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Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing

Appendices A through L

Office of Nuclear Regulatory Research

U.S.NRC
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Appendices A through L

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M. Drouin, NRC Project Manager

Office of Nuclear Regulatory Research

ABSTRACT

The purpose of this NUREG is to establish the feasibility of developing a risk-informed and performance-based regulatory structure for the licensing of future nuclear power plants (NPPs). As such, this NUREG documents a “Framework” that provides an approach, scope and criteria that could be used to develop a set of requirements that would serve as an alternative to 10 CFR 50 for licensing future NPPs; however, this Framework is not the entire process. It is an initial phase in is to demonstrate the feasibility of such a concept, recognizing that for full implementation there will be outstanding programmatic, policy, and technical issues to be resolved. As such, this feasibility study does not represent a staff position, but rather a significant piece of research. The second phase, which involves implementation, is comprised of several, iterative steps: resolution of issues, development of draft requirements and regulations, pilots and tests, and rulemaking.

The information contained in this NUREG is intended for use by the US Nuclear Regulatory Commission (NRC) staff in developing requirements applicable to the licensing of commercial NPPs. Similar to 10 CFR 50, it covers the design, construction and operation phases of the plant lifecycle up to and including the initial stages of decommissioning (i.e., where spent fuel is still stored on-site). It covers the reactor and support systems. Fuel handling and storage are not addressed, but rather would be considered as part of implementation. The approach taken is one that integrates deterministic and probabilistic elements and builds upon recent policy decisions by the Commission related to the use of a probabilistic approach and mechanistic radioactive source terms in establishing the licensing basis.

At the highest level, the Framework has been developed from the top down with the safety expectation that future NPPs are to achieve a level of safety at least as good as that defined by the Quantitative Health Objectives in the Commission’s 1986 Safety Goal Policy Statement. Criteria are then developed that utilize an integrated deterministic and probabilistic approach for defining the licensing basis and safety classification. Implementation of these criteria would require a design specific probabilistic risk assessment and would result in a design specific licensing basis. Defense-in-depth remains a fundamental part of the requirements development process and has as its purpose applying deterministic principles to account for uncertainties. Defense-in-depth has been defined as an element in NRC’s safety philosophy that is used to address uncertainty by employing successive measures, including safety margins, to prevent or mitigate damage if a malfunction, accident, naturally or intentional caused event occurs. The approach taken in the Framework continues the practice of ensuring that the allowable consequences of events are matched to their frequency such that frequent events are to have very low consequences and less frequent events can have higher consequences. This is expressed in the form of a frequency-consequence curve. The allowable consequences are based upon existing dose limits, and the associated frequencies are based on guidance contained in International Commission on Radiological Protection 64 and engineering judgment.

Part of the process involves development of guidance to be used for actually writing the requirements. This guidance addresses writing the requirements in a performance-based fashion, incorporating lessons learned from past experience, and utilizing existing requirements and guidance, where practical. The guidance also ensures that the probabilistic process for establishing the licensing basis are incorporated. All of the above are integrated and results in a set of potential requirements which serve to illustrate and establish the feasibility of developing a risk-informed and performance-based licensing approach.

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FOREWORD

The Commission, in its Policy Statement on Regulation of Advanced Nuclear Power Plants, stated its intention to "improve the licensing environment for advanced nuclear power reactors to minimize complexity and uncertainty in the regulatory process." The staff noted in its Advanced Reactor Research Plan to the Commission, that a risk-informed regulatory structure applied to license and regulate advanced (future) reactors, regardless of their technology, could enhance the effectiveness, efficiency, and predictability (i.e., stability) of future plant licensing. Therefore, a need was identified for a "Framework" to guide the development of a risk-informed and performance-based approach for future plant licensing for advanced (non-light water) reactors. This NUREG report satisfies that need.

The development of a risk-informed and performance-based regulatory structure for the licensing of diverse reactor designs is a complicated and multi-phase program. Before thoroughly embarking on implementing such an initiative, it is prudent to understand whether such a venture is feasible. The first phase, which is the Framework, is to demonstrate the feasibility of such a concept. The second phase, which involves implementation and would only be pursued upon Commission direction, is comprised of several, iterative steps: resolution of issues, development of draft requirements and regulations, pilots and tests, and rulemaking.

This NUREG report documents one approach to establish the feasibility for development of a risk-informed and performance-based process for the licensing of future nuclear power plants. Part of this documentation is the identification of the programmatic, policy, and technical issues that would need to be addressed by the staff for implementation of such an approach.

