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Sent:	Wednesday, August 5, 2020 8:06 PM
To:	Sebrosky, Joseph
Cc:	Reckley, William; Segala, John; Cubbage, Amy; francis Akstulewicz; Shirley, Barry J.
Subject:	[External_Sender] FW: Mapping report - ready for NRC review
Attachments:	Mapping FSFs Rev B August 2020 Version 15 - NRC review (1).docx

Greetings,

Please find attached the next TICAP product referred to as the Mapping report.

TICAP is requesting an expedited review of this report. The NRC has been briefed on several occasions about the content and results of the mapping and binning activities. And the NRC has provided some comments on the presentations of this material following the June 11, 2020 meeting. TICAP provided responses to the NRC comments at the most recent July 30, 2020 TICAP-NRC working meeting.

TICAP requests comments on the report material no later than COB Monday 8/24/20 in order to provide a revised report including resolution of NRC comments to INEL no later than 8/31/20. TICAP will accept formal (letter) or informal (email) comments to facilitate the NRC review and comment process. TICAP is also willing to meet with the NRC to discuss their comments informally if necessary to meet our scheduler requirements. Please inform us promptly of your planned response plan.

Sincerely
Amir

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Technology Inclusive Content of Application Project
For Non-Light Water Reactors

Mapping Regulatory Requirements to Fundamental Safety Functions

Draft Report Revision B
Issued for NRC Review and Comment

Document Number
SC-16166-101 Rev B

Battelle Energy Alliance, LLC
Contract No. 221666
SOW-16166

August 6, 2020

Prepared for:
U.S. Department of Energy (DOE)
Office of Nuclear Energy
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

Technology Inclusive Content of Application Project
For Non-Light Water Reactors

Mapping Regulatory Requirements to Fundamental Safety Functions

Draft Report Revision B
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August 6, 2020

Issued for Collaborative Review by:

____ ________________________
Amir Afzali, Next Generation Licensing and Policy Director Date
Southern Company Services

Disclaimer

Abstract

Non-light water reactor (non-LWR) technologies will play a key role in meeting the world's future energy needs and will build on the foundation established by the current light water reactor (LWR) nuclear energy fleet. The Technology Inclusive Content of Application Project (TICAP) is an important step in establishing that licensing framework. This Department of Energy (DOE) cost-shared, owner/operator-led initiative will produce guidance for developing risk-informed and performance-based content for specific portions of the Nuclear Regulatory Commission (NRC) license application Safety Analysis Report (SAR) for non-LWR designs.

TICAP's objective is to propose a formulation for application content for the portions of the SAR that are based on implementation of Nuclear Energy Institute (NEI) 18-04, "Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development."[1] The TICAP guidance will help ensure completeness of information submitted to NRC while avoiding unnecessary burden on the applicant and providing the appropriate scope and level of detail for the content of application commensurate with the risk profile and complexity of the design. This objective will be achieved through (1) proposing an application content formulation that is based on demonstrating that a set of Fundamental Safety Functions (FSFs) are met and (2) demonstrating that the formulation will provide an adequate level of information consistent with the same type of information required for an LWR-based design.

As part of demonstrating that a content of application formulation outlined above will satisfy the underlying safety basis of current regulations, the following steps are taken:

A. A set of FSFs is defined. These FSFs, defined by an earlier TICAP activity (Southern Company Document Number SC-16166-100[4]), are the following:
    1. Retaining Radioactive Materials
    2. Controlling Reactivity
    3. Removing Heat from the Reactor and Waste Stores

TICAP will use the NEI 18-04-based safety case to formulate its content proposal. NEI 18-04 included a set of FSFs that was also endorsed by the NRC in Regulatory Guide (RG) 1.233. The sets of FSFs contained in NEI 18-04 and RG 1.233 are, respectively:

- Controlling heat generation, controlling heat removal, and retaining radionuclides, and
- Reactivity and power control, heat removal, and radionuclide retention.

TICAP has evaluated the differences between the set defined in SC-16166-100 and those contained in NEI 18-04 and RG 1.233 and has concluded that the sets of FSFs are functionally equivalent.

These FSFs provide comprehensive coverage of important plant functions for a spectrum of reactor technologies and postulated initiating events and design basis accidents that, if demonstrated, will provide reasonable assurance of adequate protection of the health and safety of the public and the environment.

B. The current requirements are mapped to the TICAP FSFs defined in Paragraph A above. The objective is to demonstrate that the existing design requirements contained in 10 CFR Parts 50 and 52 (and regulations referenced by those parts) are in place to substantiate that one or more fundamental safety functions are met.

C. The General Design Criteria for LWRs can be organized into several information categories that provide answers to the following questions:

- **What** are the performance objectives for the FSFs?
- **When** do the FSF's performance objectives need to be demonstrated?
- **How** do plant capabilities (functional and structural) demonstrate that the FSFs are met?
- **How well** do these capabilities need to be performed to provide reasonable assurance?

The LMP-based safety case, which would be developed for an advanced non-LWR reactor meeting the FSFs, answers the same set of questions.

This report documents the results of analyses performed for Steps B and C.

Table of Contents

List of Abbreviations

AOO	Anticipated Operational Occurrence
CFR	Code of Federal Regulations
COL	Combined License
DG	Draft Regulatory Guide
DID	Defense-in-Depth
DOE	Department of Energy
ECCS	Emergency Core Cooling System
FSAR	Final Safety Analysis Report
FSF	Fundamental Safety Function
GDC	General Design Criteria
HTGR	High Temperature Gas Reactor
INL	Idaho National Laboratory
LBE	Licensing Basis Event
LMP	Licensing Modernization Project
LWR	Light water reactor
NEI	Nuclear Energy Institute
NGNP	Next Generation Nuclear Plant
non-LWR	Non-light water reactor
NRC	Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Commission technical report designation
PDC	Principal Design Criteria
PRA	Probabilistic Risk Assessment
PSF	PRA Safety Function
QHO	Quantitative Health Objective
RIPB	Risk-informed and performance-based
SAR	Safety Analysis Report
SECY	NRC Office of the Secretary designation for documents containing policy, rulemaking, and adjudicatory matters as well as general information for action or consideration by NRC
SSCs	Structures, Systems, and Components
TICAP	Technology Inclusive Content of Application Project
WWHHW	"What," "When," "How," and "How Well" binning categories for mapping GDC

1.0 INTRODUCTION AND BACKGROUND

1.1 TICAP Description

Non-light water reactor (non-LWR) technologies will play a key role in meeting the world's future energy needs and will build on the applicable foundation established by the current light water reactor (LWR) nuclear energy fleet. An efficient and cost-effective non-LWR licensing framework that facilitates safe and cost-effective construction and operation is a critical element for incentivizing private sector investment. The Technology Inclusive Content of Application Project (TICAP) is seen as a critical part of the new licensing framework. This Department of Energy (DOE) cost-shared, owner/operator-led initiative will produce guidance for developing content for specific portions of the Nuclear Regulatory Commission (NRC) license application Safety Analysis Report (SAR) for non-LWR designs.

Existing LWRs are the country's largest source of emissions-free, dispatchable electricity, and they are expected to remain the backbone of nuclear energy generation for years to come. However, as the energy and environmental landscape has evolved, interest has grown in advanced nuclear energy systems that promise increased margins of safety, superior economics, improved efficiency, greater fissile-fuel utilization, and reduced high-level waste generation. In addition to electricity generation, these technologies can expand upon what up until the present has been the traditional use of nuclear energy by providing a viable alternative to fossil fuels for industrial process heat production and other applications.

The current regulatory framework for nuclear reactors was developed over decades for LWRs using zirconium-clad uranium oxide fuel coupled with the Rankine power cycle. Many advanced, non-LWRs are in development, with each reactor design differing greatly from the current generation of LWRs. For example, advanced reactors might employ liquid metal, gas, or molten salt as a coolant, enabling them to operate at lower pressures but higher temperatures than LWRs. Some employ a fast rather than a thermal neutron spectrum. A range of fuel types is under consideration, including fuel dissolved in molten salt and circulated throughout the primary coolant system. In general, advanced reactors emphasize passive safety features that do not require rapid action from powered systems to prevent radionuclide releases. Given these major technical differences, changes to the current license application content are needed to present a risk-informed safety case for the deployment of advanced reactor designs.

