



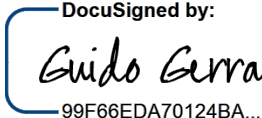
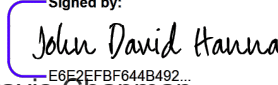
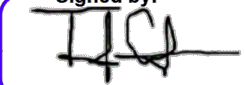
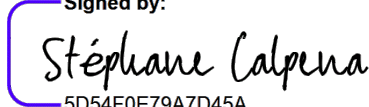
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Licensing Topical Report

Topical Report: LFR-AS-200 Principal Design Criteria Development Approach

ABSTRACT:

Topical Report for review by the U.S. Nuclear Regulatory Commission (NRC) on the approach to develop and define Principal Design Criteria for the LFR-AS-200 nuclear reactor technology.

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0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	3/71

TABLE OF CONTENTS

	EXECUTIVE SUMMARY	4
	ACRONYMS AND ABBREVIATIONS	5
1	INTRODUCTION.....	7
2	LFR-AS-200 TECHNOLOGY DESCRIPTION	8
3	REGULATORY ANALYSIS	9
	3.1 Title 10 of the Code of Federal Regulations, Part 53	9
	3.2 Regulatory Guide 1.232	10
	3.3 Regulatory Guide 1.233	11
	3.4 Regulatory Guide 1.253	11
4	LFR-AS-200 PDC DEVELOPMENT METHODOLOGY	12
5	SUMMARY OF CHANGES TO THE RG 1.232 ARDC	14
	5.1 Adaptation to the Licensing Modernization Project Framework	14
	5.1.1 Safety-Significant Structures, Systems, and Components	14
	5.1.2 10 CFR Part 53 Terminology	14
	5.1.3 Use of Specified Acceptable Radionuclide Release Design Limits.....	15
	5.1.4 Inspection and Testing Criteria Consolidated into PDC 6.....	16
	5.1.5 Removal of the Single-Failure Criterion	17
	5.2 LFR-AS-200 Technology-Specific Modifications.....	17
	5.2.1 Reactor Inherent Protection	17
	5.2.2 Reactivity Control Systems	18
	5.2.3 Reactor Coolant Inventory Maintenance	19
	5.2.4 Emergency Core Cooling System	19
	5.2.5 Functional Containment	20
	5.2.6 Additional LFR-AS-200-Specific Criteria	24
6	LFR-AS-200 PRINCIPAL DESIGN CRITERIA.....	25
7	CONCLUSION	61
8	REFERENCES.....	63



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	4/71

EXECUTIVE SUMMARY

This topical report describes the development of Principal Design Criteria (PDC) for the LFR-AS-200 advanced reactor. PDCs were developed using Regulatory Guides 1.232 [1] and 1.233 [2] based on design features specific to the LFR-AS-200 advanced reactor. *newcleo* is requesting United States (U.S.) Nuclear Regulatory Commission (NRC) review and approval of these PDC for use by *newcleo* in its technology development, including the development of functional design criteria for future use in licensing applications under Title 10 of the Code of Federal Regulations (10 CFR) Part 53 *Risk-Informed, Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants*.

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DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	5/71

ACRONYMS AND ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
ANS	American Nuclear Society
AOO	Anticipated Operational Occurrence
ARCAP	Advanced Reactor Content of Application Project
ARDC	Advanced Reactor Design Criteria
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BDBE	Beyond Design Basis Event
CDC	Complementary Design Criteria
CFR	Code of Federal Regulations
DBA	Design Basis Accident
DBE	Design Basis Event
DBHL	Design Basis Hazard Level
DID	Defense-in-Depth
DPA	Displacement Per Atom
ECCS	Emergency Core Cooling System
F-C	Frequency–Consequence
FOAK	First-of-a-Kind
GDC	General Design Criteria
IAEA	International Atomic Energy Agency
I&C	Instrumentation and Control
IDP	Integrated Decision-making Process
LBE	Licensing Basis Event
LFR	Lead-cooled Fast Reactor
LMP	Licensing Modernization Project
LOCA	Loss-of-Coolant Accident
LWR	Light-Water Reactor
MHTGR	Modular High Temperature Gas-Cooled Reactor
MHTGR-DC	Modular High Temperature Gas-Cooled Reactor Design Criteria
MOX	Mixed-Oxide
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSRSS	Non-Safety-Related but Safety-Significant
NSRST	Non-Safety-Related with Special Treatment
PDC	Principal Design Criteria
PRA	Probabilistic Risk Assessment
R&D	Research and Development

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0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	6/71

REP	Regulatory Engagement Plan
RG	Regulatory Guide
RFDC	Required Functional Design Criteria
RIPB	Risk-Informed, Performance-Based
RSF	Required Safety Function
RV	Reactor Vessel
SAFDL	Specified Acceptable Fuel Design Limit
SARRDL	Specified Acceptable Radionuclide Release Design Limit
SFR	Sodium-Cooled Fast Reactor
SFR-DC	Sodium-Cooled Fast Reactor Design Criteria
SR	Safety-Related
SRM	Staff Requirements Memoranda
SSC	Structure, System, and Component
TEDE	Total Effective Dose Equivalent
TICAP	Technology-Inclusive Content of Application Project
TRISO	TRistructural ISOtropic
U.S.	United States

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0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	7/71

1 INTRODUCTION

This topical report describes the approach taken by *newcleo* Americas LLC to develop Principal Design Criteria (PDC) for the LFR-AS-200 advanced nuclear reactor. The LFR-AS-200 is a 480 MWth (200 MWe) Lead-cooled Fast Reactor (LFR) technology designed for both electrical generation and industrial co-generation applications. A complete set of PDC are defined with bases as part of this development effort to enable execution of U.S.-oriented design activities. The *newcleo* Group (collectively *newcleo* in this report) is comprised of several business units with its newest subsidiary, *newcleo* Americas, established in late 2025 for the purpose of commercializing a Mixed Oxide (MOX) fuel facility and reactor (LFR) technology in the North American market.

newcleo intends to pursue licensing of the LFR-AS-200 through Title 10 of the Code of Federal Regulations (10 CFR) Part 53 *Risk-informed, Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants* [3] with the intent for each plant to be managed under a Class 103 commercial license. Options to enable efficient deployment of a first-of-a-kind (FOAK) LFR-AS-200 project are being developed that may leverage prototype testing with license conditions for the first unit to augment the demonstration of safety features through other test programs and analytical tools as envisioned in §53.440(a). U.S. regulatory engagement for the LFR-AS-200 was initiated with a pre-application meeting with the U.S. Nuclear Regulatory Commission (NRC) staff on March 24, 2026, at which the company presented its technology, preliminary licensing basis, and deployment ambitions. This was followed by the submission of a Regulatory Engagement Plan (REP), which laid out the necessary licensing actions for an initial U.S. deployment project and subsequent projects to follow.

The goal of this topical report is to outline the regulatory framework and development methodology used to define the PDC for the LFR-AS-200 reactor. A complete initial set of the PDC is provided for NRC review, though further revision may occur through the course of detailed design and will be subject to future reviews through amendment to this topical report or inclusion in a licensing action (e.g., Construction Permit application). While PDC are not defined within Part 53 specifically, the need to develop functional design criteria remains a regulatory requirement and this PDC development approach will support that future activity.

newcleo requests the NRC review and approval of these PDC for the LFR-AS-200 design. They will be used as part of, or in development of, the necessary functional design criteria to support future applications that may range from Limited Work Authorizations, Standard Design Approvals, Standard Design Certifications, Manufacturing Licenses, Construction Permits, Operating Licenses, or Combined Licenses under the applicable regulations in 10 CFR Part 53 [3]. This topical report will serve as a reference for developing the license application content for LFR-AS-200 advanced reactors.

Compliance of the LFR-AS-200 design with these PDC will be demonstrated in the Safety Analysis Report submitted as part of each license application. This topical report was prepared at an early stage of design, prior to completion of the plant's Probabilistic Risk Assessment (PRA) and full implementation of the Nuclear Energy Institute (NEI) 18-04 Licensing Modernization Project's risk-informed design approach and processes [4]. The PDC presented herein should therefore be understood as subject to revision as the design matures and PRA-informed insights become available, consistent with the iterative approach of the NEI 18-04 methodology and Part 53 functional design criteria development.

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0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	8/71

2 LFR-AS-200 TECHNOLOGY DESCRIPTION

The LFR-AS-200 is a 480 MWth (200 MWe) pool-type Lead-cooled Fast Reactor (LFR) developed by *newcleo* for commercial applications in the European and U.S. markets. The reactor core uses mixed-oxide (MOX) fuel in a pin-based fuel assembly, with multiple assemblies forming the reactor core. Heat is transferred by forced circulation of pure liquid lead to steam generators for power conversion in a Rankine power cycle. The plant architecture under consideration supports single and dual-unit configurations, with a possible two-on-one reactor-to-turbine generator arrangement. The use of lead as the primary coolant enables operation at near-atmospheric pressure and eliminates the need for an intermediate heat transfer system, owing to lead's chemical compatibility with water and air. The lead pool configuration also supports both forced and natural circulation as heat transfer means to various heat removal systems. Axially-oriented control rods and shutdown devices provide reactivity control similar to many other reactor technologies.

An introduction to the LFR-AS-200 technology, system nomenclature, and naming conventions is provided in previously submitted NRC presentation material [5].



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	9/71

3 REGULATORY ANALYSIS

newcleo intends to pursue licensing of the LFR-AS-200 through 10 CFR Part 53 [3], which is therefore the overarching applicable regulation for the present document. As the rule was still undergoing revision during initiation of this report, the specific term PDC was retained as it is commonly used in existing NRC regulation and guidance on safety design criteria development. Since the initial edition of Part 53 was fully published in 91 Federal Register Notice 15794, effective April 2026, the use of the technology-inclusive functional design criteria is acknowledged as part of the rule. It is *newcleo*'s position that development of PDC through the recent advanced reactor regulatory resources described here remains a helpful initiative to support design development and identifying acceptable approaches to key safety functions in a technology. Further development of these design criteria will occur through the development of the technology and any further adaptations necessary to address the scope of functional design criteria for use in Part 53 will occur, especially with the inclusion of new guidance developed to address Part 53 more specifically.

Regulatory Guide (RG) 1.232 [1] provides guidance on how to adapt the General Design Criteria (GDC) in 10 CFR Part 50, Appendix A [6], for non-Light Water Reactor (LWR) designs. While the technology-inclusive Advanced Reactor Design Criteria (ARDCs) proposed in RG 1.232 are derived from the GDCs of Part 50, Appendix A, the latter were developed for LWRs and are therefore judged not applicable to the LFR-AS-200.

RG 1.233 [2] endorses the technology-inclusive, Risk-Informed, Performance-Based (RIPB) methodology described in NEI 18-04, Rev. 1 [4] as an acceptable approach for informing the licensing basis and determining the appropriate scope and level of detail for applications for licenses, certifications, and approvals for non-LWRs.

3.1 TITLE 10 OF THE CODE OF FEDERAL REGULATIONS, PART 53

10 CFR Part 53, §§ 53.410 and 420 define the requirements in terms of functional design criteria for Safety-Related (SR) and Non-Safety-Related but Safety-Significant (NSRSS). Specifically, § 53.410 requires that functional design criteria be defined for each SR design feature in terms of its role in meeting the Design Basis Accident (DBA) safety criteria of § 53.210. Special treatments for SR Structure, System, and Components (SSCs) must be identified to achieve and maintain the reliability and capability needed to satisfy those functional design criteria. § 53.420 establishes an analogous requirement for SR and NSRSS design features for Licensing-Basis Events (LBEs) other than DBAs, in terms of their role in meeting the safety criteria of § 53.220 and the evaluation criteria of § 53.450(e). Special treatments for SR and NSRSS SSCs must be identified accordingly.

The following paragraphs of 10 CFR Part 53 [3] are also relevant to the design of safety-significant SSCs for an advanced non-LWR reactor:

- § 53.210 sets the quantitative DBA dose acceptance criteria that SR SSCs must be designed to satisfy.
- § 53.220 sets the safety criteria for all LBEs other than DBAs. This is the RIPB complement to § 53.210, requiring that SSCs, personnel, and programs provide the necessary capabilities and reliability to address the full LBE spectrum.

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0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	10/71

- § 53.230 defines the safety functions that must be fulfilled throughout plant life: the primary function is limiting radionuclide release; additional functions (reactivity control, heat removal, etc.) must be identified. All safety functions must be fulfilled by design features, human actions, and programmatic controls to satisfy §§ 53.210 and 53.220.
- § 53.250 requires that appropriate defense in depth be provided for each commercial nuclear plant to compensate for uncertainties—in modeling capabilities, barrier performance, SSC and personnel reliability, and programmatic controls—such that there is reasonable assurance that the safety criteria of §§ 53.210 and 220 are met over the plant's lifetime. The safety analysis may not exclusively rely on any single engineered design feature, human action, or programmatic control, however robust, to address the range of LBEs other than DBAs.
- § 53.400 requires that design features be provided—in combination with human actions and programmatic controls—to satisfy the safety criteria of §§ 53.210 and 53.220 and to fulfill the safety functions of § 53.230 during all LBEs.
- § 53.415 requires that SR SSCs be protected against or designed to withstand natural phenomena and constructed hazards up to the design-basis external hazard levels, without losing safety function capability. Seismic-specific requirements are addressed in § 53.480.
- § 53.425 requires design features and functional design criteria to support the radiation protection program (§ 53.850) during normal operation, including dose objectives for the maximally exposed member of the public and control of liquid, gaseous, and solid radioactive wastes per 10 CFR Part 20.
- § 53.430 requires design features and functional design criteria to protect plant workers, such that—combined with programmatic controls—the occupational dose requirements of § 53.270 are met.
- § 53.440 establishes eleven specific design requirements for SR and NSRSS SSCs: (a) performance demonstration through analysis, testing, or experience; (b) use of NRC-endorsed consensus codes and standards for SR SSCs; (c) material qualification over the design life; (d) evaluation of degradation mechanisms; (e) fire and explosion protection; (f) safety-security integration; (g) subcriticality capability; (h) long-term cooling capability; (i) multi-unit considerations; (k) chemical hazard risk; (l) contamination minimization and decommissioning facilitation; (m) criticality monitoring; and (n) human factors engineering, including functional requirements analysis and function allocation.
- § 53.460 requires SSCs to be classified as SR, NSRSS, or Non-Safety-Significant. For SR and NSRSS SSCs, service conditions must be identified and special treatments established commensurate with safety significance. SR SSCs must meet the quality assurance requirements of Appendix B to 10 CFR Part 50 [6], while NSRSS SSCs may receive graded special treatments. Human actions credited in the safety analysis must also be accounted for in the special treatment identification.

3.2 REGULATORY GUIDE 1.232

RG 1.232 [1] provides examples and guidance on how to adapt the GDCs contained in 10 CFR Part 50, Appendix A [6], for non-LWR designs. It presents three sets of design criteria that can be implemented in the PDC for a technology: the ARDCs, a technology-inclusive implementation, the Sodium-Cooled Fast Reactor Design Criteria (SFR-DCs), and the Modular High-Temperature Gas-Cooled Reactor Design Criteria (MHTGR-DCs).

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0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	11/71

RG 1.232 predates the Licensing Modernization Project (LMP) and is therefore prescriptive and deterministic in nature, rather than being aligned with the RIPB framework of RG 1.233 [2]. It is nevertheless the reference guidance document for the development of PDCs for an advanced non-LWR design, and its ARDCs serve as the structural starting point from which LMP-aligned adaptations are made. The LFR-AS-200 PDC development follows this guidance.

3.3 REGULATORY GUIDE 1.233

RG 1.233 [2] endorses the methodology described in NEI 18-04 [4] as an acceptable approach for informing the licensing basis and content of applications for licenses, certifications, and approvals for non-LWRs. It is methodological rather than prescriptive in character: rather than specifying design criteria, it describes how to select LBEs, classify SSCs, assess defense-in-depth adequacy, and determine the appropriate scope and level of detail for safety analysis reports.

RG 1.233 was developed after RG 1.232 and is fully aligned with the RIPB framework. Its technology-inclusive, LBE-driven methodology is the conceptual predecessor to the 10 CFR Part 53 framework, and the LFR-AS-200 PDC development follows this methodology to ensure consistency with Part 53.

3.4 REGULATORY GUIDE 1.253

RG 1.253 [7] endorses two complementary sets of guidance for non-LWR license applications: the Technology-Inclusive Content of Application Project (TICAP), documented in NEI 21-07 [8], and the Advanced Reactor Content of Application Project (ARCAP), documented in the nine DANU-ISG-2022 interim staff guidance documents. Together, these constitute the NRC staff's guidance on the scope and depth of information required in a non-LWR license application under 10 CFR Parts 50 and 52.