This work is intended to assess the feasibility of alternative licensing structures for future designs. It does not represent staff positions with respect to the licensing of current or new plants.

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EXECUTIVE SUMMARY

The purpose of this NUREG is to establish the feasibility of developing a risk-informed and performance-based regulatory structure for the licensing of future (advanced non-light water reactor (LWR)) nuclear power plants (NPPs). This NUREG documents a “Framework” that provides an approach, scope and criteria that could be used to develop a set of requirements that would serve as an alternative to 10 CFR 50 for licensing future NPPs. This alternative to 10 CFR 50 would have the potential following advantages:

- It would require a broader use of design specific risk information in establishing the licensing basis, thus better focusing the licensing basis, its safety analysis and regulatory oversight on those items most important to safety for that design.

- It would stress the use of performance as the metric for acceptability, thus providing more flexibility to designers to decide on the design factors most appropriate for their design and facilitate the development of an U.S. Nuclear Regulatory Commission (NRC) reactor oversight program that focuses on safety performance.

- It could be written to be applicable to any reactor technology, thus avoiding the time consuming and less predictable process of reviewing non-light-water reactor (LWR) designs against the LWR-oriented 10 CFR 50 regulations, which requires case-by-case decisions (and possible litigation) on what 10 CFR 50 regulations are applicable and not applicable and where new requirements are needed.

This Framework, as such, is not the entire process. It is an initial phase in developing such a regulatory structure. The development of a risk-informed and performance-based regulatory structure for the licensing of diverse reactor designs would be a complicated and multi-phase program. Before thoroughly embarking on implementing such an initiative, it is prudent to understand whether such a venture is feasible. Therefore, the first phase, and major objective of this report, is to demonstrate the feasibility of such a concept, recognizing that for full implementation there would be outstanding programmatic, policy, and technical issues to be resolved. The second phase, which would involve implementation, is comprised of several, iterative steps: resolution of issues, development of draft requirements and regulations, pilots and tests, and rulemaking. Figure ES-1 shows this multi-phased program

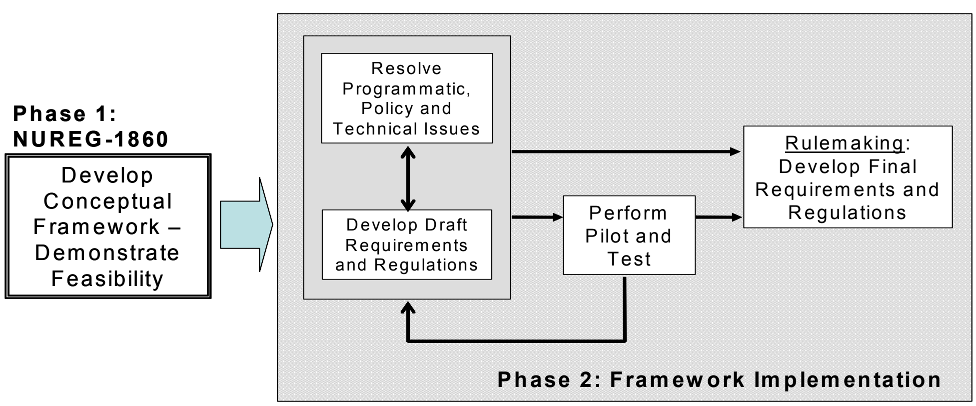

Figure ES-1 Development of Regulatory Structure for Future Plant Licensing

This first phase, feasibility of a conceptual framework, is the focus and objective of this report. The technical objectives this NUREG is intended to achieve, in establishing feasibility, are:

- be risk-informed
- be performance-based
- incorporate defense-in-depth
- provide flexibility.

Achievement of these objectives should result in a more effective, efficient and stable licensing process for advanced non-LWR designs.

Similar to 10 CFR 50, it covers the design, construction and operation phases of the plant lifecycle up to and including the initial stages of decommissioning (i.e., where spent fuel is still stored on-site). It covers the reactor and support systems. Fuel handling and storage are not addressed, but rather would be considered as part of implementation. The technical basis and process described in this NUREG are directed toward the development of a stand alone set of requirements (containing technical as well as administrative items) that would be compatible and interface with the other existing parts of 10 CFR (e.g., Part 20, 51, 52, 73, 100, etc.) just as 10 CFR 50 is today. The approach taken in developing the technical basis and process is one that integrates deterministic and probabilistic elements and builds upon recent policy decisions by the Commission related to the use of a probabilistic approach and mechanistic radioactive source terms in establishing the licensing basis.