In order to help pave the way for the licensing of these new technologies, the DOE provided cost-share funding and support for the TICAP initiative, a utility-led initiative to improve the effectiveness and efficiency of licensing non-LWRs given the NRC's current regulatory framework. The initiative recognizes that significant levels of industry input and advocacy are needed in collaboration with the NRC to make the necessary regulatory changes needed to facilitate efficient licensing of advanced reactor technologies.

The portions of the SAR on which this work will focus are those directly informed by the methodologies defined in the Nuclear Energy Institute (NEI) publication NEI 18-04, "Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development."[1] The TICAP guidance will help ensure sufficiency and completeness of information submitted to the NRC while avoiding unnecessary burden on the applicant and will provide the appropriate scope and level of detail for the content of application commensurate with the complexity of the design.

The goal of TICAP is to develop license application content guidance with the following attributes:

- Technology inclusive to be generically applicable to all non-LWR designs.
- Risk-informed and performance-based (RIPB) to:
  - Ensure the NRC review is focused on information that impacts the safety case of reactors.
  - Create coherency and consistency in the scope and level of detail requirements in the license application for various advanced technologies and designs.
  - Provide for flexibility during construction.
  - Encourage innovation by focusing on the final results as opposed to the pathway taken to achieve the results.

The use of a modernized, technology inclusive RIPB license application content will advance:

- The nuclear industry (developers and owners/operators) longstanding focus on and commitment to continuous improvement.
- The NRC's goal of having a safety-focused review that minimizes the burden of generating and supplying information that is not important to a safety determination.
- The NRC and industry objective of reaching agreement on how to implement reasonable assurance of adequate protection for non-LWRs.
- NRC's stated objective and policy statement regarding the use of risk-informed decision-making to remove unnecessary regulatory burden.

TICAP intends to build on the foundation established by the Licensing Modernization Project (LMP) as documented in NEI 18-04 and the associated white papers. These documents presented a modern, technology-inclusive, RIPB process for selection of Licensing Basis Events (LBEs); safety classification of Structures, Systems, and Components (SSCs) and associated risk-informed special treatments; and determination of Defense-in-Depth (DID) adequacy for non-LWRs. The TICAP application guidance will focus on the portion of the application addressed by the LMP methodology and central to the applicant's safety case. Ultimately, the full scope of information presented in the application must demonstrate reasonable assurance of adequate protection of public health and safety.

1.2 Summary of TICAP Approach

As stated above, the TICAP objective is to propose a formulation for application content for the portions of the SAR that are based on implementation of NEI 18-04. The TICAP guidance will help ensure completeness of information submitted to the NRC while avoiding unnecessary burden on the applicant and providing the appropriate scope and level of detail for the content of application commensurate with the risk profile and complexity of the design. This objective will be achieved through (1) proposing an application content formulation that is based on demonstrating a set of Fundamental Safety Functions (FSFs) are met and (2) demonstrating that the formulation will provide an adequate level of information consistent with the same type of information required for an LWR-based design.

As part of demonstrating that a content of application formulation, as outlined above, will demonstrate the underlying safety basis of the current regulation, the following steps are taken:

A. A set of FSFs is defined. These FSFs (defined by an earlier TICAP activity (Southern Company Document Number SC-16166-100[4]) are the following:

   1. Retaining Radioactive Materials
   2. Controlling Reactivity
   3. Removing Heat from the Reactor and Waste Stores

   TICAP will use the NEI 18-04-based safety case to formulate its content proposal. NEI 18-04 included a set of FSFs that were also endorsed by the NRC in Regulatory Guide (RG) 1.233. The set of FSFs contained in NEI 18-04 and RG 1.233 are, respectively:

   - Controlling heat generation, controlling heat removal, and retaining radionuclides, and
   - Reactivity and power control, heat removal, and radionuclide retention.

   TICAP has evaluated the differences between the set defined in SC-16166-100 and those contained in NEI 18-04 and RG 1.233 and has concluded that the sets of FSFs are functionally equivalent.

   These FSFs provide comprehensive coverage of important plant functions for a spectrum of reactor technologies and postulated initiating events and design basis accidents that, if demonstrated, will provide reasonable assurance of adequate protection of the health and safety of the public and the environment.

B. The current LWR requirements are mapped to the above FSFs. The objective is to demonstrate that the existing design requirements contained in 10 CFR Parts 50 and 52 (and regulations referenced by those parts) are in place to substantiate that one or more fundamental safety functions are met.

C. The LWR General Design Criteria are organized into information categories that provide answers to the following questions:

   - **What** are the performance objectives for the FSFs?
   - **When** do the FSF’s performance objectives need to be demonstrated?
   - **How** plant capabilities (functional and structural) demonstrate that the FSFs are met?

- **How well** do these capacities need to be performed to provide reasonable assurance?

The NEI 18-04-based safety case, which would be developed based on meeting the FSFs, provide the answers to the same set of questions.

The objective of this report is to document the results of analyses performed for steps B and C.

1.3 Objective of Regulation Mapping

The objective of this report is to document the results of analyses performed for Steps B and C, as described in Section 1.2. In summary, this mapping effort examines the regulatory requirements within 10 CFR Part 50, including any referenced regulations. However, it is not the intent of this mapping report to expand the LMP process beyond the evaluation scope presented in the NEI 18-04 guidance document. The NEI guidance document describes acceptable processes for selection of LBEs, safety classification of SSCs, and determination of the DID adequacy for a technology-inclusive array of advanced non-LWR designs. The regulatory acceptability of these processes and their results is based on a design's safety case that demonstrates that it meets regulatory radiological release performance requirements for a set of technology-inclusive fundamental safety functions.

In an earlier TICAP report,[4] a set of technology inclusive FSFs was defined. The objective of this mapping report is to demonstrate that technical design requirements contained in 10 CFR Parts 50, 52, and 100, and their respective appendices can be mapped to the same set of FSFs. Technical design requirements in other parts of the NRC regulations are beyond the scope of the current TICAP effort and have not been mapped. To provide additional confidence that the technical requirements are captured in the mapping process, the information requirements for the contents of applications found in 10 CFR 50.34 (a) and (b), 10 CFR 52.79, and 10 CFR 52.80 were examined for any additional technical or programmatic requirements. Because no additional technical requirements were identified, no discussion of the additional examination is included in this report.

An application that uses the methodology in the NEI 18-04 guidance document to demonstrate that the set of FSFs is satisfied is not a sufficient legal basis for the NRC to issue a license. In addition, an applicant will also need to address in its application topics such as quality assurance, radiation protection of workers, security, safeguards, financial assurance, emergency preparedness, maintenance, and operator licensing, to name just a few. To the extent that additional guidance is needed for satisfaction of these ancillary requirements, that guidance for non-LWR advanced reactors is beyond the scope of TICAP and may be addressed by other initiatives.

2.0 PREVIOUS EFFORTS AT MAPPING CURRENT REQUIREMENTS TO ADVANCED REACTOR TECHNOLOGIES

The nuclear industry and NRC have performed several studies that assessed the regulatory framework and the applicability of the current regulations for use in the review of advanced non-LWR technologies. Few, if any, have tried to establish a direct relationship between the current set of regulations to a set of fundamental safety functions. Despite the fact that these studies of regulatory applicability did not employ mapping to fundamental safety functions, they do provide insights into the body of regulations that NRC reviews for compliance as part of individual applications and can serve as a starting point for the effort to map the regulations to FSFs.

2.1 NUREG-1860, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing"

In NUREG-1860,[5] the NRC staff examined the existing regulatory framework with the objective of creating a stand-alone set of requirements that would employ a risk-informed, performance-based structure for licensing advanced reactors. The technical basis and process discussed in the NUREG examined the existing regulations in 10 CFR 50 to identify technical and administrative items that would be compatible and interface with other parts of 10 CFR, such as Parts 20, 51, 52, 73, and 100. While not a specific mapping of regulatory requirements to fundamental safety functions, the staff assessment of 10 CFR 50 identified where the requirements can be used directly or with modification in a risk-informed and performance-based licensing approach. This study showed that there are many 10 CFR 50 requirements and General Design Criteria that can be used directly for licensing advanced reactors. The report also examined how other parts of the regulations relate back to 10 CFR 50, for example, how sections in 10 CFR 54 link specifically to requirements contained in 10 CFR 50.

