion of these additional MHTGR-DC is appropriate.

	Natrium PDC	Source Language	Basis for Change
82	*Provisions for periodic reactor building inspection.* The reactor building shall be designed to permit (1) appropriate periodic inspection of all important structural areas ~~and the depressurization pathway~~, and (2) an appropriate surveillance program.	MHTGR-DC 72	Adopted MHTGR-DC 72 based on a review of RG 1.232 rationale for functional containment criteria. See additional notes in Natrium PDC 81.

7 CONCLUSION

Natrium PDC were developed in accordance with the requirements in 10 CFR 50.34(a)(3)(i). These PDC provide an appropriate set of requirements for use by future applicants to facilitate the design and licensing of the Natrium advanced reactor.

8 REFERENCES

1. U.S. Nuclear Regulatory Commission, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors,” RG 1.232, Revision 0, April 2018.

2. U.S. Nuclear Regulatory Commission, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors,” RG 1.233, Revision 0, June 20.

3. Nuclear Energy Institute, “Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light-Water Reactor Licensing Basis Development,” NEI Technical Report 18-04, Revision 1, August 2019.

4. Letter, Benjamin Beasley (NRC) to Peter Hastings (Kairos), “Safety Evaluation for Kairos Power LLC Topical Report “Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor” (Revision 1) (CAC No. 000431),” May 22, 2020. (ADAMS Accession Number ML20111A118)

5. U.S. Department of Energy Idaho National Laboratory, “Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors”, INL/EXT-14-31179, Revision 1, December 2014.

6. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Division 5.

7. U.S. Nuclear Regulatory Commission, “Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” SECY-94-084, March 1994. (ADAMS Accession No. ML003708068)

8. U.S. Nuclear Regulatory Commission, “Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements,” SRM-SECY-93-092, July 1993. (ADAMS Accession No. ML003760774)

9. U.S. Nuclear Regulatory Commission, “Policy Issues Related to Licensing Non-Light-Water Reactor Designs,” SRM-SECY-03-0047, June 2003. (ADAMS Accession No. ML031770124)

10. November 10, 2022, Public Meeting Presentation Materials, “Principal Design Criteria” and “Reactivity Control”. (ADAMS Accession No. ML22301A073)

END OF DOCUMENT