Of particular relevance to this topical report, RG 1.253 endorses Section C, Chapter 5 of NEI 21-07 [8], which provides the systematic approach for identifying PDCs under the LMP process—including the Quality Assurance PDC (Section 5.3.1), the Required Functional Design Criteria (RFDC, Section 5.3.2), and the Complementary Design Criteria (CDC, Section 5.6). RG 1.253 confirms that both PDC-RFDCs and PDC-CDCs together constitute a complete set of PDCs, and that any gaps in PDC coverage—primarily relating to NSRSS SSCs—should be addressed through PDC-CDCs consistent with NEI 21-07, Section 5.6. The LFR-AS-200 PDC development follows this guidance. *newcleo* has also reviewed the precedent guidance developed in the NRC and industry's efforts in the Technology Inclusive Content of Application and Advanced Reactor Content of Application efforts of the early 2020's to understand how PDC development was addressed through these initiatives.

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	12/71

4 LFR-AS-200 PDC DEVELOPMENT METHODOLOGY

10 CFR Part 53, § 53.410 requires that functional design criteria be defined for each design feature classified as SR in terms of its role in demonstrating compliance with the DBA safety criteria of § 53.210 [3]. § 53.420 establishes an analogous requirement for SR and NSRSS design features in terms of their role in demonstrating compliance with the safety criteria for LBEs other than DBAs (§ 53.220). The systematic, RIPB approach described in NEI 21-07 [8] for identifying PDCs is an alternative to the traditional deterministic approach used for LWRs. The information contained within a design-specific set of PDCs developed using this approach may differ from that contained within the GDCs in 10 CFR 50, Appendix A, or the ARDCs in RG 1.232 [1]. Neither the GDCs nor the ARDCs are regulatory requirements for non-LWR applicants, though they do provide guidance on the scope of PDCs to be developed.

PDCs establish the necessary design, fabrication, construction, testing, and performance requirements for safety-significant SSCs and are composed of three types of criteria [7]:

- The Quality Assurance Principal Design Criterion, which addresses the graded approach to special treatments for SSCs performing safety-significant functions.
- Principal Design Criteria – Required Functional Design Criteria (PDC-RFDCs), which establish the functional requirements of the plant that are required to meet the performance objectives of the Required Safety Functions (RSFs) and are satisfied by SR SSCs.
- Principal Design Criteria – Complementary Design Criteria (PDC-CDCs), which establish requirements for SSCs identified as NSRSS because they perform risk-significant functions or are identified as necessary for Defense in Depth (DID).

The PDC-RFDCs and PDC-CDCs are established by applying the NEI 18-04 [4] methodology¹. NEI 18-04 describes acceptable processes for the selection of LBEs, the safety classification of SSCs and associated risk-informed special treatments, and the determination of DID adequacy applicable to advanced non-LWR designs. Central to the design of the DID provisions is the Integrated Decision-making Process (IDP) defined in NEI 18-04. The reactor designer is responsible for ensuring that DID is achieved through the incorporation of DID features and programs in the design phases and, in turn, conducting the evaluation that arrives at the decision of whether adequate DID has been achieved. The process in NEI 18-04 calls for the reactor designer to establish an IDP, which supports the overall design effort (including development of plant capability and programmatic DID features), conducts the DID adequacy evaluation of that resulting design, and documents the DID baseline. Any SSCs required to meet the plant capability DID guidelines, as determined by the IDP, would be regarded as performing a safety function necessary for adequacy of plant capability DID.

The NEI 18-04 process is inherently iterative, resulting in direct consequences for the present stage of design. Before the first introduction of the design-specific PRA, a designer needs to develop a technically sound understanding of the potential failure modes of the reactor concept, how the reactor plant would respond to such failure modes, and how protective strategies will be incorporated into formulating the safety-design

¹ Use of the NEI 18-04 methodology will inform the development of PDC for the facility but may not include the entirety of PDC necessary to demonstrate adequate protection of public health and safety. PDC not informed by the NEI 18-04 methodology and not addressed in the TICAP guidance [7],[8] should be addressed in the ARCAP guidance [8].

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	13/71

approach. A designer can use safety analysis methods appropriate to early stages of design, such as failure modes and effects analyses and process hazard analyses.

Accordingly, the initial definition of PDCs and the associated safety functions presented in this report are grounded in a deterministic analysis of the fundamental safety functions (reactivity control, heat removal, and radionuclide confinement/functional containment) rather than in a completed design-specific PRA. The subsequent use of the PRA by the designer will support the development or confirmation of the events, safety functions, key SSCs, and adequacy of DID, and will provide a structured framework to risk-inform the application for the specific reactor design. The PRA will identify which safety functions are required to be fulfilled to demonstrate compliance with the safety criteria of § 53.210 for DBAs². Such safety functions will be classified as RSFs and will lead to the creation of Required Functional Design Criteria (RFDC) for the SR SSCs relied upon to perform them. Safety functions identified as necessary to meet the safety criteria of § 53.220 for LBEs other than DBAs, or as necessary for adequate defense in depth, will similarly inform the development of Complementary Design Criteria (CDCs) for the NSRSS SSCs relied upon to perform them. Ultimately, the determination of such RFDCs and CDCs may be used to supplement or modify the PDCs.

It should be noted that, per RG 1.253 [7], both PDC-RFDCs and PDC-CDCs together constitute a complete set of PDCs. At this stage of the design, PRA-informed insights are not yet available to differentiate design criteria into PDC-RFDCs and PDC-CDCs. The initial safety architecture of the LFR-AS-200 was developed following deterministic criteria inspired by the International Atomic Energy Agency (IAEA) SSR-2/1 [9], which is broadly consistent with Task 3 of the NEI 18-04 IDP (“Define SSC Safety Functions for PRA Modeling”) [4]. **The PDCs presented in this document should therefore be construed as the overarching design criteria applicable to all safety-significant SSCs.** Differentiation into PDC-RFDCs and PDC-CDCs is deferred to a later design stage, consistent with the iterative approach of NEI 18-04.

Moreover, given the inherently iterative nature of the NEI 18-04 process, further refinement of a given PDC may be required for each future design iteration. As a result, the PDCs presented in this document do not yet fully reflect the NEI 18-04 IDP and will be revisited and refined in the future with the introduction of PRA-informed insights and with each design iteration.

The PDCs defined for the LFR-AS-200 are based on the ARDCs contained in RG 1.232 Appendix A [1], which are judged broadly applicable to the LFR-AS-200 technology. The ARDCs have been adapted to align with the LMP framework, specifically with 10 CFR Part 53 [3]. Several ARDCs were modified to account for the specificities of pool-type LFRs and for the functional containment approach adopted by the LFR-AS-200 (cf. Section 5.2.5). Some PDCs were derived from the corresponding SFR-DCs or MHTGR-DCs of RG 1.232 Appendices B and C, respectively.

A summary of the changes and adaptations implemented to the RG 1.232 ARDCs for the development of the LFR-AS-200 PDCs is provided in Section 5. The full list of LFR-AS-200 PDCs, along with their bases, are provided in Section 6.

² The different definition of SR SSCs in Part 53 (SSCs relied upon to demonstrate compliance with the DBA safety criteria) compared to NEI 18-04 (SSCs required to maintain the consequence of one or more DBEs or the frequency of one or more high-consequence BDBEs within the NEI 18-04 Frequency-Consequence target) implies a different meaning of RFDC between the two frameworks. The Part 53 sense is retained in this document (cf. subsection 5.1.2).

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	14/71

5 SUMMARY OF CHANGES TO THE RG 1.232 ARDC

5.1 ADAPTATION TO THE LICENSING MODERNIZATION PROJECT FRAMEWORK

5.1.1 Safety-Significant Structures, Systems, and Components

The term “important to safety” used in RG 1.232 (and in the GDCs of 10 CFR Part 50, Appendix A [6]) is replaced throughout the LFR-AS-200 PDCs with “safety-significant”. In the LMP framework, as endorsed by RG 1.233, the collective term for SSCs warranting special treatment—encompassing both SR and NSRSS SSCs—is “safety-significant.” The NRC has confirmed that “safety-significant” is equivalent to “important to safety” in this context and is consistent with the guidance in NEI 18-04 [4] (as endorsed in RG 1.233 [2]) and NEI 21-07 [8] (as endorsed in RG 1.253 [7]), as well as several precedent technology PDC reviews that implement LMP.

5.1.2 10 CFR Part 53 Terminology

The methodology of NEI 18-04 [4] (endorsed by RG 1.233 [2]) and the 10 CFR Part 53 [3] framework employ partially different terminology to describe equivalent or largely equivalent concepts. The terms “anticipated event sequences”, “unlikely event sequences”, and “very unlikely event sequences” in Part 53 [3] are defined to be equivalent to the LMP’s definitions of Anticipated Operational Occurrences (AOOs), Design-Basis Events (DBEs), and Beyond-Design-Basis Events (BDBEs), respectively.

§ 53.020 [3] defines SR SSCs as “*those SSCs that are relied upon to demonstrate compliance with the safety criteria in § 53.210 and warrant special treatment*”. In contrast, NEI 18-04 [4] defines SR SSCs as “*SSCs selected by the designer from the SSCs that are available to perform the RSFs to mitigate the consequences of DBEs to within the LBE F-C Target, and to mitigate DBAs that only rely on the SR SSCs to meet the dose limits of 10 CFR 50.34 using conservative assumptions, and SSCs selected by the designer and relied on to perform RSFs to prevent the frequency of BDBE with consequences greater than the 10 CFR 50.34 dose limits from increasing into the DBE region and beyond the F-C Target*” [4]. Similarly, RG 1.233 [2] defines SR SSCs as those “*selected by the reactor designer from the SSCs available to perform the required safety functions to mitigate the consequences of DBEs to within the LBE F-C target and to mitigate DBAs that only rely on SR SSCs to meet the 10 CFR 50.34 dose limits*”, as well as SSCs “*relied on to perform required safety functions to prevent the frequency of BDBEs with consequences greater than the 10 CFR 50.34 dose limits from increasing into the DBE region and beyond the F-C target.*”

The § 53.020 definition of DBAs (“*postulated event sequences that are used to set functional design criteria and performance objectives for the design of SR SSCs through deterministic analyses. Design-basis accidents are a type of licensing-basis event and are based on the capabilities and reliabilities of SR SSCs needed to mitigate and prevent event sequences, respectively*”) establishes a bijective correspondence between the SR SSC class and DBAs³. This bijective correspondence is absent in the NEI 18-04/RG 1.233 framework, where SR SSCs have a broader role. It is newcleo’s position that the Part 53 definition of SR

³ This is confirmed by the requirement in § 53.450(f)(1) that the events selected as DBAs must be those that, if not terminated, have the potential for exceeding the safety (dose) criteria in § 53.210.



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	15/71

SSCs therefore has a narrower scope than that of NEI 18-04 and RG 1.233: Part 53 SR SSCs may be a subset of the NEI 18-04/RG 1.233 SR SSCs. **newcleo intends to adopt the Part 53 definition of SR SSCs.**

It is noted that the § 53.020 definition of DBAs does not provide an explicit relationship with DBEs. The latter is mentioned in the preamble for issuance of Part 53 in 91 FR 15696 (Federal Register Vol. 91, No. 60 [3]), which states that “*Design-basis accidents are derived from the unlikely event sequences from the PRA [...] and then analyzed in a conservative approach by prescriptively assuming that only SR SSCs are available to mitigate postulated accident scenarios.*” Since “*unlikely event sequences*” are the Part 53 equivalent of DBEs, the Part 53 and NEI 18-04 definitions of DBAs⁴ are considered equivalent.

The definition of NSRSS SSCs contained in § 53.020 (“*those SSCs which are not SR but are relied on to achieve adequate defense in depth or perform risk-significant functions and warrant special treatment*”) differs slightly from the NEI 18-04 definition of Non-Safety-Related with Special Treatment (NSRST) SSCs (*non-safety-related SSCs relied on to perform risk-significant functions or functions requiring special treatment for DID adequacy. Risk-significant SSCs are those that perform functions that prevent or mitigate any LBE from exceeding the F-C Target or make significant contributions to the cumulative risk metrics selected for evaluating the total risk from all analyzed LBEs*). The two definitions are considered to be effectively equivalent.

Unless otherwise noted, the Part 53 terminology is employed in this document.

5.1.3 Use of Specified Acceptable Radionuclide Release Design Limits

The Specified Acceptable Radionuclide Release Design Limit (SARRDL) concept was introduced in the MHTGR-DC as a direct replacement for Specified Acceptable Fuel Design Limits (SAFDLs). Rather than setting limits on fuel cladding integrity, SARRDLs limit the total radionuclide inventory released by the system—fuel, coolant circuit, and all connected systems that may contribute to dose—during normal operation and anticipated event sequences (cf. basis for MHTGR-DC 10 in RG 1.232 [1]). The concept is therefore source-term-based rather than barrier-integrity-based. It was developed specifically around the characteristics of coated particle fuel (e.g., tri-structural isotropic, or TRISO, fuel), which is the primary fission product barrier in an MHTGR and is expected to have very low incremental fission product release during AOOs. The TRISO fuel fission product transport and retention behavior under all expected operating conditions is the key to meeting dose limits, as a different approach to defense in depth is employed in an MHTGR (cf. basis for MHTGR-DC 10 in RG 1.232 [1]).

At the time of publication of RG 1.232 [1], the NRC had not approved the concept of replacing SAFDLs with SARRDLs for advanced reactors in general, noting that the concept of TRISO fuel being the primary fission product barrier is intertwined with the concept of a functional containment for MHTGR technologies (cf. basis for MHTGR-DC 10 in RG 1.232 [1]). Subsequent licensing activity has established that SARRDLs are not

⁴ NEI 18-04 defines DBAs as “*Postulated accidents that are used to set design criteria and performance objectives for the design of Safety-Related SSCs. DBAs are derived from DBEs based on the capabilities and reliabilities of Safety-Related SSCs needed to mitigate and prevent accidents, respectively. DBAs are derived from the DBEs by prescriptively assuming that only SR SSCs classified are available to mitigate postulated accident consequences to within the 10 CFR 50.34 dose limits.*”

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	16/71

inherently tied to coated particle fuel. For instance, the NRC issued a final safety evaluation in August 2024 accepting the use of SARRDLs in lieu of SAFDLs for a sodium-cooled fast reactor technology [10].

It is noted that the original association between SARRDLs and MHTGRs with coated particle fuel has evolved into a more generic association between SARRDLs and functional containment. As affirmed in SECY-18-0096 [11], SARRDLs represent a performance-based approach for determining functional containment performance criteria that is applicable to any non-LWR design, consistent with the RIPB framework of NEI 18-04 [4]. *newcleo* implements a functional containment approach and overall safety design approach that address the limits and conditions issued previously to other non-HTGR technologies that chose to adopt SARRDLs.

The LFR-AS-200 technology:

- It is pursuing a RIPB licensing process under 10 CFR Part 53 [3] and based on the methodology endorsed by RG 1.233 [2].
- It adopts a functional containment approach, as described in detail in Section 5.2.5.
- It follows the SARRDL-related MHTGR-DCs defined in RG 1.232 [1] as the basis for its applicable PDCs.
- It relies on a single-phase coolant operating at near atmospheric pressure, on high-stability ceramic oxide fuel and on a high-temperature, high-burnup, high-dpa-resistant, and liquid-metal corrosion-resistant titanium alloy and/or austenitic stainless-steel cladding material.

5.1.4 Inspection and Testing Criteria Consolidated into PDC 6

RG 1.232 does not include a single generic monitoring, inspection, testing, and surveillance criterion applicable to all safety-significant SSCs. Instead, inspection and testing requirements are distributed across six individual ARDCs, each addressing a specific system or system category: ARDC 18 (electric power systems), ARDC 32 (reactor coolant boundary), ARDC 36 (inspection of emergency core cooling system), ARDC 37 (testing of emergency core cooling system), ARDC 45 (inspection of structural and equipment cooling systems), and ARDC 46 (testing of structural and equipment cooling systems).

The LFR-AS-200 PDC set consolidates these six criteria into a single generic criterion (the new PDC 6), which establishes monitoring, inspection, testing, and surveillance requirements for all safety-significant SSCs commensurate with their safety significance. This approach is consistent with the precedent established by other recent advanced reactor technologies [12], which similarly combined the corresponding design criteria elements into a single criterion. The consolidation is motivated by the fact that the functional performance intent of the six source ARDCs—ensuring the integrity and operational readiness of safety-significant SSCs through periodic monitoring, inspection, functional testing, and surveillance—is common to all of them and is more efficiently expressed as a single technology-inclusive criterion.