At the highest level, the Framework has been developed from the top down with the safety expectation that future NPPs are to achieve a level of safety at least as good as that defined by the Quantitative Health Objectives (QHOs) in the Commission's 1986 Safety Goal Policy Statement. This approach is consistent with the Commission's 1986 Policy Statement on Advanced Reactors which states that the Commission expects advanced reactor designs will comply with the Commission's Safety Goal Policy Statement, and is discussed further in Chapter 3. Possible criteria are then developed, consistent with the QHOs, that utilize a probabilistic approach for defining the licensing basis (discussed later). Implementation of these criteria would require a design specific probabilistic risk assessment (PRA) and would result in a design specific licensing basis.

Defense-in-depth remains a fundamental part of the requirements development process and has as its purpose applying deterministic principles to account for uncertainties. Defense-in-depth is discussed in Chapter 4 and has been defined as an element in NRC's safety philosophy that is used to address uncertainty by employing successive measures, including safety margins, to prevent or mitigate damage if a malfunction, accident, naturally or intentional caused event occurs. The defense-in-depth approach taken in this NUREG calls for:

- providing multiple lines of defense (called protective strategies) against off-normal events and their consequences which represent a high level defense-in-depth structure; and

- the application of a set of defense-in-depth principles to each protective strategy that result in certain deterministic criteria to account for uncertainties (particularly completeness uncertainties).

The protective strategies, discussed in Chapter 5, address accident prevention and mitigation and consist of the following:

- physical protection (provides protection against intentional acts);

- stable operation (provides measures to reduce the likelihood of challenges to safety systems);

- protective systems (provides highly reliable equipment to respond to challenges to safety);

- barrier integrity (provides isolation features to prevent the release of radioactive material into the environment); and

- protective actions (provides planned activities to mitigate any impacts due to failure of the other strategies).

These protective strategies, in effect, provide for successive lines of defense, each of which would need to be included in the design.

The defense-in-depth principles, discussed in Chapter 4, would require designs to:

- provide measures against intentional as well as inadvertent events;

- provide accident prevention and mitigation capability;

- ensure key safety functions are not dependent upon a single element of design, construction, maintenance or operation;

- ensure uncertainties in equipment and human performance are accounted for and appropriate safety margins provided;

- provide alternative capability to prevent unacceptable releases of radioactive material to the public; and

- be sited at locations that facilitate protection of public health and safety.

As discussed earlier, a set of probabilistic criteria (Chapter 6) have been developed consistent with the Safety Goal QHOs that address:

- overall plant risk and the use of risk-information in design, construction and operations;
- allowable consequences of event sequences versus their frequency;
- selection of event sequences which to be considered in the design;
- safety classification of equipment; and
- security performance standards.

These criteria would also replace the single failure criterion, unless imposed as a defense-in-depth consideration. The approach taken in the Framework continues the practice of ensuring that the allowable consequences of events (defined in Chapter 6) are matched to their frequency such that frequent events are to have very low consequences and less frequent events can have higher consequences. This is expressed in the form of a frequency-consequence (F-C) curve as discussed in Chapter 6. The allowable consequences are based upon existing dose limits, and the associated frequencies are based on guidance contained in International Commission on Radiological Protection 64 and engineering judgment.

Certain event sequences (defined in Chapter 6) from the design specific PRA are chosen for use in establishing plant design parameters for safe operation and equipment safety classification. These events are called licensing basis events (LBEs) and are sequences from the PRA that have to meet stringent acceptance criteria related to the F-C curve and additional deterministic criteria that depend on three broad ranges of accident frequency:

frequent	$\geq$	10^{-2}/yr		
infrequent	<	10^{-2}/yr but	$\geq$	10^{-5}/yr
rare	<	10^{-5}/yr but	$\geq$	10^{-7}/yr

Chapter 6 provides additional descriptions of the event categories, the LBE selection process , acceptance criteria, analysis guidelines and additional discussion on the safety classification process.

As discussed above, risk assessment would have a more prominent and fundamental role in the licensing process than it does today under 10 CFR 50, since the risk assessment would be an integral part of the design process and licensing analysis. Because of this more prominent use of PRA, the Framework is considered fully risk-informed. Therefore, a high level of confidence would be needed in the results of the risk assessment used to support licensing. In addition, under this risk-informed licensing approach, the risk assessment would need to be maintained up to date over the life of the plant, since it would be an integral part of decision-making with respect to operations (e.g., maintenance, plant configuration control), plant modifications, and maintaining the licensing basis up to date (e.g., assessing the impact of plant operating experience, modifications, etc. on items such as safety classification, LBEs, etc.). Possible guidance on the scope and technical acceptability of the risk assessment needed to support this licensing approach is provided in Chapter 7.