This NUREG is helpful to the TICAP mapping process because it provides a regulation-by-regulation listing of the necessary requirements for licensing a plant under 10 CFR 50. The reference material provides a comprehensive set of regulations that will need to be mapped to the fundamental safety functions. It also provides references to the programmatic requirements that are essential in maintaining the integrity of the design and therefore can be mapped to each of the individual FSFs. Finally, it provides a listing of the administrative requirements for fulfilling application filing requirements.

2.2 Next Generation Nuclear Plant Project Regulatory Gap Analysis for Modular HGTRs

Under the auspices of the DOE, the Idaho National Laboratory (INL) conducted a study[6] to evaluate existing regulatory requirements and guidance against the design characteristics specific to a generic modular high-temperature gas reactor (HTGR). The study focused on regulations and supporting guidance considered relevant to the development of an HTGR licensing framework and did not attempt to evaluate all regulations that would be of interest to future HTGR license applicants. The study evaluated 3,611 items and concluded that 1,022 items could be excluded from further analysis because they were administrative in nature. Of the remaining 2,589 items, only 108 were identified as needing further evaluation; and of the 108 items, only 15 items were identified as regulations that needed some modification. The 93 remaining items were items that related to modifications of existing regulatory guidance such as the Standard Review Plan (NUREG 0800).

The INL study was a comprehensive examination of the regulatory framework and included a review of the following parts from 10 CFR:

- 10 CFR Part 20, Standards for Protection Against Radiation
- 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities
- 10 CFR Part 51, Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions
- 10 CFR Part 52, Licenses, Certifications, and Approvals for Nuclear Power Plants
- 10 CFR Part 55, Operators' Licenses
- 10 CFR Part 70, Domestic Licensing of Special Nuclear Material
- 10 CFR Part 73, Physical Protection of Plants and Materials
- 10 CFR Part 100, Reactor Site Criteria
- 10 CFR Part 140, Financial Protection Requirements and Indemnity Agreements
- 10 CFR Part 961, Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste
- NUREG-0737, Clarification of TMI Action Plan Requirements
- NUREG-0933, Unresolved and Generic Safety Issues

As comprehensive as this study was, it did not evaluate the full set of regulations that forms the complete licensing framework. For example, it did not evaluate regulations in Parts 2, 21, 26, 71, 72, 74, and 95.

Although the INL study did not attempt to map the regulations or guidance to specific FSFs, it proved valuable to the TICAP mapping activity because it provided insights into the relationships of regulations within the licensing framework and served as an additional cross-check with other studies for determining the essential regulations within the LMP scope that will need to be mapped to FSFs.

2.3 Oklo, Inc. Pilot Application of DG-1353

As part of licensing activities supporting its advanced micro-reactor effort, Oklo, Inc. developed a pilot application structure[7] for implementing DG-1353[8] and submitted the application structure to NRC. The pilot application provided one perspective on how analysis of LBEs, safety-classification of SSCs, and resulting confirmation of DID adequacy could inform the level of detail needed to meet existing regulatory requirements in a possible risk-informed and performance-based license application. The structure of the pilot application simulated the structure of regulatory requirements for a Combined License (COL) application (10 CFR 52.77, 52.79, and 52.80). The focus of the pilot structure study was on the content of the Final Safety

Analysis Report (FSAR). In the Oklo study, the content directly followed the order of the regulations, excluding sections with specific applicability only to LWRs.

The pilot application study was limited in scope in that it analyzed only internal events for an operating plant. It did not assess natural phenomena events or include safeguards information. The intent of the Oklo study was to achieve alignment with the NRC that the level of detail for sections within the scope of its report simulated as closely as possible the type and amount of information required for an actual license application.

The value of the Oklo study to the TICAP mapping activity is that its scope was focused on specific content of the FSAR for a COL application. While the scope of the Oklo study was narrower than the LMP scope in that it was limited to only internal events, it provided another cross-check with other studies about the specific regulations that are applicable to a COL application and would need to be mapped to an FSF. It also provided insights into those topic areas that could be more risk-informed in a risk-informed licensing structure and could be used to support future activities in the TICAP effort.

3.0 REGULATIONS INCLUDED IN THE MAPPING AND THE OUTCOMES RELATED TO MAPPING EACH OF THE FUNDAMENTAL SAFETY FUNCTIONS

3.1 Guidelines for the Mapping Process

In order to effectively map regulations to the appropriate FSFs, it is necessary to understand each of the defined roles of the FSFs. For ease of reference, the FSFs, as presented in a previous TICAP report,[4] are repeated below. The performance objectives of FSFs are expected to be met during normal operation or in response to licensing basis events (which in TICAP terminology are binned in the "When" category) that are within the scope of the NEI 18-04 methodology.

The set of FSFs used in the mapping process includes the following:

1. Retaining Radioactive Materials—The FSF of Retaining Radioactive Materials is defined as the active, passive, or inherent means provided to prevent or mitigate the release of radioactive materials from the plant to the public and the environment.
2. Controlling Reactivity—The FSF of Controlling Reactivity is defined as the active, passive, or inherent means provided (1) to control the nuclear chain reaction consistent with the intended plant operating conditions, (2) to terminate the nuclear chain reaction when transient or accident conditions dictate that the facility must be shut down, and (3) to prevent inadvertent criticality in the reactor core, primary system, or other areas of the plant where inadvertent criticality is an adverse condition that could result in unacceptable radiological consequences.
3. Removing Heat from the Reactor and Waste Stores—The FSF of Removing Heat from the Reactor and Waste Stores is defined as the active, passive, or inherent means provided (1) to remove the heat generated from the nuclear chain reaction during normal plant operating modes so that the nuclear fuel and primary system retain their integrity, (2) to remove the decay or residual heat from the reactor and primary system when the nuclear chain reaction is terminated and when the facility is shut down, and (3) to remove the residual heat from material that is being stored in waste handling and fuel handling areas so that unplanned releases of radioactive materials from the plant do not occur.

As stated earlier, these FSFs are consistent with those provided in NEI 18-04.

The FSFs' performance objectives shall be met during full-power, low-power, and shutdown operations and for both internal and external events (such as seismic events, fire, and flooding [internal and external] as well as high winds and tornados).

The mapping effort of this report will focus, at a macro level, on technical requirements prescribed by the regulations in 10 CFR 50. The regulations that are important for the mapping process are those that currently establish technical requirements that are incorporated into the design to address technology-specific initiating events and resulting LBEs. This report will not attempt to map existing regulatory guidance that has traditionally been used to describe processes that are acceptable for demonstrating that the regulations have been met.

The activities under this section will examine the full spectrum of regulatory requirements at the regulation level and will not perform a sub-paragraph-by-sub-paragraph assessment of each of the regulations. An effort to map every regulation at the sub-paragraph level is beyond the scope of the current TICAP activities. As mentioned earlier, the information requirements for the content of applications found in 10 CFR 50.34 (a) and (b) and 10 CFR 52.79 and 52.80 were also examined for any additional technical requirements. Because no additional technical requirements were identified, a discussion of the examination is not included in this report.

3.2 Structure of the Mapping Results

Table 1 in Section 4 of this report provides the structure for the TICAP FSF mapping efforts. The mapping activity included 157 regulation sections found in 10 CFR 50, General Design Criteria, and 10 CFR 50 appendices.

Column 1 of Table 1 lists the specific regulatory section that will be mapped. Column 2 contains a summary of the nature of the regulatory requirement or the safety objective that the regulation provides. Columns 3, 4, and 5 are the specific columns for the FSFs. Column 3 is the FSF for retaining radionuclide materials; Column 4 is the FSF for controlling reactivity; and Column 5 is the FSF for removing heat. Column 6 documents the regulatory programs that are programmatic in nature and would assure the reliability and operability of SSCs needed to assure that the FSFs are performed. Column 7 lists the regulation sections that are procedural or administrative in nature; sections listed in Column 7 contain no technical requirements. Column 8 contains a brief discussion or rationale that supports why the FSFs are mapped to the specific regulation. In those cases where the regulation is descriptive and the selection of the FSFs is straightforward, only a short discussion is provided.