The specific monitoring, inspection, testing, and surveillance requirements for each SR and NSRSS SSC will be established as special treatments in accordance with the NEI 18-04 IDP [4], consistent with § 53.460(b) of 10 CFR Part 53 [3]. This approach ensures that monitoring, inspection, testing, and surveillance objectives—including accessibility, inspectability, surveillance provisions, and inspection qualification requirements—are defined commensurate with the safety significance of each SSC and meet the functional

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DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	17/71

performance intent of the individual ARDCs that PDC 6 replaces. It is noted that certain requirements of the source ARDCs—specifically the structural and leak-tight integrity provisions of ARDCs 32, 37, and 46, and the material surveillance program for the reactor vessel (RV) of ARDC 32—are implementation-level requirements that flow from the SSC safety classification process and are therefore addressed as special treatments rather than as explicit design criteria in PDC 6.

5.1.5 Removal of the Single-Failure Criterion

The single-failure criterion is a deterministic design requirement embedded in several ARDCs of RG 1.232 [1], most notably in ARDCs 17 (electric power systems), 21 (protection system reliability and testability), 34 (residual heat removal), and 44 (structural and equipment cooling), as well as SFR-DC 72 (which is used as the basis for LFR-AS-200 PDC 72). In its standard formulation, it requires that the safety function of a system can be accomplished assuming a single active component failure.

The single-failure criterion is a legacy of the deterministic LWR regulatory framework of 10 CFR Parts 50 and 52 and is not aligned with the RIPB philosophy of the LMP. As stated in Staff Position C.1.d of RG 1.233 [2], the NRC staff determined that the approach described in NEI 18-04 [4] (which subjects the design to a probabilistic reliability criterion and to an evaluation of defense-in-depth adequacy through the IDP) is consistent with the Commission's direction in SRM-SECY-03-0047 [13] to replace the single-failure criterion with a probabilistic reliability criterion. The NEI 18-04 IDP provides a more comprehensive treatment of system reliability than the single-failure criterion: by requiring a full probabilistic assessment of SSC reliability—including uncertainty quantification—it ensures that all failure modes are addressed and that the effect of uncertainties is accounted for, without the need to impose a prescriptive deterministic margin.

Accordingly, any reference to the single-failure criterion is removed from all applicable LFR-AS-200 PDCs. In PDC 21, this reference is replaced with the phrase “sufficient reliability”, reflecting the graded, risk-informed approach of the LMP framework. The redundancy of SR and NSRSS SSCs will be established through the NEI 18-04 IDP and reflected in the functional design criteria developed in accordance with §§ 53.410 and 53.420 of 10 CFR Part 53 [3].

5.2 LFR-AS-200 TECHNOLOGY-SPECIFIC MODIFICATIONS

5.2.1 Reactor Inherent Protection

The core of the LFR-AS-200 was designed with negative core-wide reactivity feedback. A core with sufficiently negative overall reactivity feedback provides an inherent stabilizing response to power excursions by reducing reactor power as fuel and coolant temperatures increase. In an Anticipated Transient Without Scram (ATWS) event sequence, the inherent reactivity feedback limits the magnitude of the power excursion and the resulting thermal challenge to the core, thereby supporting core cooling capability and contributing to the prevention or mitigation of severe plant states. More broadly, the inherent feedback mechanisms contribute to the safe response to licensing-basis power excursions in the power operating range.

PDC 11 strengthens ARDC 11 by extending the requirement for negative *prompt* inherent nuclear feedback to a requirement for negative *global* reactivity feedback. GDC 11 [6] and ARDC 11 [1] require only that the

SÉCURITÉ NATIONALE FR

NON PROTEGE

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	18/71

prompt inherent nuclear feedback (primarily the Doppler coefficient) tends to compensate for a rapid increase in reactivity, while the delayed feedback contributions (e.g., positive coolant temperature or core expansion coefficient) and the net total feedback may remain positive. Requiring negative global reactivity feedback therefore ensures that the reactor is self-stabilizing not only on the prompt timescale but throughout the event. This design goal is relevant to the LFR-AS-200, where the large thermal inertia of the primary system means that delayed feedback terms play a significant role in determining the ultimate power level reached during a transient.

The negative core-wide reactivity feedback of the LFR-AS-200 also contributes to PDC 26. The self-stabilizing character of the core ensures core cooling capability is maintained by limiting the thermal challenge to the core during power excursions, thereby contributing to the core cooling capability portion of ARDC 26 (3) by inherent features (see subsection 5.2.2).

Since the LFR-AS-200 PDC 11 assigns a wider safety role to the core inherent reactivity properties than the original ARDC 11 language—contributing not only to reactivity control but also to the core cooling function—a performance criterion consistent with the LMP framework is needed to capture this broader scope. The following phrase is therefore added at the end of the sentence: “to adequately address the additional heat generation via the available cooling provisions and to ensure that the specified acceptable fuel design limits and/or the specified acceptable radionuclide release design limits are met during anticipated event sequences, unlikely event sequences, and design basis accidents.”

N.B.: the requirement for negative global reactivity feedback in PDC 11 reflects a designer's choice aimed at minimizing the risk of severe plant conditions resulting from an ATWS, made prior to the full application of the NEI 18-04 methodology. The NEI 18-04 IDP will confirm or refine this choice as the design matures.

5.2.2 Reactivity Control Systems

The LFR-AS-200 initial reactivity control architecture was developed following deterministic criteria inspired by IAEA SSR-2/1 (in particular Requirement 46 [9]), consistent with Task 3 of the NEI 18-04 IDP. Two independent engineered reactivity control systems (control rods and shutdown devices) are provided, relying on movable neutron absorbers. In addition, the overall negative inherent reactivity feedback (cf. Section 5.2.1) ensures the self-stabilization of the core neutronic power in case of failure of both systems. Negative global reactivity feedback is a design requirement of the LFR-AS-200 (cf. Section 5.2.1 and PDC 11) that limits the magnitude of power excursions during reactivity insertion events, thereby effectively supporting the core cooling capability even accounting for the large thermal inertia of the primary circuit. The application of the NEI 18-04 IDP methodology will ensure the initial architecture complies with the DID adequacy criteria.

PDC 26 replaces the prescriptive requirement of ARDC 26—which mandates a minimum of two reactivity control systems or means irrespective of the specific technology and overall reactivity control architecture—with a more consistent DID approach, where the adequacy of the DID architecture is demonstrated via the application of the NEI 18-04 IDP [4]. The four points of ARDC 26 are collectively addressed by the reactivity control provisions of the LFR-AS-200, with a graded level of DID commensurate with the safety significance of the licensing-basis event to be addressed. The phrase “with appropriate reliability, independence and

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	19/71

diversity commensurate with the safety significance of the event" in the introductory sentence of the LFR-AS-200 PDC 26 addresses the entirety of the ARDC 26 DID requirements.

As discussed in Section 5.2.1, the core's inherent overall negative reactivity feedback contributes to satisfying the first element of ARDC 26 (3)—maintaining core cooling capability—by limiting the thermal challenge to the core. While the insertion of the movable poisons (control rods and shutdown devices) reduces core thermal power and thereby reduces the demand on the heat removal provisions, these engineered features are not required to fulfill the core cooling function. By contrast, the overall negative reactivity feedback does not guarantee the second element of ARDC 26 (3), i.e., reaching and maintaining a subcritical state following a postulated accident, for which movable poisons are required. The LFR-AS-200 PDC 26 point (3) therefore only addresses the latter RSF ("shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following an unlikely event sequence or design basis accident").

The NRC staff found this discretization of PDC 26 previously acceptable because it, along with PDC 11, addresses the underlying intent of MHTGR-DC 26 in RG 1.232, conditioning acceptance on the implementation of the NEI 18-04 methodology applied to the reactivity control function [12]. The LFR-AS-200 PDC 26 follows the same approach: the underlying intent of ARDC 26 is fully addressed through the PDC 11 and 26 and the NEI 18-04 IDP, which will establish the specific reliability, independence, and diversity requirements for the LFR-AS-200 reactivity control systems commensurate with their safety significance.

5.2.3 Reactor Coolant Inventory Maintenance

The need for reactor coolant inventory maintenance in an LWR results from the need to maintain primary coolant inventory, which is necessary to ensure adequate heat removal from the fuel. For a pool-type LFR, the primary coolant system is fundamentally different from that of an LWR, with the primary coolant maintained at atmospheric pressure and the reactor core immersed in a large pool of coolant. Core cooling in the LFR-AS-200 is ensured by maintaining a sufficient flow path for the primary coolant to transfer decay heat from the fuel to an available heat sink, which in turn is ensured by maintaining a sufficient primary coolant level inside the reactor vessel. By minimizing the annular gap between the reactor vessel and the reactor cavity, the minimum primary coolant level is ensured by design even in event sequences involving reactor vessel leakage, and primary coolant makeup is therefore not an applicable safety function for a pool-type LFR design.

As in the case of MHTGRs, although for different physical reasons, the LFR-AS-200 does not require reactor coolant inventory maintenance for small leaks to meet the SARRDLs. Therefore, ARDC 33 is not retained in the LFR-AS-200 PDC set.

5.2.4 Emergency Core Cooling System

The need for an Emergency Core Cooling System (ECCS) in an LWR results from the main family of accident scenarios for those reactor technologies, specifically Loss of Coolant Accidents (LOCAs), in which primary coolant inventory must be maintained to ensure adequate heat removal from the fuel. For a pool-type LFR, the primary coolant system is fundamentally different from that of an LWR, with the primary coolant maintained at atmospheric pressure and the reactor core immersed in a large pool of coolant. Consequently,

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	20/71

the high-pressure primary-system LOCA mechanism that is characteristic of LWRs—and therefore the need for an LWR-type ECCS—is not applicable to a pool-type LFR design. Core cooling for the LFR-AS-200 is instead ensured by maintaining a sufficient flow path for the primary coolant to transfer decay heat from the fuel to an available heat sink.

The emergency core cooling function as defined in ARDC 35 is therefore not applicable to the LFR-AS-200 design, and PDC 35 is renamed accordingly to “Passive Decay Heat Removal”. Following a convention common in the Generation IV community, *newcleo* uses “*residual* heat removal” for the cooling function in normal operation and anticipated event sequences, and “*decay* heat removal” for the safety function in accident scenarios (unlikely event sequences, design basis accidents, and very unlikely event sequences). Since cooling in normal operation and anticipated event sequences is addressed by PDC 34, the title of PDC 35 uses “decay heat removal” rather than “residual heat removal”. The “passive” qualifier originates from the passive nature of the LFR-AS-200 safety architecture in accident scenarios, in line with Generation IV goals.

It is noted that the separation of residual heat removal (PDC 34) and passive decay heat removal (PDC 35) design criteria follows the defense-in-depth requirement of § 53.250(c) of 10 CFR Part 53 [3] that no single engineered design feature can be exclusively relied upon to address the range of licensing basis events other than design basis accidents.

5.2.5 Functional Containment

As mentioned in Section 5.1.3 above, the concept of ‘functional containment’ was originally introduced for the MHTGR technology with coated particle (TRISO) fuel. As stated in the basis for MHTGR-DC 16 in RG 1.232 [1], “*the term ‘functional containment’ is applicable to advanced non-LWRs without a pressure retaining containment structure. A functional containment can be defined as ‘a barrier, or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, AOOs, and accident conditions.’ Functional containment is relied upon to ensure that dose at the site boundary as a consequence of postulated accidents meets regulatory limits. Traditional containment structures also provide the reactor and SSCs important to safety inside the containment structure protection against accidents related to external hazards (e.g., turbine missiles, flooding, aircraft impacts). The NRC staff has brought the issue of functional containment to the Commission, and the Commission has found it generally acceptable, as indicated in the staff requirements memoranda (SRM) to SECY-93-092 (Ref. 8) and SECY-03-0047 (Ref. 9).*”

Shortly after the publication of RG 1.232, the NRC’s Advisory Committee on Reactor Safeguards (ACRS) supported the application of the functional containment concept to all non-LWR designs in a letter dated May 10, 2018 (ML18108A404 [14]), which informed the preparation of the NRC staff guidance for establishing functional containment performance criteria for generic non-LWR designs (SECY-18-0096 [11]). The following paragraphs describe the key pillars on which the LFR-AS-200 functional containment strategy is built.

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	21/71

(i) **Risk-informed performance-based radionuclide retention criteria**

SECY-18-0096 [11] documents a proposed RIPB process for developing functional containment performance criteria that takes advantage of mechanistic source term analyses and indicates that the use of a functional containment approach necessitates the identification of performance criteria for SSCs that play a role in radionuclide retention. SRM-SECY-18-0096 [15] records the Commission's approval of this methodology. The design of the LFR-AS-200 follows the LMP framework of 10 CFR Part 53 [3], NEI 18-04 [4], and RG 1.233 [2], which align with the process documented in SECY-18-0096 [11].

(ii) **Incorporation of the functional containment concept into the LFR-AS-200 PDCs**

The LFR-AS-200 PDCs incorporate the MHTGR-DCs from RG 1.232 that are related to the functional containment concept. In particular, the LFR-AS-200 adopts language from MHTGR-DC 16 (adapted in line with the LMP framework and to account for the different technology), eliminates ARDCs 38–43 and 50–57, and adopts MHTGR-DCs 70 and 71 (with LMP-compliant and technology-specific modifications). Moreover, use of the functional containment concept also implies the use of SARRDLs (as discussed in Section 5.1.3), which are introduced in PDCs 10, 11, 12, 20, 25, 26, 34, and 35.

(iii) **Reliance on intrinsic coolant properties to minimize radionuclide releases**

The LFR-AS-200 relies on inherent design features based on the intrinsic properties of the lead coolant to maximize radionuclide retention in all plant operating states and LBEs. Multiple physical and functional barriers are credited to provide the confinement function in the LFR-AS-200:

1. **Fuel matrix (functional barrier):** key safety objectives of the LFR-AS-200 are the establishment of sufficient margins against fuel and clad melting in all LBEs and the demonstration of the extreme unlikelihood of generalized core melt scenarios⁵. When substantiated, these claims will allow crediting the oxide fuel matrix as an effective radionuclide retention barrier across the whole range of LBEs.
2. **Fuel cladding (first physical barrier):** the LFR-AS-200 employs a high-temperature, high-burnup, high-dpa-resistant, and liquid-metal corrosion-resistant titanium alloy and/or austenitic stainless-steel cladding material whose properties—combined with appropriate SAFDLs—ensure only aleatory fuel pin ruptures are tolerated during normal operation and anticipated event sequences. The extreme unlikelihood of generalized core melt scenarios and the absence of fast cladding-coolant chemical reactions (such as zircaloy oxidation in LWRs) ensure that its overall integrity is maintained across the whole range of LBEs. Fuel pin failure in the LFR-AS-200 is expected to occur by creep failure, which implies a localized fracture that stabilizes as soon as the outward stress exerted by the fission gas is released (noting that the

⁵ In a pool-type LFR, generalized core melt scenarios can be made physically impossible or extremely unlikely by design. This claim relies on the intrinsic properties of lead (notably, an extremely high boiling point—1749°C for pure lead, its high density, and its relative chemical inertness when in contact with water, air, and fuel) combined with appropriate design features that ensure loss-of-coolant accidents, prompt-criticality scenarios, and prolonged total loss of the core cooling function are made physically impossible or at least extremely improbable. Appropriate design features include the pool-type design, minimized annular gap between the reactor vessel and the reactor cavity (to exclude the possibility of uncovering the fuel or losing all heat removal paths in event sequences involving reactor vessel leakage), negative reactivity feedback and minimized head losses across the primary circuit (to facilitate natural circulation of the primary coolant).

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	22/71

primary coolant operates at atmospheric pressure). The fuel pin rupture event therefore induces the release of volatile fission products and the partial dissolution of non-volatile fission products into the primary coolant⁶, but the geometry of the fuel pellets and the overall physical function of the cladding are preserved.