Chapter 8 describes the process for developing potential requirements consistent with the guidance in Chapters 3 through 7. The process for identifying the potential requirements begins with the protective strategies. Each one is examined with respect to what are the various threats or challenges that could cause the strategy to fail. These challenges and threats are identified using a logic tree to perform a “systems analysis” of the strategy to identify potential failures. The defense-in-depth principles are then applied to each protective strategy. Defense-in-depth measures are identified which are incorporated into the potential requirements to help prevent protective strategy failure. This approach forms the process for the selection of “topics.” Potential hypothetical requirements are then identified for each topic.

Part of the process involves development of guidance that would be used for actually writing the requirements. This guidance addresses writing the requirements in a performance-based fashion, incorporating lessons learned from past experience, and utilizing existing requirements and guidance, where practical. The guidance also would ensure that the probabilistic process for establishing the licensing basis are incorporated. All of the above are integrated and results in a set of potential requirements which serve to illustrate and establish the feasibility of developing a risk-informed and performance-based licensing approach.

A set of potential requirements is provided in Appendix J. A few examples include the following:

- **Potential Design Requirement #2: Criteria for Selection of the Licensing Basis**

  “Event sequences from the design specific PRA which needs to be considered in the licensing analysis needs to be categorized as follows:

- frequent $\geq 10^{-2}$/reactor year (ry) (mean frequency)
- infrequent $< 10^{-2}$/ry but $\geq 10^{-5}$/ry (mean frequency)
- rare $< 10^{-5}$/ry but $\geq 10^{-7}$/ry (mean frequency)

Within each of these categories, the applicant/licensee need to designate those sequences of each event type (e.g., loss of coolant accidents, external events, etc.) with the largest consequences as Licensing Basis Events (LBEs) which need to meet the acceptance criteria in Design Requirement #3.

A postulated LBE for plant siting purposes needs to be selected in accordance with and meet the acceptance criteria in Design Requirement #8."

(This potential requirement does not have an equivalent in 10 CFR Part 50.)

- **Potential Design Requirement #27: Control Room Design**

  "The main control room needs to be designed with sufficient shielding and atmospheric control to ensure habitability by control room personnel for all accident sequences that have a frequency greater than 10^{-7}/ry (mean value). Habitability needs to encompass assuring the dose to control room operating personnel does not exceed 5 rem for the duration of the accident and that hazardous chemicals are not allowed entry in sufficient concentrations to affect the health and safety of control room personnel.

  The control room needs to have sufficient instrumentation, control and communication capability to allow all safety significant functions to be performed from this location."

  (This potential requirement would be the equivalent to GDC #19.)

- **Potential Design Requirement #29: Reactor Core Flow Blockage and Bypass Prevention**

  "Each reactor design needs to provide measures to prevent bypass and blockage of flow through the reactor core that is sufficient to cause localized fuel damage."

  (This potential requirement does not have an equivalent in 10 CFR Part 50.)

A completeness check was also made by comparing the draft example requirements to other safety requirements documents (e.g., International Atomic Energy Agency (IAEA) Standards, 10 CFR 50). The results of the completeness check are discussed in Chapter 8, and generally concludes that the topics identified are reasonably complete.

In addition, there are a number of programmatic, policy and open technical issues that would need to be resolved if, and when, a decision is made to pursue Framework implementation. These issues are described in Appendix C. The programmatic issue addresses the manner in which, if decided by the Commission, the Framework should be implemented (e.g., technology-neutral versus technology-specific, rule-making versus design-specific). The policy issues, for example, include such items as level of safety (e.g., acceptability of using the QHOs as the level of safety new plants are to achieve); and integrated risk (e.g., apply the QHOs on a per reactor or per site basis). The technical issues, for example, would involve such items as use of a complementary cumulative distribution function as an additional risk criterion; assessment of environmental protection; development of risk objectives subsidiary to the QHOs addressing accident prevention and mitigation; and development of risk-importance measures for non-LWRs and guidance for their use. However, the fact that a number of open items remain does not detract from the validity of the technical information contained in this document.

Finally, Chapter 9 discusses the conclusions and steps needed if, and when, the requirements resulting from application of the Framework would be implemented and used in plant licensing.