4.0 SUMMARY OF MAPPING RESULTS FOR REGULATIONS TO FUNDAMENTAL SAFETY FUNCTIONS

Table 1 presents the results of the TICAP mapping process. For the purposes of this report, programmatic regulations are those regulations that assure the design can actually perform as predicted throughout the operating life of the plant. These regulations cover a wide range of regulatory topics, including the evaluation and mitigation of uncertainties in the design, as well as programs that keep and sustain the design within the approved design envelope. The regulations that are specifically mapped to the individual FSFs establish plant capabilities and performance outcomes. It is the plant capabilities and performance outcomes that serve as the principal basis for adequate protection conclusions.

The order of regulations presented in the table follows the outline of 10 CFR 50 and its appendices and will be supplemented with regulations outside of Part 50 when those regulations are referenced in the content of application sections of Parts 50 and 52, specifically, 50.34 (b), the FSAR for an operating license, and 52.79 and 52.80 for a COL. Each of the General Design Criteria of Appendix A of Part 50 is also individually mapped to one or more of the FSFs. The mapping of the General Design Criteria is important because it represents the minimum requirements for the principal design criteria for LWRs. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public, and by extension, present the minimum requirements necessary for reasonable assurance of adequate protection.

For sections that are categorized as administrative, the plain reading of the regulation usually provides sufficient information to understand why the regulation would be determined to be administrative. Therefore, additional discussion about why a requirement is categorized as administrative is unnecessary.

As discussed earlier, the information requirements found in the specific contents of application sections of 10 CFR Parts 50 and 52 were examined to provide added confidence that all safety-significant technical requirements are identified and mapped to the set of FSFs. The examination of the information requirements did not identify any additional technical or programmatic requirements.

Table 1. Results of Mapping Regulations to Fundamental Safety Functions

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
50.1	General Provisions					X	
50.2	General Provisions					X	
50.3	General Provisions					X	
50.4	General Provisions					X	
50.5	General Provisions					X	
50.7	General Provisions					X	
50.8	General Provisions					X	
50.9	General Provisions					X	
50.10	License required to construct					X	
50.11	Exceptions to 50.10					X	
50.12	Process for relief from regulations					X	Administrative because establishes a process that governs requests for relief from existing regulations
50.13	Relief from design against foreign enemies					X	
50.20	Class of licenses					X	
50.21	Class of licenses					X	
50.22	Class of licenses					X	
50.23	Class of licenses					X	
50.30	Filing applications					X	
50.31	Combine licenses					X	
50.32	Eliminate repetition					X	

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
50.33	Content of Application, General—financial and emergency plan, de-commissioning funds					X	Governs information needs related to non-technical content of applications; the programmatic elements of some of the topics are prescribed in the specific regulations for those topics
50.34	Content of Application, Technical Information					X	Governs information needs related to technical contents of application; does not establish technical requirements for safety but only the topic areas where design information must be provided to NRC as part of a license application
50.34—Referenced Regulations	Part 100—Reactor Site Criteria	X	X	X			Establishes maximum dose criteria for releases following postulated maximum hypothetical accident. This sets one of the performance objectives for the FSFs. As such, it can be mapped into more than one FSF.
	Appendix B—Quality Assurance				X		Programmatic because it governs all aspects of design, procurement, construction of SSCs important to safety
	Appendix S—Earthquake Engineering Criteria for Nuclear Power Plants				X		Programmatic because it governs the seismic design requirements for safe operation of SSCs
	Part 20—Standards for Protection Against Radiation	X	X	X			Governs releases of effluents from facilities as well as radiation worker safety and exposure limits. This sets one of the performance objectives for the FSFs. As such, it can be mapped into more than one FSF.
50.35	Issuance of Construction Permit					X	

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
50.36	Technical Specifications				X		Programmatic because it requires technical specifications for technical areas affecting safe operation of SSCs
50.36a	Technical Specifications, Effluents				X		Programmatic emphasis is on controlling routine plant releases only
50.36b	Environmental Conditions					X	
50.37	Access to Classified Information					X	
50.38	Eligibility of Applicants					X	
50.39	Public Access to Applications					X	
50.40	Common Standards					X	
50.41	Class 104					X	
50.42	Class 103					X	
50.43	Extra Class 103				X		
50.44	Combustible Gas Control to Avoid Energetic Loss of Containment	X					Regulation to control available hydrogen to avoid possible conflagration or combustion that leads to over-pressurization and loss of containment
50.45	Standards for Licenses					X	
50.46	Emergency Core Cooling System Requirements Appendix K—ECCS Evaluation Models	X	X	X			Requires the demonstration that for zirconium clad fuel, the emergency core cooling system is designed such that its cooling performance following a postulated loss of coolant accident will not result in the design exceeding the acceptance criteria contained within the rule; protection of cladding barrier, and hydrogen generation

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
50.46a	RCS Venting Requirements			X			Contains requirements for reactor coolant system venting to maintain core cooling if accumulation of non-condensable gases could cause loss of function, avoid rupture of primary system piping or reactor vessel, reduce pressures to provide makeup cooling water
50.47	Emergency Preparedness				X		Emergency preparedness does not directly map to any FSF, but it has been considered an important element of DID for the large LWRs in providing added assurance of protection of the public in an event that the performance objective of the FSFs were not met. It is recognized that 50.47 does not consider new reactor design enhanced safety margins and use of high reliability SSC which use passive and inherent features, to meet performance objectives of the FSFs. As a result, NRC has issued proposed 50.160 for use by SMRs and ARs in recognition that achieving performance objectives of the FSFs with reasonable assurance will meet the intent of the regulation. There are specific requirements that are programmatic in nature.
50.48	Evaluation of Fire Protection Requirements, including Fire Protection Program				X		Programmatic requirements for fire protection program to protect equipment important to safety; Sections III.G, J, and O are adequate protection
50.49	Environmental Qualification of Safety-Related Equipment				X		Programmatic requirement for qualification of equipment to assure operability of safety equipment in harsh environmental conditions

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
50.50	Issuance of Licenses					X	
50.51	Continuation of Licenses					X	
50.52	Combining Licenses					X	
50.54	Required License Conditions				X		Establishes requirements for all licenses in multiple topic areas; some of the requirements are administrative as well.
50.55	More Conditions of Licenses					X	Requires reporting of defects and implementation of QA program.
50.55a	Required Construction Codes and Standards	X	X	X			Establishes acceptable codes for construction for use at nuclear power plants to assure that safety margins are included in the design
50.56	Process for License Conversions					X	
50.57	Findings to Issue an Operating License					X	
50.58	Hearings and ACRS Reviews					X	
50.59	Process for Changes to Approved Licensing Bases				X		Establishes process for changes to approved licensing basis; preserves integrity of approved licensing basis for safe operation of SSCs

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
50.60	Acceptance Criteria for Fracture Prevention Measures for LWRs Appendix G – Fracture Toughness Requirements Appendix H – Reactor Vessel Material Surveillance Program Requirements	X		X			Establishes requirements to assure that vessel failure is minimized, and retention of primary coolant is assured
50.61	Preserve Integrity of Reactor Vessel from Temperature Event	X		X			Establishes requirements to assure that vessel failure is minimized, and retention of primary coolant is assured
50.61a	Alternatives to 50.61 Requirements	X		X			See above
50.62	Mitigate Worst Case Failure to Scram Event	X	X	X			Assures that design is capable of handling a failure to scram event from a primary system overpressure event; examines other means for introducing cooling to vessel; requires alternate scram system
50.63	Mitigate Station Blackout Event	X		X			Requires examination of coping capability of design to protect the core and minimize releases in the event of loss of all station power; essential power supplies
50.64	Use of High Enriched Uranium					X	Outside of scope of the TICAP activity

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
50.65	Required Maintenance Plan for Important Equipment				X		Programmatic requirements for maintaining plant equipment important to safety; would impact all components supporting performance of FSFs
50.66	Thermal Annealing—Prevent Failure of Reactor Vessel	X		X			Provides options for annealing to prevent reactor vessel failure and maintain primary coolant. This regulation would not be applicable to an application for an initial operating license.
50.67	Accident Source Terms for Design Basis Accident Evaluations	X					Provides for calculation of realistic source terms following postulated accidents; used in assessment of offsite exposures and need for potential design features to reduce doses to within regulatory limits
50.68	Criticality Accident Requirements—Spent Fuel Pool and Dilution Events		X				Inadvertent criticality in the spent fuel pool; concerns with loss of subcriticality resulting from dilution events; exposure to radioactivity
50.69	Risk-Informed SSC Classification and Treatment Requirements				X		Programmatic because it governs the application of risk assessments to SSC safety classifications that would support satisfaction of all FSFs; used to establish special treatment requirements for safety important equipment
50.70	Inspection Requirements—Resident Inspector Requirements					X	
50.71	Records Retention and Maintenance					X	
50.72	Event Reporting					X	
50.73	Event Reporting					X	