3. **Fission product chemical retention in lead (functional barrier):** The chemical and thermodynamic properties of lead provide an inherent radionuclide retention function for the most radiologically significant volatile fission products released into the primary coolant following a fuel pin rupture [16]. For iodine, thermodynamic analyses indicate that molecular iodine (I_2) reacts with lead to form lead iodide (PbI_2), a stable compound with very low absolute vapor pressure above the lead coolant surface. The fraction of iodine transferred from the primary coolant to the cover gas—the release factor—is estimated at approximately 10^{-7} at normal operating temperatures ($\sim 530^\circ\text{C}$) and remains below 10^{-5} even at temperatures in the range of $750\text{--}800^\circ\text{C}$, implying that the primary coolant acts as an effective chemical trap for iodine across the full range of expected LBEs. For cesium, the strong thermodynamic affinity of elemental cesium for the liquid lead phase similarly suppresses its transfer to the cover gas; the estimated release factor is of the same order of magnitude as for iodine across the relevant temperature range. Under conservative assumptions of daily cover gas renewal over one year, these release factors rise by approximately two to three orders of magnitude but remain well below unity even at higher temperatures. It is further noted that the above estimates consider only the binary interactions of cesium and iodine with lead. More complex ternary or higher-order compounds—formed between lead, fission products, and other elements present in the primary system such as fuel cladding constituents or lead impurities—may also exist that could affect the transfer of these species to the cover gas.⁷
4. **Primary coolant at atmospheric pressure (functional barrier):** The primary coolant of the LFR-AS-200 has a considerable margin to its boiling point (1749°C for pure lead), operates at near-atmospheric pressure and is not susceptible to LWR-like loss-of-coolant accidents, thereby eliminating the risk of containment pressurization events involving the release of radioactive material into the containment. Moreover, lead does not undergo energetic coolant interactions of the type that could result in a high-energy release of radioactive material into the containment—such as sodium-air reactions or zircaloy oxidation in steam. For these reasons, a pressure-retaining containment structure is not needed for the LFR-AS-200, consistent with the functional containment approach described herein. It is recognized in this context that the above conclusion is contingent on the definition of quantitative functional containment performance criteria and on the final detailed design of the plant (as clarified by

⁶ It is worth noting here that lead has a relatively low affinity for oxygen compared to other liquid metal coolants, resulting in limited chemical interactions with the oxide fuel (cf. Vigier et al. [20]). Ingress of lead into the fuel pin is expected to cause only limited dissolution of actinide oxides, while non-gaseous fission-product phases may dissolve up to their respective solubility limits. However, no strong exergonic chemical reaction between the fuel and lead is expected. Consequently, the macroscopic integrity of the fuel pellets is expected to be largely preserved, owing to the thermodynamic stability of the fuel oxide phases in lead.

⁷ The ongoing LFR-AS-200 R&D program includes dedicated experimental studies on the behavior of key fission products (including but not limited to cesium and iodine) in lead to validate these estimates and characterize the influence of chemical speciation—including the possible formation of complex compounds—on retention and release behavior.

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	23/71

the NRC staff in the safety evaluation for the construction permit application for Kemmerer Unit 1 [17]) and that final NRC approval of the LFR-AS-200 functional containment design is therefore anticipated at the operating license application stage.

5. **Reactor coolant boundary (second physical barrier):** The reactor coolant boundary constitutes the second physical confinement barrier of the LFR-AS-200 and includes:

- The reactor vessel and reactor roof with its penetrations.
- The auxiliary systems connected to the reactor roof—such as the cover gas management system, the primary coolant chemistry control system, and the overpressure protection system—up to the first isolation valve or hydraulic seal.

The reactor roof assembly, including its components and rotating plugs, provides the upper confinement barrier of the reactor; leak-tightness of its penetrations is ensured by static or dynamic seals. The reactor vessel provides the lower confinement barrier. The integrity of the reactor coolant boundary during normal operation is maintained and monitored through cover gas pressure management and leakage detection. Performance criteria for the second confinement barrier are defined based on the relevant degradation mechanisms (notably thermal creep and liquid-metal corrosion) evaluated across the expected range of conditions during LBEs.

6. **Reactor cavity (third physical barrier):** The third physical confinement barrier of the LFR-AS-200 is formed by the reactor attic and adjacent rooms. It fulfills two of the three safety functions traditionally associated with a containment structure: radionuclide confinement and radiation shielding⁸. The third barrier is designed to retain gaseous and liquid radioactive substances against release to the atmosphere or to the ground in most LBEs⁹. It is also designed to protect the reactor and safety-significant SSCs from internal hazards (such as internal flooding and fire).

The main volume of the third barrier is the reactor attic, which houses the reactor cavity and the reactor roof access area. The reactor attic is separated from the operations hall above by a removable reinforced concrete slab that forms an integral part of the third barrier. The third barrier also encompasses adjacent rooms located at the same level as the reactor cavity, including cover gas management system rooms. All external walls of the third barrier rooms are constructed of reinforced concrete, ensuring structural robustness, leak-tightness, and radiation protection performance consistent with the applicable design requirements. The structural integrity of the third barrier under seismic events and other internal and external hazards is verified through a dedicated structural analysis model, ensuring that the confinement and shielding functions are maintained across the full range of LBEs.

⁸ The third traditional function—protection of the reactor and safety-significant SSCs against external hazards such as turbine missiles, flooding, or aircraft impact—is provided in the LFR-AS-200 by the reactor building, which encloses the reactor attic and serves as the external hazard shield, consistent with the basis for MHTGR-DC 16 and MHTGR-DC 71 in RG 1.232 [1].

⁹ Notable exceptions are high-energy line breaks (when the slightly activated secondary steam is discharged to the atmosphere to relieve the pressure in the attic) and fuel or component handling operations involving the removal of the concrete slab above the attic.

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	24/71

7. **Operations hall (functional barrier):** The operations hall above the reactor attic is credited as a functional barrier because it constitutes an effective hold-up volume for any gaseous and liquid radioactive materials released from the third barrier. The operations hall houses the main fuel handling equipment and the spent fuel pool and contributes to the confinement function during fuel handling operations and fuel route-related LBEs, when the third barrier is open. Its upper boundary is a reinforced concrete internal slab with a leak-tightness requirement. Above this slab, the external reinforced concrete building envelope protects the operations hall against aircraft impact.

As a consequence of the functional containment strategy of the LFR-AS-200, the ARDCs related to a traditional pressure-retaining containment structure (criteria 38–43 and 50-57) are not applicable to the LFR-AS-200 reactor design and have therefore been deleted (as in the case of the MHTGR-DCs).

5.2.6 Additional LFR-AS-200-Specific Criteria

The following new PDCs have been developed specifically for the LFR-AS-200:

- **PDC 70** (Reactor Vessel and Reactor System Structural Design Basis): This PDC is adapted from MHTGR-DC 70, which was introduced in RG 1.232 to ensure that the reactor vessel and reactor system integrity is preserved for passive heat removal and for the insertion of neutron absorbers. Despite the technological differences between LFRs and MHTGRs, the role of the reactor vessel is similar in both technologies: not to provide a pressure boundary but a circulation path for the primary coolant to transfer heat from the core to the heat sink and to preserve the core geometry for reactivity control.
- **PDC 71** (Reactor Building Design Basis): This PDC is adapted from MHTGR-DC 71, which has been accepted by the NRC as applicable to advanced non-LWR designs with functional containment [10]. The main safety function of the reactor building in the LFR-AS-200 is to protect safety-significant SSCs from the effects of external hazards so that they can ensure the fundamental safety functions. MHTGR-DC 72 (provisions for periodic reactor building inspection) is not required given the introduction of the generic PDC 6, which applies to all safety-significant SSCs.
- **PDC 72** (Primary Coolant and Cover Gas Chemistry Control): This PDC is adapted from SFR-DC 71, given its relevance to LFR technology. Chemical attack is replaced with liquid metal corrosion-erosion based on the relevant degradation mechanisms of stainless steel in an LFR. Air or moisture ingress as a result of a cover gas leak is not a significant risk in the LFR-AS-200 because the cover gas is slightly depressurized in normal operation and lead does not react energetically with air or water.
- **PDC 73** (Lead Heating System): This PDC is adapted from SFR-DC 72, given its relevance to a liquid-metal coolant with a high melting point.

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	25/71

6 LFR-AS-200 PRINCIPAL DESIGN CRITERIA

Table 1 presents the complete set of PDCs for *newcleo*'s LFR-AS-200 along with their basis. The PDCs are based on the corresponding ARDCs, SFR-DCs or MHTGR-DCs contained in RG 1.232 Appendices A, B and C, respectively [1]. The original ARDCs have been adapted to account for the specific features of the LFR-AS-200 design and in line with the LMP philosophy [2], [4]. Consistency with 10 CFR Part 53 [3] has furthermore been ensured. Some PDCs were borrowed from the SFR-DCs or MHTGR-DCs contained in RG 1.232 Appendices B and C, respectively [1]. Unless otherwise stated in the basis for a given PDC, modified text refers to the original ARDC.

The LFR-AS-200 PDCs are organized following the structure of the RG 1.232 ARDCs [1]:

- Section I – Overall Requirements (Criteria 1–6¹⁰)
- Section II – Multiple Barriers (Criteria 10–19)
- Section III – Reactivity Control (Criteria 20–29)¹¹
- Section IV – Fluid Systems (Criteria 30–46)
- Section V – Reactor Containment (Criteria 50–57)¹²
- Section VI – Fuel and Radioactivity Control (Criteria 60–64)
- Section VII – Additional PDC (Criteria 70–73)

Table 2 in APPENDIX A presents the LFR-AS-200 PDC set without markup.

¹⁰ PDC 6 is a new criterion introduced to replace and combine the RG 1.232 ARDCs related to monitoring, inspection, testing, and surveillance (see subsection 5.1.4).

¹¹ RG 1.232 deleted ARDC 27 as a standalone criterion, incorporating its content into ARDC 26; there is therefore no ARDC 27. This choice is confirmed for the LFR-AS-200 PDCs.

¹² Section V has been retained for consistency with the structure of the RG 1.232 ARDCs, although it is considered not applicable to the LFR-AS-200 given the latter's functional containment approach (see subsection 5.2.5).



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	26/71

Table 1 – Principal Design Criteria for the LFR-AS-200. Unless otherwise stated in the rationale for a given PDC, modified text refers to the original ARDC. Deleted text is marked in **red** with a strikethrough. Added text is marked in **blue** when the modification is related to the PDC adaptation to the LMP framework or Part 53 regulation and in **green** when the change reflects an adaptation to the LFR-AS-200 technology. Editorial changes are marked in orange.

Criterion	Title and Content	Basis for Change
I. Overall Requirements		
PDC 1	<i>Quality standards and records.</i> 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-significant function. A quality assurance program shall be established and implemented in order to provide adequate reasonable assurance that these structures, systems, and components will satisfactorily perform their safety-significant 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 for an appropriate period of the unit time.	Modified per NEI 21-07 Section 5.3.1, which requires that the Quality Assurance PDC address, at a minimum, the design, fabrication, construction, and testing quality standards applicable to safety-significant SSCs. The NEI 18-04 [4] methodology approach to special treatments is a graded one: the applicability of any category of special treatment to any SSC must be evaluated on a case-by-case basis and in the context of the SSC functions in the prevention and mitigation of applicable LBEs. The Quality Assurance PDC sets forth the requirements that these special treatments and programs have been developed and applied consistently with the NEI 18-04 methodology. PDC 1 is based on the example wording provided in NEI 21-07, Section 5.3.1, with the following modifications relative to ARDC 1: "Important to safety" is replaced with "safety-significant", consistent with the LMP framework (see Section 5.1.1 of this report). In the LMP framework, "safety-significant" encompasses both SR and NSRSS SSCs, both of which warrant special treatment and are thus within the scope of PDC 1. "Adequate assurance" is replaced with "reasonable assurance," consistent with the graded quality assurance approach of NEI 18-04 (cf. special treatment for NSRSS SSCs under § 53.460(b)(2)). "Throughout the life of the unit" is replaced with "for an appropriate period of time," consistent with the graded records retention approach of NEI 18-04. Since both SR and NSRSS SSCs warrant special treatment, PDC 1 defines the quality assurance requirements associated with the special treatment referred to in 10 CFR Part 53, § 53.460(b) [3].

SÉCURITÉ NATIONALE FR

NON PROTEGE



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	27/71

Criterion	Title and Content	Basis for Change
PDC 2	<i>Design bases for protection against natural phenomena.</i> 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 functions to be performed. Safety-related structures, systems, and components shall be protected against or shall be designed to withstand the effects of natural phenomena (e.g., earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches) and constructed hazards (e.g., dams, transportation routes, military and industrial facilities) considering an event severity up to the design-basis external hazard levels as determined under 10 CFR Part 53, § 53.510 without losing the capability to perform the safety functions identified under § 53.230.	Modified in accordance with § 53.415, "Protection against external hazards" [3]: <i>"Safety-related SSCs must be protected against or must be designed to withstand the effects of natural phenomena (e.g., earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches) and constructed hazards (e.g., dams, transportation routes, military and industrial facilities) considering an event severity up to the design-basis external hazard levels as determined under § 53.510 without losing the capability to perform the safety functions identified under § 53.230. Specific requirements for earthquake engineering are included in § 53.480."</i> <i>N.B.: the occurrences of the verb "must" are replaced with "shall" to reflect the original Regulatory Guide 1.232 tone. This change reflects the distinction between design criteria—which are applicant commitments expressed as design requirements—and regulatory requirements, which use "must" as the mandatory language of the CFR. This substitution is applied consistently throughout the LFR-AS-200 PDC set.</i>

SÉCURITÉ NATIONALE FR

NON PROTEGE



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	28/71

Criterion	Title and Content	Basis for Change
PDC 3	<i>Fire protection.</i> Safety-related and, where appropriate, non-safety-related but 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-facility, particularly in locations with safety-significant structures, systems, or components important to safety. Fire detection and fighting-fire suppression 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. Fire fighting-suppression systems shall be designed to ensure that their rupture or inadvertent operation does not significantly impair the safety-capability of these safety-significant structures, systems, and components to perform their safety functions.	Modified in accordance with 10 CFR Part 53, § 53.440, "Design requirements" [3]: (e)(1) <i>Safety-related SSCs and, where appropriate, NSRSS SSCs must be designed and located to minimize, consistent with other safety requirements in this part, the probability and effect of fires and explosions.</i> (2) <i>Noncombustible and fire-resistant materials must be used wherever practical throughout the facility, particularly in locations with SR and NSRSS SSCs.</i> (3) <i>Fire detection and fire suppression systems of appropriate capacity and capability must be provided and designed to minimize the adverse effects of fires on SR and NSRSS SSCs.</i> (4) <i>Fire suppression systems must be designed to ensure that their rupture or inadvertent operation does not significantly impair the ability of SR and NSRSS SSCs to perform their safety functions to satisfy § 53.230.</i>

SÉCURITÉ NATIONALE FR

NON PROTEGE

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	29/71

Criterion	Title and Content	Basis for Change
PDC 4	<i>Environmental and dynamic effects design bases.</i> 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 event sequences, unlikely event sequences, and postulated design basis accidents, <i>anticipating event sequences, unlikely event sequences, and postulated design basis accidents, commensurate with the safety-significance of the functions to be performed.</i> 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-facility. However, dynamic effects associated with postulated pipe ruptures in nuclear power-units-facilities 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. <i>Chemical interactions resulting from primary coolant leakage, such as liquid-metal corrosion of safety-significant structures, systems, and components, shall be appropriately considered in the design.</i>	Replaced “important to safety” with the LMP-compliant “safety-significant,” and adapted based on the corresponding SFR-DC: - The phrase “the environmental conditions associated with [...] design basis accidents” is added to ensure that the criterion would apply to all LFR LBEs, as suggested in NUREG-1368 [18]. The original version of the phrase (“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”) has been modified in accordance with the NEI 18-04 [4] definition of LBEs (including AOOs, DBEs and DBAs). - The phrase “commensurate with the safety-significance of the functions to be performed” is added at the end of the first sentence to clarify that environmental qualification of the SSCs follows the same graded approach introduced by NEI 18-04 as quality assurance. SR SSCs will have design requirements to protect against DBHLs as described in NEI 18-04 [4] and NEI 21-07 [8], while NSRSS SSCs will have capability targets less than the Design Basis Hazard Levels (DBHLs) but sufficient for DID adequacy as assessed by the NEI 18-04 IDP. - The term “nuclear power unit” is replaced with “nuclear facility” to reflect the possible multi-module nature of the plant. - A new sentence is added to ensure the designer considers the effects of lead leakage and associated chemical interactions (notably, liquid-metal corrosion) with safety-significant SSCs. Although the effect of lead is much more benign than sodium (which can interact exothermically with SSCs over much shorter timescales), the long-term integrity of safety-significant SSCs must still be guaranteed. The reference scenario here is reactor vessel leakage, which must not degrade the structural integrity of the reactor pit—a requirement addressed through an appropriate design feature.

SÉCURITÉ NATIONALE FR

NON PROTEGE

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	30/71

Criterion	Title and Content	Basis for Change
PDC 5	<i>Sharing of structures, systems, and components.</i> 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, <i>commensurate with the safety-significance of the functions to be performed</i>, including, in the event of an <i>anticipated event sequence, unlikely event sequence or design basis</i> accident in one unit, an orderly <i>the ability to achieve and maintain safe shutdown</i> and cooldown of the remaining units.	Replaced “important to safety” with the LMP-aligned version of “safety-significant,” and “accident” with “anticipated event sequence, unlikely event sequence or design basis accident” (in accordance with the Part 53 terminology). Added the phrase “commensurate with the safety-significance of the functions to be performed” to specify that the safety functions of the shared SSCs have different grades of safety significance, in line with the LMP framework. Added the phrase “anticipated event sequence, unlikely event sequence or design basis accident” (instead of the original, generic “accident”) for consistency with the LMP framework and Part 53 terminology. The rationale is that the term “cooldown” implies a temperature target inconsistent with high-temperature systems whose primary coolant melting point is significantly above ambient temperatures. RG 1.232 ARDC 26 [1] rationale and SECY-94-084 [19] describe a “safe shutdown” condition as: reactor subcriticality, decay heat removal, and radioactive materials containment. Given that the LFR-AS-200 is also a high-temperature system (with a primary coolant melting point of 327°C), the same rationale applies here. It is noted that “interventions such as refueling, inspection, and repair” (for which a subcritical state is required, as noted in the ARDC 26 rationale [1]) are designed to be performed above 327°C in the LFR-AS-200. It is furthermore noted that the NRC acceptance of a high-temperature safe shutdown condition in lieu of cold shutdown was conditional on the recommendation of “acceptable passive safety system performance and acceptable resolution of the issue of regulatory treatment of non-safety systems” [21]. For the LFR-AS-200 design, performance of the passive decay heat removal system is addressed by PDC 35, while the regulatory treatment of non-safety systems is addressed through use of NEI 18-04 [4] and RG 1.233 [2].

SÉCURITÉ NATIONALE FR

NON PROTEGE



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	31/71

Criterion	Title and Content	Basis for Change
PDC 6	<i>Monitoring, Inspection, Testing, Surveillance</i> Safety-significant structures, systems, and components shall be designed to permit appropriate monitoring, periodic inspection, functional testing, and/or surveillance of important areas and features, to ensure integrity and functional capability commensurate with the safety significance of the functions to be performed. Functional testing shall ensure the operability and performance of the system components, 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 and associated systems into operation for reactor shutdown.	New PDC (no generic monitoring, inspection, testing, and/or surveillance ARDC exists in RG 1.232). It replaces and combines ARDC 18, 32, 36, 37, 45 and 46 into a single criterion. It is noted in this context that point (1) of ARDCs 32, 37, and 46 (relating to the structural and leaktight integrity of important areas, features and components) and point (2) of ARDC 32 (material surveillance program for the reactor vessel) are considered special treatments—implementation-level requirements resulting from the SSC safety classification of NEI 18-04 [4], rather than being design criteria. Structural and leaktight integrity and material surveillance are therefore not explicitly mentioned in the new PDC 6. Monitoring, inspection, testing, and surveillance requirements will be established as special treatments in accordance with the NEI 18-04 IDP and will meet the functional performance intent of the corresponding ARDC. These requirements will be validated through the development of a dedicated program and supporting design process. This should ensure that monitoring, inspection, testing, and surveillance objectives (accessibility, inspectability, surveillance provisions, and inspection qualification requirements) are appropriately defined and can support U.S. regulatory requirements and expectations through a harmonized design approach. ARDC 18 is included because PDC 6 applies to all safety-significant SSCs; electric power systems may perform a safety-significant function in the LFR-AS-200 during anticipated event sequences and are therefore within the scope of this criterion. The terms "<i>important</i>" and "<i>important to safety</i>" in ARDC 18, 32, 36, and 45 are replaced with the LMP equivalent "<i>safety-significant</i>". It is noted that the term "<i>postulated accidents</i>" in ARDC 46 is the technology-inclusive version of "loss-of-coolant accidents" of GDC 46. In the LFR-AS-200, the structural and equipment cooling function is not credited in unlikely event sequences and design basis accidents (the LMP equivalent of postulated accidents) and is therefore not included in PDC 6.

SÉCURITÉ NATIONALE FR

NON PROTEGE



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	32/71

Criterion	Title and Content	Basis for Change
II. Multiple Barriers		
PDC 10	<i>Reactor design.</i> The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assureensure that specified acceptable fuel design limits and/or specified acceptable radionuclide release design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrencesevent sequences.	Modified in accordance with the LMP framework, specifically to allow the possibility of defining performance-based design limits (SARRDLs), rather than relying exclusively on traditional deterministic SAFDLs. The verb "assure" is replaced with "ensure" for grammatical consistency with the rest of the PDCs. "Anticipated operational occurrences" is replaced with the corresponding Part 53 term "anticipated event sequences".



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	33/71

Criterion	Title and Content	Basis for Change
PDC 11	<i>Reactor inherent protection.</i> 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 with sufficient overall negative reactivity feedback so that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate, in the power operating range, for a rapid increase in reactivity, to adequately address the additional heat generation via the available cooling provisions and to ensure that the specified acceptable fuel design limits and/or the specified acceptable radionuclide release design limits are met during anticipated event sequences, unlikely event sequences, and design basis accidents.	PDC 11 strengthens ARDC 11 by replacing the requirement that “<i>the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity</i>” with a requirement for “<i>sufficient overall negative reactivity feedback</i>”. GDC/ARDC 11 requires only that the prompt inherent nuclear feedback (primarily the Doppler coefficient) tends to compensate for a rapid increase in reactivity, while the delayed feedback contributions (e.g., positive coolant temperature or core expansion coefficient) and the net total feedback may remain positive. Requiring negative global reactivity feedback therefore ensures that the reactor is self-stabilizing not only on the prompt timescale but throughout the event. This design goal is relevant to the LFR-AS-200, where the large thermal inertia of the primary system means that delayed feedback terms play a significant role in determining the ultimate power level reached during a transient. The negative core-wide reactivity feedback of the LFR-AS-200 also contributes to PDC 26. The self-stabilizing character of the core ensures core cooling capability is maintained by limiting the thermal challenge to the core during power excursions, thereby contributing to the core cooling portion of ARDC 26 (3). Since the LFR-AS-200 PDC 11 assigns a wider safety role to the core inherent reactivity properties than the original ARDC 11—contributing not only to reactivity control but also to the core cooling function—a performance criterion consistent with the LMP framework is needed to capture this broader scope. The following phrase is therefore added at the end of the sentence: “<i>to adequately address the additional heat generation via the available cooling provisions and to ensure that the specified acceptable fuel design limits and/or the specified acceptable radionuclide release design limits are met during anticipated event sequences, unlikely event sequences, and design basis accidents.</i>”



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	34/71

Criterion	Title and Content	Basis for Change
PDC 12	<i>Suppression of reactor power oscillations.</i> 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 fuel design limits and/or specified acceptable radionuclide release design limits are not possible or can be reliably and readily detected and suppressed.	The reference to fuel design limits is extended to SARRDLs to allow compliance with the NEI 18-04 frequency–consequence target [4] (cf. basis for PDC 10). It is noted here that in the ARDC the word “<i>associated structures</i>” was added because items such as reflectors, which could be considered either outside or not part of the reactor core, may affect susceptibility of the core to power oscillations. No ex-core reflectors are employed in the LFR-AS-200 design; however, given that the concept of ‘core’ in a pool-type LFR is not yet defined unambiguously, the term is retained to cover any component outside of the fuel assembly/control rod system which may contribute to the neutronic stability of the core.
PDC 13	<i>Instrumentation and control.</i> Instrumentation shall be provided-designed to monitor variables and systems over their anticipated ranges for normal operation, for and during anticipated operational occurrences-event sequences, and for accident conditions-unlikely event sequences and design basis accidents, as appropriate to ensure adequate-reliable operation commensurate with their safety significance, including those-Due consideration shall be given to variables and systems that can affect the fission process, the integrity of the reactor core, safety-significant elements of the reactor coolant boundary, and the functional containment and its associated systems. Appropriate controls shall be provided-designed to maintain these variables and systems within prescribed operating ranges.	Replaced “<i>anticipated operational occurrences</i>” with “<i>anticipated event sequences</i>” and “<i>accident conditions</i>” with “<i>unlikely event sequences and design basis accidents</i>” (in accordance with the LMP framework and Part 53 terminology) and modified in accordance with the corresponding MHTGR-DC (to account for the functional containment strategy of the US LFR-AS-200). The term “<i>provided</i>” is replaced with “<i>designed</i>” to limit the criterion to the design of safety-significant Instrument and Control (I&C) SSCs. The term “<i>provided</i>” could imply activities other than design. The phrase “<i>safety-significant elements of</i>” is added before “<i>the reactor coolant boundary</i>” to account for the functional containment strategy adopted for the LFR-AS-200 design. Since the reactor coolant boundary of the LFR-AS-200 may include non-safety-significant cover gas piping and other auxiliary circuits, the original ARDC requirement is deemed inconsistent with the LMP philosophy. The term “<i>adequate safety</i>” is replaced with “<i>reliable operation commensurate with their safety significance</i>” to emphasize the graded safety approach introduced by NEI 18-04 [4].

SÉCURITÉ NATIONALE FR

NON PROTEGE

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	35/71

Criterion	Title and Content	Basis for Change
PDC 14	<i>Reactor coolant boundary.</i> The safety-significant elements of the reactor coolant boundary shall be designed, fabricated, and erected, and tested so as to have an extremely low probability address the risk of abnormal leakage, of rapidly propagating failure, of gross rupture, and of unacceptable ingress of secondary coolant or other fluids that may compromise the integrity of the fuel cladding or of the core structural materials.	Modified in the spirit of the corresponding MHTGR-DC, with consideration of the risk of water ingress (e.g., from steam generator tube leakage or rupture events) or of contamination of the cover gas system. The phrase "<i>safety-significant elements of the</i>" is added before "<i>reactor coolant boundary</i>" to account for the functional containment strategy adopted for the LFR-AS-200 design (cf. PDC 13). The phrase "<i>and tested</i>" is removed, since all inspection, testing and surveillance criteria are grouped into the new specific PDC 6. It is noted that the reference to the "<i>extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture</i>" is derived from GDC 14. In LWRs, reactor pressure vessel leakage must be excluded by design because of its consequences (i.e., a large LOCA that makes safety injection ineffective). The LFR-AS-200, however, is designed to tolerate reactor vessel leakage without loss of core cooling capability. The original GDC 14 requirement is therefore relaxed by replacing "<i>to have an extremely low probability</i>" with "<i>to address the risk of</i>".
PDC 15	<i>Reactor coolant system design.</i> The reactor 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 reactor coolant boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences event sequences.	See basis for the addition of "<i>safety-significant elements of the reactor coolant boundary</i>" in PDC 13. Replaced "<i>anticipated operational occurrences</i>" with "<i>anticipated event sequences</i>" (in accordance with the LMP framework and Part 53 terminology). The cover gas boundary is implicitly included in the criterion, as it is part of the reactor coolant boundary (cf. corresponding SFR-DC).
PDC 16	<i>Functional Ccontainment design.</i> A reactor functional containment, consisting of multiple barriers internal and/or external to the reactor coolant system, and associated systems shall be provided designed to establish an essentially leak-tight barrier against the uncontrolled control the release of radioactivity to the environment and to assure ensure that the safety-significant functional containment design conditions important to safety are not exceeded for as long as postulated accident design basis conditions require.	Modified in accordance with the corresponding MHTGR-DC, to account for the functional containment strategy of the US LFR-AS-200 (cf. basis for MHTGR-DC 16). The wording of MHTGR-DC 16 is slightly modified to account for the different definitions of reactor coolant boundary between the two technologies and in line with the LMP framework.

SÉCURITÉ NATIONALE FR

NON PROTEGE

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	36/71

Criterion	Title and Content	Basis for Change
PDC 17	<i>Electric power systems.</i> Electric power systems shall be provided when required to permit functioning of structures, systems, and components. The safety function for each Should any power system be required to perform safety-significant functions, it shall be designed to provide sufficient capacity and capability commensurate with the safety significance of the functions to be performed, to ensure that (1) that the design limits for the fission product barriers are not exceeded as a result of during anticipated operational occurrences event sequences 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 functions is provided.	Modified consistent with the approach accepted by the NRC for non-LWR designs [12]. References to the additional (independent) power system and to the single-failure criterion are removed, in accordance with the risk-informed performance-based methodology (see NEI 18-04 [4] and SRM-SECY-03-0047 [13]). While the LFR-AS-200 relies on a passive safety architecture, selected safety functions (notably, I&C, control room habitability and active actuation of passively operated systems) may require electric power to operate (ideally internally-stored direct current power). Moreover, active systems are credited in anticipated operational occurrences other than short-term loss of offsite power and therefore require a dedicated Principal Design Criterion. Replaced "anticipated operational occurrences" with "anticipated event sequences" (in accordance with Part 53 terminology).
PDC 18	<i>Inspection and testing of electric power systems.</i> 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.	Replaced by the new generic PDC 6 (which contains inspection and testing provisions for safety-significant SSCs).

SÉCURITÉ NATIONALE FR

NON PROTEGE



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	37/71

Criterion	Title and Content	Basis for Change
PDC 19	<i>Control room.</i> 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 during anticipated event sequences and unlikely event sequences. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions during anticipated event sequences and unlikely event sequences without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) as defined in § 50.253.020 for the duration of the accident event. Adequate habitability measures shall be provided to permit access and occupancy of the control room during normal operations, anticipated event sequences and under accident conditions unlikely event sequences. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot-safe shutdown of the reactor, including any necessary instrumentation and controls to maintain the unit in a safe shutdown condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.	Modified in accordance with the LMP framework (using Part 53 terminology). The reference to the definition of "<i>Total Effective Dose Equivalent</i>" is changed from 10 CFR 50.2 to 53.020. According to the rationale for MHTGR-DC 26 in RG 1.232, the requirement to bring the plant to cold shutdown is driven by the need to inspect and repair a plant following an accident. Given that in an LFR the primary coolant must be kept above 327°C to avoid solidification and that any applicable inspection and repair provisions will be defined consistently with this operational limitation, the second point in the last sentence has been deleted. See also the basis for replacing "<i>shutdown and cooldown</i>" with "<i>safe shutdown</i>" in PDC 5.
III. Reactivity Control		
PDC 20	<i>Protection system functions.</i> 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 and/or specified acceptable radionuclide release design limits are not exceeded as a result of anticipated operational occurrences event sequences and (2) to sense accident conditions associated with unlikely event sequences and design basis accidents and to initiate the operation of safety-significant systems and components important to safety.	Modified in accordance with the LMP framework, specifically to allow the possibility of defining performance-based design limits (Specified Acceptable Radionuclide Release Limits) rather than fuel design limits. See PDC 10 for the full rationale. The phrases "<i>important to safety</i>" and "<i>anticipated operational occurrences</i>" are replaced with "<i>safety-significant</i>" and "<i>anticipated event sequences</i>," respectively, in accordance with the LMP framework. Function (1) of this PDC covers anticipated event sequences, while function (2) refers to conditions associated with more severe events—i.e., unlikely event sequences and design basis accidents in LMP/Part 53 terminology. The phrase "<i>to sense accident conditions</i>" is therefore replaced with "<i>to sense conditions associated with unlikely event sequences and design basis accidents</i>."

SÉCURITÉ NATIONALE FR

NON PROTEGE

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	38/71

Criterion	Title and Content	Basis for Change
PDC 21	<i>Protection system reliability and testability.</i> 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-ensure that (1) no single failure results in loss sufficient reliability of the protection function and (2) that 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.	Modified in accordance with the LMP framework, where the single-failure criterion has been replaced by the full probabilistic assessment of the feature's reliability.
PDC 22	<i>Protection system independence.</i> The protection system shall be designed to assure-ensure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated design basis 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.	Modified in accordance with LMP terminology. It is noted that the protection system independence requirement of ARDC 22 has been accepted by the NRC as a Special Treatment in accordance with the NEI 18-04 methodology for non-LWR designs [12], rather than a functional requirement. It is furthermore observed that the validation of the adequacy of the protection system through the NEI 18-04 IDP (which involves a full reliability assessment of each SSC) ensures that the effect of natural phenomena and of transient or accident conditions are duly accounted for in the design of the system. It is therefore assumed that the Special Treatment implied in PDC 22 only applies to SR SSCs and thus to DBAs.
PDC 23	<i>Protection system failure modes.</i> 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, pressure, steam, water, and radiation) are experienced.	No adaptation needed (as in the case of the corresponding MHTGR-DC). The corresponding SFR-DC was modified to add "sodium and sodium reaction products" to the list of adverse environments, but this is not applicable to LFRs.