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
50.74	Operator Status Changes					X	
50.75	Report—Decommissioning Planning					X	
50.76	Change of Financial Status					X	
50.78	International Safeguards Agreements					X	Relates to non-proliferation matters
50.80	Transfer of Licenses by Owner					X	
50.81	Creditor Regulations					X	
50.82	Termination of License					X	
50.83	Release Requirements for Site Use					X	
50.90	Process to Amend or Modify Approved Licensing Basis				X		Programmatic change process for modifying approved licensing basis; preserves integrity of licensing basis; all equipment important to safety would support satisfaction of FSFs
50.91	Public Notice Requirements for Amendments					X	
50.92	Process to Issue License Amendment					X	
50.100	Revocation or Suspension of License					X	
50.101	Repossession of Nuclear Materials					X	
50.102	Operation after Revocation of License					X	

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
50.103	Suspension or Operation during War or National Emergency					X	
50.109	Backfit—Limitation on Staff-Imposed Changes to Approved Licensing Basis					X	Administrative process imposed on the NRC for modification of approved licensing basis based on new NRC requirements or changes in previous staff positions; would affect all structures or components supporting satisfaction of FSFs
50.110	Violations					X	
50.111	Criminal Penalties					X	
50.120	Training Requirements for Plant Personnel for Certain Plant Positions					X	
50.150	Aircraft Impact Assessment	X		X			Requires assessment that impact from large commercial aircraft will not result in loss of core cooling or containment remains intact and spent fuel cooling or storage pool is intact; is silent on shutdown position
50.155	Mitigation Requirements Post Beyond Design Basis Event	X	X	X			Requires maintaining long-term recovery capability following beyond design basis event to preserve as many FSFs as possible
Appendices to Part 50							
Part 50 Appendix A General Design Criteria	Minimum Requirements for Adequate Protection	X	X	X	X		Each criterion is examined in greater detail later in the table. Note that programmatic and design capability requirements are both included in Appendix A.

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
Part 50—Appendix B	Quality Requirements for Important Plant Equipment				X		Programmatic requirements that govern design, procurement, and construction of all SSCs important to safety
Part 50—Appendix C	Requirements for Financial Reporting					X	
Part 50—Appendix E	Emergency Planning Requirements				X		Include programmatic requirements assurance of DID
Part 50—Appendix F	Fuel Reprocessing Plant Siting						NA
Part 50—Appendix G	Fracture Protection for Reactor Vessel—Integrity of Reactor Vessel	X		X			Governs integrity of reactor coolant pressure boundary and vessel to assure cooling capability and limit releases to containment
Part 50—Appendix H	Reactor Vessel Surveillance—Integrity of Reactor Vessel	X		X			See above
Part 50—Appendix I	ALARA Provisions—Effluents	X					Governs routine releases of radionuclide materials
Part 50—Appendix J	Containment Leak Testing for Integrity	X					Preserves integrity of containment within design limits for radionuclide retention
Part 50—Appendix K	ECCS Evaluation Models—Post Accident					X	Governs post-accident core cooing modelling and cooling evaluations; does not impose any technical requirements for LWRs with zirconium-based cladding
Part 50—Appendix Q	Site Suitability					X	
Part 50—Appendix R	Fire Protection Requirements				X		Programmatic fire protection requirements; Sections III.G, J, and O are adequate protection requirements for all LWRs

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
Part 50—Appendix S	Seismic Engineering Criteria	X	X	X			Applies seismic assessment requirements to SSCs important to safety; will impact all structures and components that support satisfaction of the FSFs
General Design Criteria							
Criterion 1	Requires design, fabrication, construction, and testing to quality standards commensurate with safety importance				X		Programmatic requirements govern all SSCs important to safety
Criterion 2	SSCs designed to withstand natural phenomena	X	X	X			Governs all SSCs important to safety
Criterion 3	SSCs designed to minimize effects of fires and explosions	X	X	X			Governs all SSCs important to safety
Criterion 4	SSCs designed to withstand all environmental conditions of operation and withstand dynamic effects of fluid discharges	X	X	X			Governs all SSCs important to safety
Criterion 5	Important SSCs may not be shared	X	X	X			Governs all SSCs important to safety
Criterion 6 – 9	Reserved						
Criterion 10	Core, coolant, control systems designed with margin to protect fuel limits, normal or transient conditions	X	X	X			Requires that multiple plant systems must operate to assure that fuel design limits are not exceeded during normal operation, including AOOs

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
Criterion 11	Core and coolant systems designed to assure that inherent negative feedback coefficient is preserved		X				Design must have means to compensate for reactivity increases
Criterion 12	Core and control systems designed to detect and suppress power oscillations		X				Control system must detect and suppress nuclear power oscillations; assures that power oscillations will not lead to core disruptive event or inability to shut down reactor
Criterion 13	Instrumentation required to measure fission process, integrity of reactor core, coolant boundary, and containment	X	X	X			Governs monitoring requirements for important plant processes and safety features
Criterion 14	Coolant boundary to be designed, fabricated, constructed, and tested to prevent leakage and rupture	X		X			Requires that pressure boundary be designed, constructed, fabricated, tested to assure extremely low probability of abnormal leakage, propagating failure, or gross rupture. Leakage or rupture could result in inability to maintain cooling that would result in releases of radionuclides to containment or environment.
Criterion 15	Coolant system and controls have margins to assure design conditions of pressure boundary not exceeded	X		X			Requires that reactor coolant and associated support, control, and protection systems be designed with margin to assure that design conditions of pressure boundary are not exceeded for normal operation or AOOs. Exceeding design conditions could result in inability to maintain cooling that would result in releases of radionuclides to containment or environment.

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
Criterion 16	Containment must be leak-tight to prevent release of radioactivity	X					Containment integrity last line of defense for limiting release of radionuclide materials
Criterion 17	Requirements for redundancy of electric power systems to minimize loss of electric power; accidents and normal operation	X	X	X			Governs electrical power system designs for systems and components with a safety nexus
Criterion 18	Inspection and testing of electric power systems				X		Programmatic requirements for inspection and testing of electrical power system designs for systems and components important to safety
Criterion 19	Control room exposure and capability requirements, prompt shutdown, place plant in safe shutdown conditions	X	X	X			Control room must remain operable following any plant upset condition; operator actions may be required to assure that all FSFs are satisfied
Criterion 20	Reactivity control and protection system designed to protect fuel, initiate actions important to safety		X				Control systems automatically initiate to protect fuel during AOOs
Criterion 21	Redundancy and independence of protection systems, high reliability		X				Control system must be highly reliable and in-service testable; no single failure defeats the protection system
Criterion 22	Protection systems design against natural phenomena, no loss of protection function		X				Control system is capable of operation following natural phenomena; requires redundancy and diversity of means

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
Criterion 23	Protection system fails in a safe plant state		X	X			Reactor protection systems could have multiple inputs from reactor protection to primary coolant systems that will actuate to protect the plant. Any failure must result in plant safe state.
Criterion 24	Control and protection systems need to be separate; interconnections between need be limited		X	X			Control systems are separate from protection systems; limit interactions that could result in loss of important safety functions affecting the reactor and primary system
Criterion 25	Reactivity control system malfunctions to not exceed fuel design limits		X				Malfunction of control systems do not result in reactivity excursions that would create operating conditions that exceed fuel design limits
Criterion 26	Two independent reactivity control systems are required, one system uses rods, other assures that fuel design limits not exceeded, one to hold reactor subcritical when cold		X				Two independent reactivity control means are provided. One shall use rods, one by other means.
Criterion 27	Combined capability of reactivity control system and poison addition to control reactivity changes to assure capability to cool the core		X				Requires alternatives to routine reactivity control systems to assure that injection of cold water during ECCS injection would not result in reactivity transient that would result in loss of core cooling