SÉCURITÉ NATIONALE FR

NON PROTEGE

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	39/71

Criterion	Title and Content	Basis for Change
PDC 24	<i>Separation of protection and control systems.</i> 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 ensure that safety is not significantly impaired.	No adaptation needed (as in the case of the corresponding SFR-DC and MHTGR-DC) other than a minor grammatical change for consistency with the rest of the criteria. <i>newcleo understands that the phrase “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” is meant to require that the protection system shall ensure its safety functions in case of unavailability of any single component or channel which is common with the control system. Specifically, the phrases ‘leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system’ should not be construed as implying that the single failure or removal from service shall leave the reliability and redundancy of the system unchanged (which is a logical impossibility), but rather that it should not prevent the protection system from fulfilling its functions.</i> <i>N.B.: reference to the “failure of any single control or protection system component or channel” should be construed as a requirement to avoid common-cause failure of both the control and protection system due to the failure of a shared component or channel, rather than an instance of the single-failure criterion. ARDC 24 points to the requirement of independence between the control and protection system, so that no single failure of a component or channel of either can lead to the failure of both. This requirement is therefore more about avoiding single-point vulnerabilities inducing the loss of the whole I&C function than about compliance with the single-failure criterion.</i>
PDC 25	<i>Protection system requirements for reactivity control malfunctions.</i> The protection system shall be designed to ensure that specified acceptable fuel design limits and/or specified acceptable radionuclide release design limits are not exceeded during any anticipated operational occurrence event sequences accounting for a single malfunction of the reactivity control systems.	Modified in accordance with the LMP framework, specifically to allow for the possibility of defining performance-based design limits (Specified Acceptable Radionuclide Release Limits) rather than fuel design limits. Cf. PDC 10 for the full rationale. The “single malfunction of the reactivity control systems” is understood to be a reference to, e.g., a spurious control rod withdrawal and not to the single-failure criterion.

SÉCURITÉ NATIONALE FR

NON PROTEGE



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	40/71

Criterion	Title and Content	Basis for Change
PDC 26	<i>Reactivity control systems.</i> A minimum of two The reactor core shall be designed with adequate reactivity control systems provisions or means shall provide to ensure, with appropriate reliability, independence and diversity commensurate with the safety significance of the event, that:	The LMP framework replaces prescriptive deterministic safety criteria with the RIPB treatment of DID, embodied in the NEI 18-04 IDP. The requirement of a minimum of two reactivity control systems or means, irrespective of the specific technology and of the overall reactivity control architecture, is not aligned with the NEI 18-04 philosophy. The text is therefore replaced by a more consistent DID approach, where the adequacy of the DID architecture is demonstrated via the application of the IDP formalism. The four points of ARDC 26 are collectively addressed by the reactivity control provisions of the LFR-AS-200, with a graded level of DID commensurate with the safety significance of the licensing-basis event to be addressed, consistent with the NEI 18-04 IDP. The phrase "<i>with appropriate reliability, independence and diversity commensurate with the safety significance of the event</i>" inserted at the end of this introductory sentence is therefore meant to apply to all four points of this PDC.
	(1) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that The specified acceptable fuel design limits for the fission product barriers and/or the specified acceptable radionuclide release design limits are not exceeded, and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences event sequences.	The term "<i>margin for malfunctions</i>" in ARDC 26 (1), which has a deterministic connotation, is replaced with the collective phrase "<i>reliability, independence and diversity commensurate with the safety significance of the event</i>" in the introductory sentence to this PDC, in line with the LMP framework. The phrase "<i>design limits for the fission product barriers</i>" is replaced with the advanced reactor variant "<i>specified acceptable fuel design limits and/or the specified acceptable radionuclide release design limits</i>" employed above (cf. PDC 20 and 25). The term "<i>anticipated operational occurrences</i>" is replaced with "<i>anticipated event sequences</i>" (in accordance with Part 53 terminology).

SÉCURITÉ NATIONALE FR

NON PROTEGE



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	41/71

Criterion	Title and Content	Basis for Change
	(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 is controlled to assure ensure that the specified acceptable fuel design limits for the fission product barriers and/or specified acceptable radionuclide release design limits are not exceeded.	The "independent and diverse" qualifier of ARDC 26(2) is not carried over prescriptively; instead, it is replaced with the global requirement of "<i>appropriate reliability, independence and diversity commensurate with the safety significance of the event</i>," contained in the introductory sentence to this PDC. The required level of independence and diversity for planned power changes will be established through the NEI 18-04 IDP. The phrase "<i>design limits for the fission product barriers</i>" is replaced with "<i>specified acceptable fuel design limits and/or specified acceptable radionuclide release design limits</i>" in agreement with the LMP framework.
	(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 an unlikely event sequence or design basis accident is provided.	PDC 11 establishes a requirement for negative core-wide reactivity feedback for the LFR-AS-200. The self-stabilizing character of the core ensures core cooling capability is maintained by limiting the thermal challenge to the core during power excursions, thereby contributing to the core cooling capability portion of ARDC 26 (3) by inherent features. The phrase "<i>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</i>" is therefore deleted. The second portion of ARDC 26 (3), maintaining a safe shutdown condition following a postulated accident, is ensured by the LFR-AS-200 engineered reactivity control systems. As noted in RG 1.232, this criterion allows for return to power during a postulated accident if sufficient heat removal capability exists. The term "<i>following a postulated accident</i>" in ARDC 26 (3) refers to the time when plant parameters are relatively stable, no additional means of mitigation are needed and margins to acceptance criteria are constant or increasing. While the inherent reactivity feedback of the LFR-AS-200 core guarantees a stable reactor state is reached, it does not ensure subcriticality, for which the insertion of movable poisons is required. The same engineered features credited for PDC 26 points (1), (2), and (4) are credited for this second portion of ARDC 26 (3). The term "<i>postulated accident</i>" is replaced with the LMP-consistent "<i>unlikely event sequence or design basis accident</i>".
	(4) A means for holding the reactor shutdown under conditions which allow for interventions such as fuel loading, inspection and repair shall be is provided.	ARDC 26 (4) is retained without modifications (other than a minor grammatical adaptation to the modified introductory sentence to this PDC).

SÉCURITÉ NATIONALE FR

NON PROTEGE



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	42/71

Criterion	Title and Content	Basis for Change
PDC 28	<i>Reactivity limits.</i> 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 unlikely event sequences and design basis accidents resulting from a reactivity increase can neither (1) result in damage to the safety-significant elements of the reactor 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.	Replaced the phrase "<i>the effects of postulated reactivity accidents</i>" with "<i>unlikely event sequences and design basis accidents resulting from a reactivity increase</i>" in line with the LMP framework and Part 53 terminology. While ARDC 28 is not fully aligned with the NEI 18-04 philosophy (the overall integrity of the reactor coolant boundary and vessel internals should be replaced with a more RIPB-consistent target, such as SARRDLs), it is deemed prudent to maintain this PDC, whose prescriptive character ensures the core design is constrained to ambitious safety goals (similar to the surrogate SAFDLs contemplated by the ASME/ANS PRA standard for advanced non-LWRs during the design development stage of the reactor). See basis for the addition of "safety-significant elements of the reactor coolant boundary" in PDC 13.
PDC 29	<i>Protection against anticipated operational occurrences.</i> 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.	Already covered in PDC 26.

SÉCURITÉ NATIONALE FR

NON PROTEGE



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	43/71

Criterion	Title and Content	Basis for Change
IV. Fluid Systems		
PDC 30	<i>Quality of reactor coolant boundary.</i> Components that are part of the reactor coolant boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical 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 reactor coolant leakage appreciable loss of lead level or leakage of radioactive cover gas outside the reactor coolant boundary. Means shall be provided for detecting ingress of water, steam, or other fluids into the reactor coolant boundary.	The phrase "<i>the highest quality standards</i>" is replaced with "<i>quality standards commensurate with the safety significance of the functions to be performed</i>", in line with the LMP philosophy (cf. PDC 1). The detection and identification of the source of reactor coolant leakage is replaced with the detection of loss of lead level or of leakage of cover gas. For a pool-type LFR operating at atmospheric pressure, the concept of "identifying the location of the source of leakage" that underpins ARDC 30's requirement to detect and identify the location of the source of reactor coolant leakage does not translate. A breach of the reactor coolant boundary in an LFR does not produce a loss-of-coolant accident in the LWR sense; it produces a drop in lead level and a release of contaminated fluid into containment. The relevant detection parameter is therefore lead level, not a leakage source location. In accordance with the corresponding MHTGR-DC, a reference to detecting the ingress of secondary coolant (water or steam) is added, to mitigate the consequences of steam generator tube leakage or rupture events. The phrase 'or other fluids' denotes, e.g., the risk of ingress of gaseous species into the cover gas, other than those expected in normal operation (argon, oxygen, and hydrogen).

SÉCURITÉ NATIONALE FR

NON PROTEGE

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	44/71

Criterion	Title and Content	Basis for Change
PDC 31	<i>Fracture prevention of reactor coolant boundary.</i> Safety-significant elements of the reactor coolant boundary shall be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions, anticipated event sequences, unlikely event sequences, and design basis accidents (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 anticipated event sequences, unlikely event sequences, and design basis accidents and the uncertainties in determining (1) material properties, (2) the effects of irradiation and coolant composition, including contaminants impurities and reaction products oxygen content, on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws.	See basis for the addition of "safety-significant elements of the reactor coolant boundary" in PDC 13. The phrase "postulated accident conditions" is replaced with the LMP-consistent "anticipated event sequences, unlikely event sequences, and design basis accidents". Modified to account for the specificities of LFR technology (i.e., the effect of coolant chemistry on liquid metal corrosion of the reactor vessel) and in line with the LMP philosophy and Part 53 terminology ("postulated accident" is replaced with "unlikely event sequence").
PDC 32	<i>Inspection of reactor coolant boundary.</i> Components that are part of the reactor 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.	Replaced by the new generic PDC 6 (which contains inspection and testing provisions for safety-significant SSCs).
PDC 33	<i>Reactor coolant inventory maintenance.</i> Not applicable to the LFR-AS-200.	As in the case of MHTGRs (though for different physical reasons), the LFR-AS-200 does not require reactor coolant inventory maintenance for reactor vessel leaks to meet the SARRDLs. Therefore, ARDC 33 is not applicable to the LFR-AS-200 design.

SÉCURITÉ NATIONALE FR

NON PROTEGE



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	45/71

Criterion	Title and Content	Basis for Change
PDC 34	<i>Residual heat removal.</i> A system to remove residual heat shall be provided. For normal operations and anticipated operational occurrences event sequences, the system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core to the ultimate heat sink at a rate such that specified acceptable fuel design limits or specified acceptable radionuclide release design limits and the design conditions of the reactor 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.	Replaced "anticipated operational occurrences" with "anticipated event sequences" in accordance with Part 53 terminology. The phrase "from the reactor core" is replaced with "to the ultimate heat sink" to emphasize the fact that in the LFR-AS-200 core cooling is always accomplished by the same system (the primary coolant system) in either forced or natural circulation for all plant states, while different systems are employed to transfer fission product heat from the primary circuit to the heat sink. Modified in accordance with the LMP framework, specifically to allow the possibility of defining performance-based design limits (SARRDLs) rather than fuel design limits. See PDC 10 for the full rationale. The reference to the single-failure criterion is deleted, in accordance with Part 53 and the risk-informed performance-based approach. <i>N.B.: it is assumed that a single system provides residual heat removal at shutdown (normal operation) and during anticipated event sequences.</i>



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	46/71

PDC 35	Emergency core cooling system <i>Passive Decay Heat Removal.</i> A system <i>Passive means to assure sufficient core cooling during postulated accidents and to remove residual decay heat following postulated accidents during unlikely event sequences and design basis 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 the specified acceptable fuel design limits and/or specified acceptable radionuclide release design limits are not exceeded.</i>	The separation of residual heat removal (PDC 34) and passive decay heat removal (PDC 35) design criteria follows the defense-in-depth requirement of § 53.250(c) of 10 CFR Part 53 [3] that no single engineered design feature can be exclusively relied upon to address the range of licensing basis events other than design basis accidents. The term "<i>decay heat removal</i>" is used in the title of this PDC to distinguish the cooling function in unlikely event sequences and design basis accidents from that in normal operation and anticipated event sequences, covered by PDC 34 ("<i>Residual heat removal</i>"). This choice reflects the historical use, common in the Generation IV community, of the term 'decay heat removal systems' to refer to the safety function in accident scenarios. A reference to the passive nature of the decay heat removal systems is added in the title, in line with the Generation IV goals of the LFR-AS-200. The term "<i>A system</i>" is replaced with "<i>Passive means</i>" to account for the possibility of relying on inherent cooling features or on passive engineered systems. This choice also addresses the possibility that different means of decay heat removal are relied upon in unlikely event sequences and in design basis accidents. The phrase "<i>to assure sufficient core cooling during postulated accidents</i>" is not applicable—since the LFR-AS-200 is designed to transfer decay heat from the core to a secondary means of cooling in natural circulation in all accident conditions (unlikely event sequences and design basis accidents) and therefore does not require a dedicated system—and is therefore omitted. The phrase "<i>postulated accidents</i>" is replaced with "<i>unlikely event sequences and design basis accidents</i>" and the phrase "<i>fuel and clad damage that could interfere with continued effective core cooling is prevented</i>" with "<i>the specified acceptable fuel design limits and/or specified acceptable radionuclide release design limits are not exceeded</i>", in line with the LMP philosophy. The two sentences from the original ARDC have been merged to avoid redundant content. It is noted that the original ARDC implies that core cooling must be ensured <i>during</i> the accident transient, while residual heat removal must be ensured <i>during and after</i> the transient. This subtle distinction may be related to the difference between hot and cold shutdown or cooldown (typical of LWRs), which requires an additional means to cool the reactor beyond the termination of the accident transient. Since this distinction is not applicable to the LFR-AS-200
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SÉCURITÉ NATIONALE FR

NON PROTEGE

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	47/71

Criterion	Title and Content	Basis for Change
		technology (cf. basis for PDCs 5 and 19), the use of the phrase "during unlikely event sequences and design basis accidents" covers the entire duration of the event, including the stabilized safe shutdown state.
PDC 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.	Replaced by the new generic PDC 6 (which contains inspection and testing provisions for safety-significant SSCs).
PDC 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.	Replaced by the new generic PDC 6 (which contains inspection and testing provisions for safety-significant SSCs).
PDC 38	Containment heat removal. Not applicable to the LFR-AS-200.	This criterion is not applicable to the LFR-AS-200, whose design does not have a "pressure retaining reactor containment structure" but instead relies on a multibarrier functional containment configuration to control the release of radionuclides (as in the case of MHTGR). See the rationale for MHTGR-DC 16.
PDC 39	Inspection of containment heat removal system. Not applicable to the LFR-AS-200.	This criterion is not applicable to the LFR-AS-200, whose design does not have a "pressure retaining reactor containment structure" but instead relies on a multibarrier functional containment configuration to control the release of radionuclides (as in the case of MHTGR). See the rationale for MHTGR-DC 16.
PDC 40	Testing of containment heat removal system. Not applicable to the LFR-AS-200.	This criterion is not applicable to the LFR-AS-200, whose design does not have a "pressure retaining reactor containment structure" but instead relies on a multibarrier functional containment configuration to control the release of radionuclides (as in the case of MHTGR). See the rationale for MHTGR-DC 16.

SÉCURITÉ NATIONALE FR

NON PROTEGE



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	48/71

Criterion	Title and Content	Basis for Change
PDC-41	Containment atmosphere cleanup. Not applicable to the LFR-AS-200.	This criterion is not applicable to the LFR-AS-200, whose design does not have a “pressure retaining reactor containment structure” but instead relies on a multibarrier functional containment configuration to control the release of radionuclides (as in the case of MHTGR). See the rationale for MHTGR-DC 16.
PDC-42	Inspection of containment atmosphere cleanup systems. Not applicable to the LFR-AS-200.	This criterion is not applicable to the LFR-AS-200, whose design does not have a “pressure retaining reactor containment structure” but instead relies on a multibarrier functional containment configuration to control the release of radionuclides (as in the case of MHTGR). See the rationale for MHTGR-DC 16.
PDC-43	Testing of containment atmosphere cleanup systems. Not applicable to the LFR-AS-200.	This criterion is not applicable to the LFR-AS-200, whose design does not have a “pressure retaining reactor containment structure” but instead relies on a multibarrier functional containment configuration to control the release of radionuclides (as in the case of MHTGR). See the rationale for MHTGR-DC 16.

SÉCURITÉ NATIONALE FR

NON PROTEGE

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	49/71

Criterion	Title and Content	Basis for Change
PDC 44	<i>Structural and equipment cooling.</i> In addition to the heat rejection capability of the passive-residual heat removal system, systems-means to transfer heat from safety-significant structures, systems, and components important to safety to an ultimate heat sink shall be provided-designed, as necessary, to transfer the combined heat load of these safety-significant structures, systems, and components under-during normal operating-operation and accident conditions-anticipated event sequences. 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.	The prefix "<i>In addition to the heat rejection capability of the passive residual heat removal system</i>" from the corresponding MHTGR-DC is adopted here, to account for the fact that the residual heat removal system (PDC 34) of the LFR-AS-200 provides residual heat removal during normal operation and anticipated event sequences, but does not necessarily cover cooling of other safety-significant SSCs (such as the reactor cavity or the reactor vessel supports). The term "<i>passive</i>" is removed from "<i>passive residual heat removal system</i>" in line with the different terminology employed in PDC 34 compared to MHTGR-DC 34. The term "<i>systems</i>" is replaced with "<i>means</i>" to account for the possibility of relying on inherent cooling features (such as natural convection or radiative heat transfer). The phrases "<i>important to safety</i>" and "<i>accident conditions</i>" are replaced by "<i>safety-significant</i>" and "<i>anticipated event sequences</i>" in accordance with the LMP framework and Part 53 terminology, noting that structural and equipment cooling in DBEs and DBAs is not strictly required to comply with the NEI 18-04 targets. This latter claim will be verified by full application of the NEI 18-04 process. The term "<i>provided</i>" is replaced with "<i>designed</i>" to limit the criterion to the design of structural and equipment cooling means (cf. basis for PDC 13). The last paragraph on suitable redundancy is deleted following the elimination of the single-failure criterion in favor of the full probabilistic reliability assessment introduced by the LMP framework. This criterion does not address passive decay heat removal, which is a function defined for unlikely event sequences and DBAs and is covered by PDC 35. While the final design of the passive decay heat removal system of the LFR-AS-200 may provide reactor cavity cooling during normal operation and anticipated event sequences, a separation between PDC 35 and 44 is maintained since the two criteria apply to different event categories.

SÉCURITÉ NATIONALE FR

NON PROTEGE

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	50/71

Criterion	Title and Content	Basis for Change
PDC-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.	Replaced by the new generic PDC 6 (which contains inspection and testing provisions for safety-significant SSCs).
PDC-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.	Replaced by the new generic PDC 6 (which contains inspection and testing provisions for safety-significant SSCs).
V. Reactor Containment		
PDC-50	Containment design basis. The containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from postulated accidents. This margin shall reflect consideration of (1) the effects of potential energy sources that have not been included in the determination of the peak conditions, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters. Not applicable to the LFR-AS-200.	This criterion is not applicable to the LFR-AS-200, whose design does not have a “pressure retaining reactor containment structure” but instead relies on a multibarrier functional containment configuration to control the release of radionuclides (as in the case of MHTGR). See the rationale for MHTGR-DC 16.

SÉCURITÉ NATIONALE FR

NON PROTEGE

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	51/71

Criterion	Title and Content	Basis for Change
PDC-51	Fracture prevention of containment pressure boundary. The boundary of the containment structure shall be designed with sufficient margin to ensure that, under operating, maintenance, testing, and postulated accident conditions, (1) its materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary materials during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws. Not applicable to the LFR-AS-200.	This criterion is not applicable to the LFR-AS-200, whose design does not have a “pressure retaining reactor containment structure” but instead relies on a multibarrier functional containment configuration to control the release of radionuclides (as in the case of MHTGR). See the rationale for MHTGR-DC 16.
PDC-52	Capability for containment leakage rate testing. The containment structure and other equipment that may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure. Not applicable to the LFR-AS-200.	This criterion is not applicable to the LFR-AS-200, whose design does not have a “pressure retaining reactor containment structure” but instead relies on a multibarrier functional containment configuration to control the release of radionuclides (as in the case of MHTGR). See the rationale for MHTGR-DC 16.
PDC-53	Provisions for containment testing and inspection. The containment structure shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leak-tightness of penetrations that have resilient seals and expansion bellows. Not applicable to the LFR-AS-200.	This criterion is not applicable to the LFR-AS-200, whose design does not have a “pressure retaining reactor containment structure” but instead relies on a multibarrier functional containment configuration to control the release of radionuclides (as in the case of MHTGR). See the rationale for MHTGR-DC 16.
PDC-54	Piping systems penetrating containment.	This criterion is not applicable to the LFR-AS-200, whose design does not have a “pressure retaining reactor containment structure” but instead relies on a multibarrier functional containment configuration to control the release of radionuclides (as in the case of MHTGR). See the rationale for MHTGR-DC 16.

SÉCURITÉ NATIONALE FR

NON PROTEGE

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	52/71

Criterion	Title and Content	Basis for Change
	Piping systems penetrating the containment structure shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities that reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with the capability to verify, by testing, the operational readiness of any isolation valves and associated apparatus periodically and to confirm that valve leakage is within acceptable limits. Not applicable to the LFR-AS-200.	
PDC-55	Reactor coolant boundary penetrating containment. Each line that is part of the reactor coolant boundary and that penetrates the containment structure shall be provided with containment isolation valves, as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.	This criterion is not applicable to the LFR-AS-200, whose design does not have a “pressure retaining reactor containment structure” but instead relies on a multibarrier functional containment configuration to control the release of radionuclides (as in the case of MHTGR). See the rationale for MHTGR-DC 16.

SÉCURITÉ NATIONALE FR

NON PROTEGE

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	53/71

Criterion	Title and Content	Basis for Change
	Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to ensure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing; additional provisions for inservice inspection; protection against more severe natural phenomena; and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs. Not applicable to the LFR-AS-200.	
PDC-56	Containment isolation. Each line that connects directly to the containment atmosphere and penetrates the containment structure shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety. Not applicable to the LFR-AS-200.	This criterion is not applicable to the LFR-AS-200, whose design does not have a "pressure retaining reactor containment structure" but instead relies on a multibarrier functional containment configuration to control the release of radionuclides (as in the case of MHTGR). See the rationale for MHTGR-DC 16.

SÉCURITÉ NATIONALE FR

NON PROTEGE



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	54/71

Criterion	Title and Content	Basis for Change
PDC 57	Closed system isolation valves. Each line that penetrates the containment structure and is neither part of the reactor coolant boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve, unless it can be demonstrated that the containment safety function can be met without an isolation valve and assuming failure of a single active component. The isolation valve, if required, shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve. Not applicable to the LFR-AS-200.	This criterion is not applicable to the LFR-AS-200, whose design does not have a "pressure retaining reactor containment structure" but instead relies on a multibarrier functional containment configuration to control the release of radionuclides (as in the case of MHTGR). See the rationale for MHTGR-DC 16.
VI. Fuel and Radioactivity Control		
PDC 60	<i>Control of releases of radioactive materials to the environment.</i> 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 and anticipated operational occurrences event sequences. Sufficient holdup capacity (or other means necessary) shall be provided for retention (or other controlling measures) 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.	The phrase "<i>anticipated operational occurrences</i>" is replaced with "<i>anticipated event sequences</i>" in line with Part 53 terminology. The phrases "<i>or other means necessary</i>" and "<i>or other controlling measures</i>" are added to allow for alternative design solutions beyond "<i>sufficient holdup capacity</i>" and "<i>retention</i>" to effectively control the release of radioactive materials to the environment when unfavorable site environmental conditions are expected, consistent with the approach accepted by the NRC for non-LWR designs [12].

SÉCURITÉ NATIONALE FR

NON PROTEGE

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	55/71

Criterion	Title and Content	Basis for Change
PDC 61	<i>Fuel storage and handling and radioactivity control.</i> The fuel storage and handling, radioactive waste, and other systems that may contain radioactivity shall be designed to ensure adequate safety, commensurate with the safety significance of the functions to be performed, under during normal operation, and postulated accident conditions anticipated event sequences, unlikely event sequences, and design basis accidents, as appropriate. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) (2) with appropriate containment, confinement, and filtering systems, (4) (3) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal commensurate with the safety significance of the functions to be performed, and (5) (4) to prevent significant reduction in fuel storage cooling under accident conditions during anticipated event sequences, unlikely event sequences, and design basis accidents, as appropriate.	As mentioned by RG 1.232, some advanced designs may use dry fuel storage that incorporates cooling jackets that can be liquid cooled or air cooled to remove heat. This modification to this GDC allows for both liquid and air cooling of the dry fuel storage containers. The phrase "<i>commensurate with the safety significance of the functions to be performed</i>" is added to account for the graded approach in defining the safety classification of the SSCs. The phrase "<i>as appropriate</i>" is added to the end of the first sentence to reiterate that not all fuel storage, handling and radwaste management SSCs may be required across the whole range of LBEs. The phrase "<i>accident conditions</i>" is replaced with "<i>anticipated event sequences, unlikely event sequences, and design basis accidents</i>" in line with the LMP framework and Part 53 terminology. Since periodic inspection and testing is covered by PDC 6, the phrases "<i>with a capability to permit appropriate periodic inspection and testing of components important to safety</i>" and "<i>and testability</i>" are deleted. The phrase "<i>with a residual heat removal capability having reliability that reflects the importance to safety of decay heat and other residual heat removal</i>" is replaced with the LMP-aligned "<i>with a residual heat removal capability having reliability commensurate with the safety significance of the functions to be performed</i>".
PDC 62	<i>Prevention of criticality in fuel storage and handling.</i> Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.	No modification required (as in the case of the corresponding SFR-DC and MHTGR-DC)
PDC 63	<i>Monitoring fuel and waste storage.</i> 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.	No modification required (as in the case of the corresponding SFR-DC and MHTGR-DC)

SÉCURITÉ NATIONALE FR

NON PROTEGE



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	56/71

Criterion	Title and Content	Basis for Change
PDC 64	<i>Monitoring radioactivity releases.</i> Means shall be provided for monitoring the reactor containment building atmosphere, effluent discharge paths, and plant environs for radioactivity that may be released from during normal operations, including anticipated operational occurrences event sequences, and from postulated accidents unlikely event sequences.	The phrase "reactor containment atmosphere" is replaced with "reactor building atmosphere" to account for the LFR-AS-200 functional containment strategy (as in the corresponding MHTGR-DC). The wording of the event categories is modified in accordance with the LMP framework and Part 53 terminology. Similar to the precedent design criteria, the term "<i>postulated accidents</i>" is replaced with "<i>unlikely event sequences</i>" because operator action is not credited in DBAs (monitoring is required only for DID and can therefore be accomplished by non-safety-related equipment).



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	57/71

Criterion	Title and Content	Basis for Change
VII. Additional PDC		
PDC 70	<i>Reactor vessel and reactor system structural design basis.</i> The design of the reactor vessel and reactor system <i>its internals</i> shall be such that their integrity is maintained during postulated accidents <i>unlikely event sequences and design basis accidents</i> (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.	Based on MHTGR-DC 70. RG 1.232 introduced this new MHTGR design-specific PDC 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. In spite of the technological differences between LFRs and MHTGRs, the role of the reactor vessel is similar: provide a circulation path for the primary coolant to transfer heat from the core to the heat sink and preserve the core geometry for reactivity control. The new PDC is therefore adopted for the LFR-AS-200 with some adaptation. The phrase "<i>and reactor system</i>" is replaced with "<i>and its internals</i>", since the primary coolant is contained entirely within the reactor vessel (the only external piping is for the cover gas, which serves no function in terms of core cooling or reactivity control). The phrase "<i>postulated accidents</i>" is replaced with the LMP-aligned "<i>unlikely event sequences and design basis accidents</i>" (in anticipated event sequences the design criterion is operational rather than safety-driven, with a requirement to comply with the normal operation limits so as to allow for fast reactor restart once the event has been terminated).



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	58/71

Criterion	Title and Content	Basis for Change
PDC 71	Reactor building design basis. The design of the reactor building shall be designed such that, during postulated accidents, it to structurally protects the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink and for reactivity control during unlikely event sequences and design basis accidents and provides a pathway for the release of reactor helium from the building in the event of depressurization accidents.	Based on MHTGR-DC 71. MHTGR-DC 71 has been accepted by the NRC as applicable to advanced non-LWR designs with functional containment, including sodium-cooled fast reactors [10]. The same rationale can be used for the LFR-AS-200, with some modifications to the original MHTGR-DC: "Postulated accidents" is replaced with "<i>unlikely event sequences and design basis accidents</i>" in accordance with the LMP framework and Part 53 terminology. - The phrase "<i>and provides a pathway for the release of reactor helium from the building in the event of depressurization accidents</i>" is replaced with "<i>and for reactivity control</i>" to reflect the difference in technology. The main safety function of the reactor building in the LFR-AS-200 is to protect safety-significant SSCs from the effect of external hazards so that they can ensure the fundamental safety functions. N.B.: MHTGR-DC 72 ("<i>Provisions for periodic reactor building inspection</i>") is not required given the introduction of the new generic PDC 6 ("<i>Inspection, Testing, Surveillance</i>"), which applies to all SSCs.



DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	59/71

Criterion	Title and Content	Basis for Change
PDC 72	Primary coolant and cover gas purity-chemistry control. Systems shall be provided as necessary to maintain the purity of primary coolant sodium-lead and cover gas within specified design limits. These limits shall be based on consideration of (1) chemical-attack-liquid metal corrosion-erosion, (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.	Based on SFR-DC-71. SFR-DC 71 is adopted here, given its relevance to the LFR technology. Minor modifications are made to adapt to the peculiarities of the specific reactor technology. Specifically, the phrase "<i>chemical attack</i>" is replaced with "<i>liquid metal corrosion-erosion</i>" since the relevant degradation mechanisms of stainless steel in an LFR are oxidation and dissolution (the two main liquid-metal corrosion mechanisms in lead) and erosion. Air or moisture ingress as a result of a leak of cover gas is not a significant risk in the LFR-AS-200, because (i) the cover gas is slightly depressurized in normal operation and (ii) lead (unlike sodium) does not react energetically with air or water, hence the presence of those two fluids in the cover gas would at most result in the oxidation of lead and its impurities at the surface of the primary coolant pool. Formation of PbO or other oxides may induce an increased risk of fouling in the primary circuit in the long term, however this risk is managed by a specific design limit defined for the primary coolant and cover gas chemistry control system. <i>N.B.: the primary coolant and cover gas chemistry control system is not necessarily used in normal operation at full power. Given the slow kinetics of the liquid-metal corrosion phenomena (hence the low expected corrosion rates), the use of this system could be limited to shutdown states.</i>

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	60/71

Criterion	Title and Content	Basis for Change
PDC 73	Sodium-Lead heating systems. A lead Hheating systems shall be provided, when necessary, for systems and components that are important to safety, and that contain or could be required to contain sodium the primary coolant system. These This heating systems and their its controls shall be appropriately designed to ensure that the bulk temperature distribution and rate of change of temperature in systems and components containing sodium of the primary coolant are is 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 important to safety. These limits shall be based on consideration of the effect of primary coolant solidification on the integrity of safety-significant structures, systems, and components, specifically during very low power conditions within normal operation and anticipated event sequences.	Based on SFR-DC 72. SFR-DC 72 is adopted here given its relevance to LFR-AS-200 design. Some modifications are made to adapt to the peculiarities of the specific reactor technology. The need for a lead heating system may be limited to reactor states characterized by very low decay heat, which are expected to be infrequent (potentially limited to shutdown states with mostly fresh fuel or abnormally long outages). The phrase "when necessary" is therefore added to emphasize the non-permanent character of the system. The reference to multiple lead heating systems is removed, given the absence of primary coolant piping outside the reactor vessel (and of an intermediate system as in the case of an SFR). The phrase "systems and components that are important to safety, and that contain or could be required to contain sodium" is deleted for the same reason (only the primary circuit contains lead coolant). The phrase "to ensure that the temperature distribution [...] are maintained within design limits" is replaced with "to ensure that the bulk temperature of the primary coolant is maintained within design limits", since lead solidification in a pool-type LFR during normal operation and anticipated event sequences is a slow and globally homogeneous phenomenon (given the very large thermal inertia of the primary coolant system). The phrase "assuming a single failure" is deleted in accordance with the LMP philosophy, where the single-failure criterion has been replaced with a full probabilistic reliability assessment. Plugging of cover gas lines is not a significant risk in the LFR-AS-200 design, especially considering the very low lead vapor pressure in the cover gas in normal operation. The sentence "If plugging of any cover gas line [...] shall be considered important to safety" is accordingly deleted. The last sentence is added to specify that the lead temperature limits will be defined to ensure that the integrity of safety-significant SSCs (notably the reactor vessel and safety-significant vessel internals) is preserved during normal operation and anticipated event sequences, specifically at very low power.

SÉCURITÉ NATIONALE FR

NON PROTEGE

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	61/71

7 CONCLUSION

This topical report described the approach taken by *newcleo* to develop an initial complete set of PDC for the LFR-AS-200 reactor technology. This approach started with an assessment of the ARDC and design criteria for the SFR and MHTGR technologies endorsed in RG 1.232. Once applicable design criteria were screened, adaptations were proposed to ensure the existing criteria are aligned with more recent regulatory approaches such as LMP and the terminology now codified in 10 CFR Part 53 [3]. Similar to existing precedent, *newcleo* intends to adopt certain safety design approaches such as functional containment and its use of SARRDL as a key figure of merit in evaluating reactor safety, with specific PDC developed from the best available resources on those topics. Where appropriate, PDC were modified to account for the unique features of the LFR-AS-200 technology.