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
Criterion 28	Reactivity limits; post accidents should not impair capability to cool core	X	X	X			Requires control systems to limit rate and amount of potential reactivity addition so that resulting transient does not result in loss of primary coolant boundary and ability to cool the core, or result in a core disruption event that would prevent the ability to shut down the reactor when all control rods are fully inserted
Criterion 29	Reactivity control systems must perform in event of AOOs, extremely high reliability		X				Governs both protection and reactivity control systems; requires extremely high reliability for accomplishing safety functions following an AOO
Criterion 30	Coolant pressure boundary designed, fabricated, constructed, tested to highest quality standards	X		X			Requires components of pressure boundary to be designed, fabricated, erected, and tested to highest quality standards practical; means provided for detecting and locating source of any coolant leakage
Criterion 31	Reactor coolant boundary designed that when stressed behaves in nonbrittle manner	X		X			Requires pressure boundary be designed with margin to assure that when stressed boundary behaves in a non-brittle manner; probability of propagating failure is minimized; design reflects service temperatures and other conditions of boundary materials and uncertainties with materials, radiation effects, stress states, and size of material flaws
Criterion 32	Coolant boundary must have inspection and surveillance program				X		Programmatic requirements to permit inspection and testing to assess structural and leak-tight integrity; requires material surveillance program for vessel integrity

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
Criterion 33	Reactor coolant makeup required to protect against small breaks in piping so fuel limits preserved			X			Requires system to supply coolant makeup for protection against small breaks in pressure boundary; protect fuel limits from loss of cooling from small leakage to rupture of small pipes; system uses equipment that is used during normal operation
Criterion 34	A residual heat removal system is required; requires suitable redundancy, leak-tightness	X		X			Requires a system to remove residual heat, remove decay heat and other residual heat at a rate that protects fuel safety limits and design conditions of pressure boundary; suitable redundancy of design and leak detection, isolation, to assure that safety function occurs assuming single failure
Criterion 35	Abundant emergency core cooling is required; assume single failure, core cooling not disruptive	X		X			Requires a system to provide abundant emergency core cooling to transfer heat from core following any loss of coolant at a rate that fuel damage would not interfere with core cooling and metal water reaction is limited to negligible amounts; suitable redundancy to assure safety function is performed assuming a single failure-limit cladding failure and hydrogen generation limits releases from fuel
Criterion 36	Emergency core cooling system must be inspectable				X		Programmatic requirement that emergency core cooling system is designed to permit periodic inspection to assure integrity and capability of the system

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
Criterion 37	Emergency core cooling system must be testable; structural and leak-tight components				X		Programmatic requirement that design permits periodic pressure and functional testing to assure structural and leak-tight integrity; operability and performance of active components; operability of the system as a whole under design conditions; full operational sequence that brings system into operation
Criterion 38	System to remove heat from containment is required; suitable redundancy, leak-tightness, isolation capability	X		X			Requires system to remove heat from the containment, rapidly reduce containment pressure and temperature following any loss of coolant and maintain them acceptably low
Criterion 39	Containment heat removal system must be inspectable to assure integrity				X		Programmatic requirement that containment heat removal system to be inspected periodically to assure integrity and capability of heat removal systems
Criterion 40	Testing of containment heat removal system; leak-tightness and integrity				X		Programmatic requirements that the design permits periodic pressure and functional testing to assure structural and leak-tight integrity; operability and performance of active components; operability of the system as a whole—under design conditions—operational sequence that brings system into operation
Criterion 41	Containment atmosphere cleanup systems to control fission products released to containment , control hydrogen, assure containment integrity	X					Containment atmosphere cleanup systems to control fission products released to containment; assure containment integrity

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
Criterion 42	Inspection of containment atmosphere cleanup for integrity and capability				X		Programmatic requirements for inspection of containment atmosphere cleanup systems for integrity and capability
Criterion 43	Testing of containment atmosphere cleanup, integrity, leak-tightness, operability				X		Programmatic requirement to assure that cleanup system is leak-tight to minimize releases to environment
Criterion 44	Cooling system to transfer heat from important to safety equipment to ultimate heat sink, normal and accident conditions, redundancy, leak tight, assume single failure	X	X	X			Governs requirements for assuring heat sink is available for long-term cooling
Criterion 45	Cooling water system must be inspectable, structure, leak-tightness, operability				X		Programmatic requirements for inspection to assure heat sink is available for long-term cooling
Criterion 46	Cooling water system must be testable, structural, leak-tight, operability				X		Programmatic requirements for testing to assure heat sink is available for long-term cooling
Criterion 47 - 49	Reserved						
Criterion 50	Containment structure must be designed so that it can withstand internal pressures and temperatures and not exceed the design leak rate	X					Governs containment capability for withstanding pressure peaks and retaining radionuclides

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
Criterion 51	Fracture prevention of containment pressure boundary, propagating failure is minimized	X					Governs containment capability for withstanding pressure peaks and retaining radionuclides
Criterion 52	Containment must be tested at containment design pressure for leak-tightness, design leak rate validation	X					Verification of containment capability
Criterion 53	Inspection of containment areas, leak-tightness of penetrations				X		Programmatic requirements for inspection to assure containment capability
Criterion 54	Piping systems that penetrate containment must have leak detection, isolation capabilities, redundancy that reflects the importance to safety	X					Verification of containment capability
Criterion 55	Coolant pressure boundary penetrating containment isolation requirements	X					Prevents releases of radionuclide materials to the environment
Criterion 56	Containment atmosphere lines penetrating containment isolation requirements	X					Prevents releases of radionuclide materials to the environment
Criterion 57	Any line that penetrates containment other than GDC 55,56, isolation requirements	X		X			Prevents releases of radionuclide materials to the environment

Regulation	Description of Regulation and Summary of Safety Objective, when Appropriate	Retaining Radioactive Materials	Controlling Reactivity	Removing Heat from the Reactor and Waste Stores	Programmatic Requirement Supporting FSFs Assurance	Administrative Requirement	Rationale
Criterion 58-59	Reserved						
Criterion 60	Design must suitably control releases of gaseous and liquid effluents and handle solid wastes	X					Prevents and controls releases of radionuclide materials to the environment
Criterion 61	Radioactivity control during fuel storage and handling; testing, inspection, confinement, containment, heat removal, prevent dilution event	X	X	X	X		This criterion includes both programmatic and technical requirements that governs all aspects of spent fuel storage management. Therefore, all programmatic and technical columns are marked.
Criterion 62	Criticality during fuel storage and fuel handling		X				Inadvertent criticality in fuel storage or fuel handling system is prevented by physical systems or processes
Criterion 63	Monitoring fuel and waste storage, loss of heat removal and excess radiation levels	X		X			Requires systems to detect loss of residual heat removal capacity and excessive radiation levels and to initiate safety actions
Criterion 64	Monitoring radioactivity releases, gas and liquid in all plant modes	X					Monitoring radioactivity releases

5.0 BINNING PRINCIPAL DESIGN CRITERIA TO "WHAT," "WHEN," "HOW," AND "HOW WELL" CATEGORIES

Section 4 of this report presents evidence that the General Design Criteria (GDC) can be successfully mapped to one or more fundamental safety functions. Section 5 of this report assesses a binning of the GDC that will be used as part of the LMP-based affirmative safety case. The binning process provides evidence that the Principal Design Criteria (PDC) for an LWR-based design (which are derived from the 10 CFR 50 Appendix A GDC) are required to include capabilities (functions and SSCs) or features (system configuration or programs) that cover :

- **How** the design demonstrates that the FSFs are met?
- **How well** do these capacities need to be performed to provide reasonable assurance?

As stated in Section 1.2, a design's safety case provides answers to the following list of questions corresponding to the categories of "What," "When," "How," and "How Well" (WWHHW):

- **What** are the performance objectives for the FSFs?
- **When** do the FSF's performance objectives need to be demonstrated?
- **How** plant capabilities (functional and structural) demonstrate that the FSFs are met?
- **How well** do these capacities need to be performed to provide reasonable assurance?

It is noteworthy that in Appendix A to 10 CFR 50 for an LWR-based design, the PDC, where they are developed based on the GDC, are considered to:

> *". . . establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is,* ***structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public."***

As such, the framework of 10 CFR 50 Appendix A implicitly assumes that the performance objectives of FSFs are met when an LWR-based design's PDC fully comply with the GDC. Therefore, such a design's safety case does not require an evaluation against the performance objectives of the FSFs.