As stated earlier, the LFR-AS-200 reactor technology is being adapted from its European development history for deployment in the U.S. market. These PDC will be used to guide the development of the technology for licensing under the Part 53 [3] regulatory framework for commercial deployments. Through detailed design it is possible that some PDC may warrant revision, especially as the risk-informed, performance-based approaches found in LMP and other advanced reactor regulatory guidance are implemented and risk-insights developed for the technology. Future amendments to this topical report will be made when needed to account for this development and incorporated into future applicants' licensing bases and/or applications as needed.

This topical report presents an initial complete set of PDC for the LFR-AS-200 advanced reactor, developed by *newcleo* for use by itself and future applicants seeking licenses, certifications, permits, and other approval types under 10 CFR Part 53 [3] that require design criteria to be established. The PDC set was developed starting from the ARDCs of RG 1.232 [1], adapted to align with the RIPB framework of the LMP as endorsed by RG 1.233 [2] and codified in 10 CFR Part 53 [3].

The PDC set comprises thirty-seven active criteria organized across seven sections consistent with the structure of RG 1.232. Twenty-three criteria were deleted as not applicable to the LFR-AS-200 design or as redundant with other criteria; fifteen criteria were retained as structural placeholders but marked not applicable due to the LFR-AS-200 functional containment approach. Key technology-specific adaptations include:

- The replacement of the prescriptive single-failure criterion with a probabilistic reliability criterion consistent with the NEI 18-04 IDP, applied across all applicable PDCs.
- The consolidation of six inspection, testing, and surveillance ARDCs (18, 32, 36, 37, 45, 46) into a single generic criterion (PDC 6), consistent with the approach accepted by the NRC for non-LWR designs [10].
- The strengthening of ARDC 11 to require negative global reactivity feedback (PDC 11), which contributes both to reactivity control and to the core cooling function addressed by ARDC 26 (3).
- The restructuring of ARDC 26 into a four-point criterion with a graded DID introductory sentence, replacing the prescriptive two-system requirement with a RIPB approach demonstrated through the NEI 18-04 IDP.

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	62/71

- The deletion of ARDC 33 (Reactor Coolant Inventory Maintenance) and ARDC 35 (Emergency Core Cooling System) as not applicable to the pool-type LFR-AS-200 design, replaced by PDC 35 (Passive Decay Heat Removal).
- The adoption of a functional containment strategy in lieu of a pressure-retaining containment structure, with SARRDLs as the key performance metric, consistent with the approach accepted by the NRC for other non-LWR technologies [10] and [12].

newcleo also provides the following considerations that may be addressed as limits or conditions related to the implementation of the proposed PDCs:

1. **10 CFR Part 53 definition of SR SSCs:** It is newcleo's position that the Part 53 definition of SR SSCs has a narrower scope than that of NEI 18-04 and RG 1.233. This interpretation requires further NRC assessment given the nascence of Part 53.
2. **Functional containment:** The conclusion that a pressure-retaining containment structure is not needed for the LFR-AS-200 is contingent on the definition of quantitative functional containment performance criteria and on the final detailed design. Final NRC approval of the LFR-AS-200 functional containment design is anticipated to occur in a review of the final LFR-AS-200 design in a future licensing application.
3. **Use of SARRDLs:** The application of the SARRDL approach to the LFR-AS-200 (oxide fuel, lead coolant) and specific limits development will be addressed in a separate regulatory engagement or in a review of the final LFR-AS-200 design in future licensing application.
4. **Deletion of ARDC 33:** The claim that primary coolant makeup is not an applicable safety function for the LFR-AS-200 pool-type design may be further addressed through implementation of the NEI 18-04 DID adequacy evaluation methodology.
5. **Deletion of ARDC 35:** The claim that the ECCS function is not applicable to the LFR-AS-200 may be further addressed through implementation of the NEI 18-04 DID adequacy evaluation methodology.
6. **PDC 11 contribution to ARDC 26(3):** The claim that negative global reactivity feedback satisfies the core cooling portion of ARDC 26 (3) by inherent means may be further addressed through implementation of the NEI 18-04 DID adequacy evaluation methodology.
7. **Chemical retention of radionuclides by lead:** The claim that lead provides inherent radionuclide retention supporting the SARRDL approach is based on thermodynamic analyses and ongoing R&D; experimental validation is in progress.

The LFR-AS-200 PDC set will be revised as the design matures and PRA-informed insights become available through the NEI 18-04 IDP. Future amendments to this topical report will be submitted as needed to reflect design evolution and to incorporate the PDC-CDCs for NSRSS SSCs.

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	63/71

8 REFERENCES

- [1] U.S. Nuclear Regulatory Commission, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," Regulatory Guide 1.232, Revision 0, April 2018. (ADAMS Accession No. ML17325A611)
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DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	64/71

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DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	65/71

APPENDIX A. LFR-AS-200 PRINCIPAL DESIGN CRITERIA

Table 2 – PDC set for the LFR-AS-200.

PDC	Title and Content
I.	Overall Requirements
1	<i>Quality standards and records.</i> Safety-significant structures, systems, and components shall be designed, fabricated, erected, and tested to quality standards commensurate with 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 safety-significant function. A quality assurance program shall be established and implemented in order to provide reasonable assurance that these structures, systems, and components will satisfactorily perform their safety-significant functions. Appropriate records of the design, fabrication, erection, and testing of safety-significant structures, systems, and components shall be maintained by or under the control of the nuclear power unit licensee for an appropriate period of time.
2	<i>Design bases for protection against natural phenomena.</i> Safety-related structures, systems, and components shall be protected against or shall be designed to withstand the effects of natural phenomena (e.g., earthquakes, tornadoes, hurricanes, floods, tsunamis, and seiches) and constructed hazards (e.g., dams, transportation routes, military and industrial facilities) considering an event severity up to the design-basis external hazard levels as determined under 10 CFR Part 53, § 53.510 without losing the capability to perform the safety functions identified under § 53.230.
3	<i>Fire protection.</i> Safety-related and, where appropriate, non-safety-related but safety-significant structures, systems, and components 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 facility, particularly in locations with safety-significant structures, systems, or components. Fire detection and fire suppression 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. Fire suppression systems shall be designed to ensure that their rupture or inadvertent operation does not significantly impair the ability of safety-significant structures, systems, and components to perform their safety functions.

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	66/71

PDC	Title and Content
4	<i>Environmental and dynamic effects design bases.</i> Safety-significant structures, systems, and components shall be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, anticipated event sequences, unlikely event sequences, and design basis accidents, commensurate with the safety-significance of the functions to be performed. 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 facility. However, dynamic effects associated with postulated pipe ruptures in nuclear facilities 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 events, such as liquid-metal corrosion following primary coolant leakage scenarios, shall be appropriately considered in the design of safety-significant structures, systems, and components.
5	<i>Sharing of structures, systems, and components.</i> Safety-significant structures, systems, and components 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, commensurate with the safety-significance of the functions to be performed, including, in the event of an anticipated event sequence, unlikely event sequence or design basis accident in one unit, the ability to achieve and maintain safe shutdown of the remaining units.
6	<i>Monitoring, Inspection, Testing, Surveillance</i> Safety-significant structures, systems, and components shall be designed to permit appropriate monitoring, periodic inspection, functional testing, and/or surveillance of important areas and features, to ensure integrity and functional capability commensurate with the safety significance of the functions to be performed. Functional testing shall ensure the operability and performance of the system components, 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 and associated systems into operation for reactor shutdown.
II. Multiple Barriers	
10	<i>Reactor design.</i> The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to ensure that specified acceptable fuel design limits and/or specified acceptable radionuclide release design limits are not exceeded during any condition of normal operation, including the effects of anticipated event sequences.
11	<i>Reactor inherent protection.</i> The reactor core and associated systems shall be designed with sufficient overall negative reactivity feedback to compensate, in the power operating range, for a rapid increase in reactivity, to adequately address the additional heat generation via the available cooling provisions and to ensure that the specified acceptable fuel design limits and/or the specified acceptable radionuclide release design limits are met during anticipated event sequences, unlikely event sequences, and design basis accidents.

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	67/71

PDC	Title and Content
12	<i>Suppression of reactor power oscillations.</i> 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 fuel design limits and/or specified acceptable radionuclide release design limits are not possible or can be reliably and readily detected and suppressed.
13	<i>Instrumentation and control.</i> Instrumentation shall be designed to monitor variables and systems over their anticipated ranges for normal operation and during anticipated event sequences, unlikely event sequences and design basis accidents, as appropriate to ensure reliable operation commensurate with their safety significance. Due consideration shall be given to variables and systems that can affect the fission process, the integrity of the reactor core, safety-significant elements of the reactor coolant boundary, and functional containment. Appropriate controls shall be designed to maintain these variables and systems within prescribed operating ranges.
14	<i>Reactor coolant boundary.</i> The safety-significant elements of the reactor coolant boundary shall be designed, fabricated, and erected so as to address the risk of abnormal leakage, of rapidly propagating failure, of gross rupture, and of unacceptable ingress of secondary coolant or other fluids that may compromise the integrity of the fuel cladding or of the core structural materials.
15	<i>Reactor coolant system design.</i> The reactor 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 reactor coolant boundary are not exceeded during any condition of normal operation, including anticipated event sequences.
16	<i>Functional containment design.</i> A reactor functional containment, consisting of multiple barriers internal and/or external to the reactor coolant system, shall be designed to control the release of radioactivity to the environment and to ensure that the safety-significant functional containment design conditions are not exceeded for as long as design basis conditions require.
17	<i>Electric power systems.</i> Electric power systems shall be provided when required to permit functioning of structures, systems, and components. Should any power system be required to perform safety-significant functions, it shall be designed to provide sufficient capacity and capability commensurate with the safety significance of the functions to be performed, to ensure that the design limits for the fission product barriers are not exceeded during anticipated event sequences.

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	68/71

PDC	Title and Content
19	<i>Control room.</i> 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 during anticipated event sequences and unlikely event sequences. Adequate radiation protection shall be provided to permit access and occupancy of the control room during anticipated event sequences and unlikely event sequences without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent as defined in § 53.020 for the duration of the event. Adequate habitability measures shall be provided to permit access and occupancy of the control room during normal operations, anticipated event sequences and unlikely event sequences. Equipment at appropriate locations outside the control room shall be provided with a design capability for prompt safe shutdown of the reactor, including any necessary instrumentation and controls to maintain the unit in a safe shutdown condition through the use of suitable procedures.
III. Reactivity Control	
20	<i>Protection system functions.</i> The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to ensure that specified acceptable fuel design limits and/or specified acceptable radionuclide release design limits are not exceeded as a result of anticipated event sequences and (2) to sense conditions associated with unlikely event sequences and design basis accidents and to initiate the operation of safety-significant systems and components.
21	<i>Protection system reliability and testability.</i> 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 ensure (1) sufficient reliability of the protection function and (2) that 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.
22	<i>Protection system independence.</i> The protection system shall be designed to ensure that the effects of natural phenomena, and of normal operating, maintenance, testing, and design basis 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.
23	<i>Protection system failure modes.</i> 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, pressure, steam, water, and radiation) are experienced.

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	69/71

PDC	Title and Content
24	<i>Separation of protection and control systems.</i> 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 to ensure that safety is not significantly impaired.
25	<i>Protection system requirements for reactivity control malfunctions.</i> The protection system shall be designed to ensure that specified acceptable fuel design limits and/or specified acceptable radionuclide release design limits are not exceeded during any anticipated event sequence accounting for a single malfunction of the reactivity control systems.
26	<i>Reactivity control systems.</i> The reactor core shall be designed with adequate reactivity control provisions to ensure, with appropriate reliability, independence and diversity commensurate with the safety significance of the event, that: (1) The specified acceptable fuel design limits and/or the specified acceptable radionuclide release design limits are not exceeded, and safe shutdown is achieved and maintained during normal operation, including anticipated event sequences. (2) The rate of reactivity changes resulting from planned, normal power changes is controlled to ensure that the specified acceptable fuel design limits and/or specified acceptable radionuclide release design limits are not exceeded. (3) A means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following an unlikely event sequence or design basis accident is provided. (4) A means for holding the reactor shutdown under conditions which allow for interventions such as fuel loading, inspection and repair is provided.
28	<i>Reactivity limits.</i> The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that unlikely event sequences and design basis accidents resulting from a reactivity increase can neither (1) result in damage to the safety-significant elements of the reactor 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.
IV. Fluid Systems	
30	<i>Quality of reactor coolant boundary.</i> Components that are part of the reactor coolant boundary shall be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. Means shall be provided for detecting appreciable loss of lead level or leakage of radioactive cover gas outside the reactor coolant boundary. Means shall be provided for detecting ingress of water, steam or other fluids into the reactor coolant boundary.

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	70/71

PDC	Title and Content
31	<i>Fracture prevention of reactor coolant boundary.</i> Safety-significant elements of the reactor coolant boundary shall be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, anticipated event sequences, unlikely event sequences, and design basis accidents (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, anticipated event sequences, unlikely event sequences, and design basis accidents and the uncertainties in determining (1) material properties, (2) the effects of irradiation and coolant composition, including impurities and oxygen content, on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws.
34	<i>Residual heat removal.</i> A system to remove residual heat shall be provided. For normal operations and anticipated event sequences, the system safety function shall be to transfer fission product decay heat and other residual heat to the ultimate heat sink at a rate such that specified acceptable fuel design limits or specified acceptable radionuclide release design limits and the design conditions of the reactor 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.
35	<i>Passive Decay Heat Removal.</i> Passive means to remove decay heat during unlikely event sequences and design basis accidents shall be provided, such that the specified acceptable fuel design limits and/or specified acceptable radionuclide release design limits are not exceeded.
44	<i>Structural and equipment cooling.</i> In addition to the heat rejection capability of the residual heat removal system, means to transfer heat from safety-significant structures, systems, and components to an ultimate heat sink shall be designed, as necessary, to transfer the combined heat load of these safety-significant structures, systems, and components during normal operation and anticipated event sequences.
VI.	Fuel and Radioactivity Control
60	<i>Control of releases of radioactive materials to the environment.</i> 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 operation and anticipated event sequences. Sufficient holdup capacity (or other means necessary) shall be provided for retention (or other controlling measures) 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.

DOC-ID	BUSINESS MARKING	EXPORT CONTROL	REFERENCE CODE	REV.	PAGE
0001012990	newcleo GENERAL	NOT DUAL USE	LFR-200-TBF-REP-002	0	71/71

PDC	Title and Content
61	<i>Fuel storage and handling and radioactivity control.</i> The fuel storage and handling, radioactive waste, and other systems that may contain radioactivity shall be designed to ensure adequate safety, commensurate with the safety significance of the functions to be performed, during normal operation, anticipated event sequences, unlikely event sequences, and design basis accidents, as appropriate. These systems shall be designed (1) with suitable shielding for radiation protection, (2) with appropriate containment, confinement, and filtering systems, (3) with a residual heat removal capability having reliability commensurate with the safety significance of the functions to be performed, and (4) to prevent significant reduction in fuel storage cooling during anticipated event sequences, unlikely event sequences, and design basis accidents, as appropriate.
62	<i>Prevention of criticality in fuel storage and handling.</i> Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.
63	<i>Monitoring fuel and waste storage.</i> 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.
64	<i>Monitoring radioactivity releases.</i> Means shall be provided for monitoring the reactor building atmosphere, effluent discharge paths, and plant environs for radioactivity that may be released during normal operation, anticipated event sequences, and unlikely event sequences.
VII. Additional PDC	
70	<i>Reactor vessel and reactor system structural design basis.</i> The design of the reactor vessel and its internals shall be such that their integrity is maintained during unlikely event sequences and design basis 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.
71	<i>Reactor building design basis.</i> The reactor building shall be designed to structurally protect the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink and for reactivity control during unlikely event sequences and design basis accidents.
72	<i>Primary coolant and cover gas chemistry control.</i> Systems shall be provided as necessary to maintain the purity of primary coolant lead and cover gas within specified design limits. These limits shall be based on consideration of (1) liquid metal corrosion-erosion, (2) fouling and plugging of passages, and (3) radionuclide concentrations.
73	<i>Lead heating system.</i> A lead heating system shall be provided, when necessary, for the primary coolant system. This heating system and its controls shall be appropriately designed to ensure that the bulk temperature of the primary coolant is maintained within design limits. These limits shall be based on consideration of the effect of primary coolant solidification on the integrity of safety-significant structures, systems, and components, specifically during very low power conditions within normal operation and anticipated event sequences.