Each criterion of the GDC was examined to determine if it prescribed a required capability. If the GDC contained a prescribed capability, it was binned to the "How" category. If the GDC prescribed special treatment requirements, then the GDC was binned to the "How Well" category. Sometimes the GDC prescribed both capabilities and special treatment requirements. Those GDC were labeled as hybrids. When a hybrid GDC was present, the prescribed capabilities were captured in the summary list of GDC capabilities. Figure 1 is a graphical representation of the binning categories and where in the LMP methodology plant capabilities and special treatment requirements are developed.

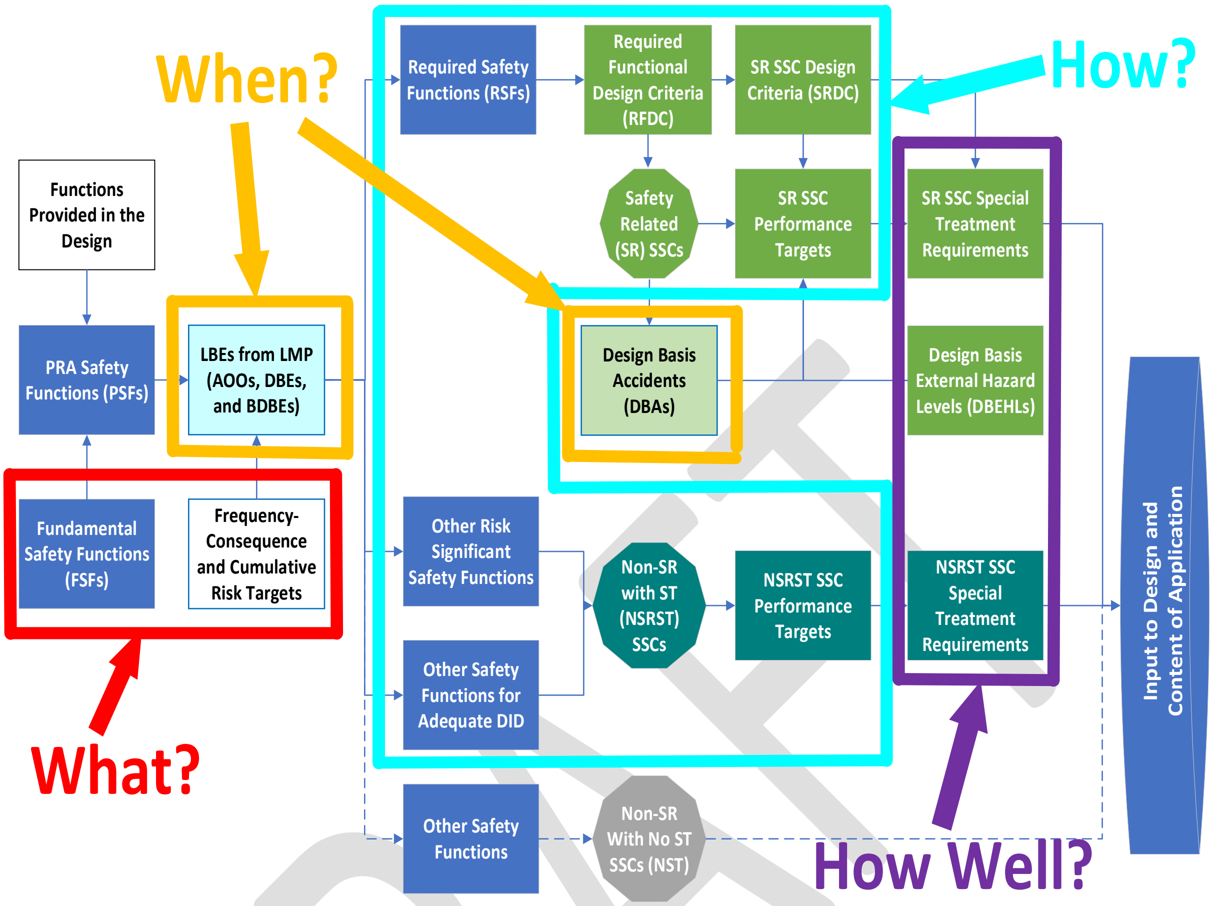

Figure 1. LMP Methodology and WWHHW Organization

Binning Summary

The results of this exercise at binning the GDC to the four categories produced some interesting insights. Using the approach outlined above, there are 26 GDC that contained principal design criteria (those that listed plant capabilities needed in the design and would include information that would be binned into the "How" category). The systems or structures in the "How" category would be classified as safety-related and would be the foundation for a finding that the technology would provide adequate assurance of public health and safety.

Using the LWR technology and the GDC as examples, the following systems or structures would be identified as performing some type of required safety function(s) and would then be classified as safety-related:

1. Reactor inherent protection (GDC 11)
2. Suppression of reactor power oscillations (GDC 12)
3. Instrumentation Systems (GDC 13)
4. Reactor Coolant Pressure Boundary (GDC 14)

5. Reactor Coolant System Design (GDC 15)
6. Reactor Containment and associated systems (GDC16)
7. Electric power systems (GDC 17)
8. Control Room (GDC 19)
9. Protective Systems functions (GDC 20)
10. Two independent reactivity control systems (GDC 26)
11. Reactor coolant makeup system (GDC 33)
12. Residual heat removal system (GDC 34)
13. Emergency core cooling system (GDC 35)
14. Containment heat removal system (GDC 38)
15. Containment atmosphere cleanup system (GDC 41)
16. Cooling water system (ultimate heat sink) (GDC 44)
17. Containment (GDC 50)
18. Piping systems penetrating containment (GDC 54)
19. Reactor coolant pressure boundary penetrating containment (GDC 55)
20. Primary containment isolation (GDC 56)
21. Closed system isolation valves (GDC 57)
22. Means to control releases of radioactive gases and liquids and handle solid wastes (GDC 60)
23. Fuel storage and handling and radioactivity control (GDC 61)
24. Prevention of criticality in fuel storage and handling (GDC 62)
25. Monitoring systems for fuel storage and handling areas (GDC 63)
26. Means for monitoring radioactive releases (GDC 64)

6.0 CONCLUSIONS

The mapping efforts presented in Table 1 of this report examined 157 items comprising regulation sections in 10 CFR 50, the General Design Criteria (Appendix A to 10 CFR 50), and other Part 50 appendices.

Regulations included in Table 1 are summarized below.

1. Regulation sections included in 10 CFR 50	89
2. Appendices to Part 50	13
3. General Design Criteria	55
Total regulation sections mapped from Part 50	157

The results of the mapping exercise determined that each of the technical requirements contained in the 157 regulation sections could be successfully mapped to one or more FSFs. Of note is that Part 50 contains a large number of regulation sections that establish procedural or administrative requirements. To assure that the mapping process was comprehensive, it was necessary to create an administrative category and to map the administrative requirements to that category.

A summary of the number of entries for each of the Table 1 categories is given below.

1. Retaining Radioactive Materials	50
2. Controlling Reactivity	30
3. Removing Heat from the Reactor and Waste Stores	38
4. Programmatic	30
5. Administrative or Procedural	63

The results of the mapping activity presented in this report highlight the importance that the NRC placed on removing heat from the fuel within the reactor, preserving the integrity of the primary coolant boundary and cooling capability, and limiting releases of any radioactive materials to the environment for LWR technologies. More importantly, the NRC regulatory emphasis easily translates to a demonstration that the use of FSFs can provide a satisfactory licensing surrogate for the set of prescriptive regulations in the existing Part 50 LWR-centric regulations. In addition, the mapping illustrates the importance of the programmatic requirements (comparable in magnitude to specific technical requirement categories) that assure that SSCs important to safety perform their required safety functions when required.

Binning Summary

The results of the binning exercise of the LWR GDC are:

1. Five GDC specify conditions other than a postulated accident for when the GDC apply (GDCs 10,15,19,29,60)
2. Nineteen GDC will bin wholly into the "How" category – these GDC specify that certain capabilities are to be provided and explain the safety functions to be performed.
3. Seven GDC are hybrid GDC in that they specify that certain capabilities are to be provided and they also provide certain special treatment requirements for those systems
4. Twenty-nine GDC can be wholly binned into the "How Well" category as they specify special treatment requirements for those systems.

The net result is that there are 26 GDC that contained principal design criteria (those that listed plant capabilities needed in the design) that using this binning process would be binned into the "How" category. The systems or structures in the "How" category would be classified as safety-related and would be the foundation for a finding that the technology would provide adequate assurance of public health and safety.

Applying the binning process to LWR technology and using the current GDC as PDC examples, the following systems or structures would be identified as performing some required safety function(s) within the LMP process and would be classified as safety-related:

1. Reactor inherent protection (GDC 11)
2. Suppression of reactor power oscillations (GDC 12)
3. Instrumentation Systems (GDC 13)
4. Reactor Coolant Pressure Boundary (GDC 14)
5. Reactor Coolant System Design (GDC 15)
6. Reactor Containment and associated systems (GDC16)
7. Electric power systems (GDC 17)
8. Control Room (GDC 19)
9. Protective Systems functions (GDC 20)
10. Two independent reactivity control systems (GDC 26)
11. Reactor coolant makeup system (GDC 33)
12. Residual heat removal system (GDC 34)
13. Emergency core cooling system (GDC 35)
14. Containment heat removal system (GDC 38)
15. Containment atmosphere cleanup system (GDC 41)
16. Cooling water system (ultimate heat sink) (GDC 44)
17. Containment (GDC 50)

18. Piping systems penetrating containment (GDC 54)
19. Reactor coolant pressure boundary penetrating containment (GDC 55)
20. Primary containment isolation (GDC 56)
21. Closed system isolation valves (GDC 57)
22. Means to control releases of radioactive gases and liquids and handle solid wastes (GDC 60)
23. Fuel storage and handling and radioactivity control (GDC 61)
24. Prevention of criticality in fuel storage and handling (GDC 62)
25. Monitoring systems for fuel storage and handling areas (GDC 63)
26. Means for monitoring radioactive releases (GDC 64)

7.0 REFERENCES

[1] NEI 18-04, Rev 0, "Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development," 2019.

[2] 10 CFR 50, Domestic Licensing of Production and Utilization Facilities.

[3] 10 CFR 52, Licenses, Certifications, and Approvals for Nuclear Power Plants.

[4] Southern Company Document Number SC-16166-100 Rev 0, "Definition of Fundamental Safety Functions for Advanced Non-Light Water Reactors," March 2020.

[5] NUREG-1860, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing," December 2007.

[6] INL/EXT-11-23216, "NGNP Project Regulatory Gap Analysis for Modular HTGRs," September 2011.

[7] OKLO-2018-R10-NP, Oklo Draft Guide 1353 Pilot Report, September 2018.

[8] Draft Regulatory Guide DG-1353, "Guidance for a Technology Inclusive, Risk-Informed, and Performance-Based Approach to Inform the Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors," April 2019.

Appendix A: Mapping of 10 CFR Part 50 GDC to "What," "When," "How," and "How Well" Categories

This appendix presents the mapping of the GDC to the four binning categories of "What," "When," "How," and "How Well." Some of the GDC can be wholly mapped directly to one binning category. Other GDC are referred to as blended GDC. Blended GDC are general design criteria that contain information that belongs in one or more of the binning categories. When blended GDC occur, the respective binning categories are illustrated within the text box for the blended GDC.

The colors of the text box borders are taken from the LMP methodology graphic presented in Section 5 of this report. For ease of reference, the color designations are repeated here:

1. The "What" category, represented in red, specifies the radiological performance objectives that the design must meet. Because the GDC do not specify the radiological performance objectives that a design must meet, there are no red text boxes around the listed GDC.
2. The "When" category, represented by yellow, specifies the events or accidents for which the performance objectives must be met. There are five instances within the listed GDC that specify that the GDC are applicable to AOOs, normal operation, or postulated accidents. The mapping given below does not highlight those GDC with color, but the specific GDC are listed in the binning summary.
3. The "How" category, represented in light blue, specifies the functions and plant capabilities that must be present in order that the radiological performance objectives can be met.
4. The "How Well" category, represented by purple, specifies the special treatment requirements both functional and programmatic that are necessary to provide the assurance that the plant functions and capabilities will perform as required.

The GDC were constructed to provide the minimum requirements for adequate protection for light water reactors. Recall that at the time the GDC were promulgated, the principal postulated accident of concern was a double-ended guillotine break of a primary system pipe, coincidental with the loss of office power, and a single failure, leading to a large radiological release to the containment and eventually to the environment.

In summary, the binning results are:

1. Five GDC specify conditions other than the MHA for when the GDC apply (GDCs 10, 15, 19, 29, 60).
2. Nineteen GDC will bin wholly into the "How" category – these GDC specify that certain capabilities are to be provided and explain the safety functions to be performed.
3. Seven GDC are hybrid GDC in that they specify both that certain capabilities are to be provided and the special treatment requirements for those systems.

4. Twenty-nine GDC can be wholly binned into the "How Well" category as they specify design requirements and special treatment requirements for those systems.

Mapping Assessment of General Design Criteria to WWHHW Categories

I. Overall Requirements

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

Criterion 2—Design bases for protection against natural phenomena. 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.

Criterion 3—Fire protection. 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 heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Criterion 4—Environmental and dynamic effects design bases. 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, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

Criterion 5—Sharing of structures, systems, and components. Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

II. Protection by Multiple Fission Product Barriers

Criterion 10—Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Criterion 11—Reactor inherent protection. The reactor core and associated coolant systems shall be designed so that in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Criterion 12—Suppression of reactor power oscillations. The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Criterion 13—Instrumentation and control. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Criterion 14—Reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Criterion 15—Reactor coolant system design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Criterion 16—Containment design. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Criterion 17—Electric power systems. An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a **single failure**. Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained. Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

Criterion 18—Inspection and testing of electric power systems. 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 the nuclear power unit, the offsite power system, and the onsite power system.

Criterion 19—Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under Part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.

III. Protection and Reactivity Control Systems

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

Criterion 21—Protection system reliability and testability. The protection system shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no **single failure** results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Criterion 22—Protection system independence. The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Criterion 23—Protection system failure modes. The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

Criterion 24—Separation of protection and control systems. The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Criterion 25—Protection system requirements for reactivity control malfunctions. The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Criterion 26—Reactivity control system redundancy and capability. Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Criterion 27—Combined reactivity control systems capability. The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Criterion 28—Reactivity limits. The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Criterion 29—Protection against anticipated operational occurrences. The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

IV. Fluid Systems

Criterion 30—Quality of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Criterion 31—Fracture prevention of reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

Criterion 32—Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Criterion 33—Reactor coolant makeup. A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary.

The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Criterion 34—Residual heat removal. A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 35—Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a **single failure**.

Criterion 36—Inspection of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

Criterion 37—Testing of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the system, 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 applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 38—Containment heat removal. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a **single failure**.

Criterion 39—Inspection of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

Criterion 40—Testing of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 41—Containment atmosphere cleanup. Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a **single failure**.

Criterion 42—Inspection of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

Criterion 43—Testing of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves 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 sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Criterion 44—Cooling water. A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a **single failure.**

Criterion 45—Inspection of cooling water system. The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

Criterion 46—Testing of cooling water system. The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and the performance of the active components of the system, 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 for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

V. Reactor Containment

Criterion 50—Containment design basis. The reactor 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 any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (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.

Criterion 51—Fracture prevention of containment pressure boundary. The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic 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 material 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.

Criterion 52—Capability for containment leakage rate testing. The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

Criterion 53—Provisions for containment testing and inspection. The reactor containment 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 which have resilient seals and expansion bellows.

Criterion 54—Piping systems penetrating containment. Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Criterion 55—Reactor coolant pressure boundary penetrating containment. Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment 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.
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 assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for in-service 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 environment.

Criterion 56—Primary containment isolation. Each line that connects directly to the containment atmosphere and penetrates primary reactor containment 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.

Criterion 57—Closed system isolation valves. Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which 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.

VI. Fuel and Radioactivity Control

Criterion 60—Control of releases of radioactive materials to the environment. The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Criterion 61—Fuel storage and handling and radioactivity control. The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Criterion 62—Prevention of criticality in fuel storage and handling. Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Criterion 63—Monitoring fuel and waste storage. Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Criterion 64—Monitoring radioactivity releases. Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

Footnotes to Appendix A

[1] Further details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development.

[2] **Single failures** of passive components in electric systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.