In summary, this NUREG has met the objectives and established the feasibility of developing a risk-informed and performance-based approach for future plant licensing. This conclusion is based upon the successful development of risk criteria that would be implemented using design-specific risk information, integration of probabilistic and deterministic (e.g., defense-in-depth) elements, demonstration of the LBE selection and safety classification process, development of potential requirements and the results from the check against other requirements documents.

In addition to the resolution of programmatic policy and technical issues described above, the following steps would also need to be taken to fully implement the Framework:

- completion of requirements development,
- development of implementing guidance,
- pilot testing on an actual reactor design,
- reactor oversight program development, and
- rule-making, if necessary.

This NUREG report documents one approach to establish the feasibility for development of a risk-informed and performance-based process for the licensing of future nuclear power plants. Part of this documentation is the identification of the programmatic, policy, and technical issues that would need to be addressed by the staff for implementation of such an approach.

This work is intended to assess the feasibility of alternative licensing structures for future designs. It does not represent staff positions with respect to the licensing of current or new plants.

ACKNOWLEDGMENTS

This report documents the feasibility of developing a risk-informed and performance-based regulatory structure for the licensing of future nuclear power plants. It documents a "Framework" that provides an approach, scope and criteria that could be used to develop a set of requirements that would serve as an alternative to 10 CFR 50 for licensing future NPPs.

Overall management and technical leadership of the project was provided by: Mary Drouin

The principal technical authors[1] of this report, by alphabetic order, were:

Dennis Bley (Buttonwood Consulting)
Mary Drouin (NRC)
Tom King (ISL)
Jeffrey LaChance (SNL)
John Lehner (BNL)
Bruce Mrowca (ISL)
Vinod Mubayi (BNL)

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Benjamin Beasley
Les Cupidon
Farouk Eltawila
Mirela Gavrilas
Eric Haskins (independent consultant)
Alan Kuritzky
Inn Kim (formerly of ISL)
John Monninger
Bruce Musico
Gareth Parry
Trevor Pratt (BNL)
Joshua Reinert (ISL)
Stuart Rubin
Martin Stutzke
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Colleen Amoruso (ISL)
Danielle Burnette
Cheryl Conrad (BNL)
Jean Frejka (BNL)
Michelle Gonzalez
Paul Kleene
Tammy Pfiester (ISL)
Lauren Killian
Katherine Orta (BNL)
Brian Wagner

[1]ISL – Information Systems Laboratory, SNL – Sandia National Laboratories, BNL – Brookhaven National Laboratory

ACRONYMS AND ABBREVIATIONS

ACR	Advanced CANDU Reactor
ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
ANPR	Advance Notice of Proposed Rulemaking
ANS	American Nuclear Society
AO	Abnormal Occurrence
AOO	Anticipated Operational Occurrences
ASME	American Society of Mechanical Engineer
ATWS	Anticipated Transient Without Scram
BDBT	Beyond Design Basis Threat
CCDF	Complementary Cumulative Distribution Function
CCFP	Conditional Containment Failure Probability
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CLB	Current Licensing Basis
CO	Carbon Monoxide
CO_2	Carbon Dioxide
CP	Construction Permit
CPEF	Conditional Probability of Early Fatality
CPLF	Conditional Probability of Latent Fatality
DCF	Dose Conversion Factor
DBA	Design Basis Accidents
DBT	Design Basis Threat
EAB	Exclusion Area Boundary
ECC	Emergency Core Cooling
ECI	Emergency Coolant Injection
EF	Early Fatality
EIS	Environmental Impact Statement
EP	Emergency Preparedness
EPA	Environmental Protection Agency
EXF[1]	1 rem Exceedance Frequency
F-C	Frequency Consequence
FSAR	Final Safety Analysis Report
F-V	Fussell-Vesely
GDC	General Design Criteria
GFR	Gas-cooled Fast Reactor
GPRA	Government Performance and Results Act
GTCC	Greater than Class C
HAZOP	Hazard and Operability Analysis
HEU	Highly Enriched Uranium
HLR	High Level Requirement
HLW	High Level Waste
HSE	Health and Safety Executive
HTGR	High Temperature Gas-cooled Reactor
IAEA	International Atomic Energy Agency
I&C	Instrumentation and Control
ICRP	International Commission on Radiation Protection
IEEE	Institute of Electrical and Electronics Engineers
IER	Individual Early Risk
ILR	Individual Late Risk

ACRONYMS AND ABBREVIATIONS (continued)

IM	Importance Measure or Measures
ISFSI	Independent Spent Fuel Storage Installation
ISGTR	Induced Steam Generator Tube Rupture
LBE	Licensing Basis Events
LERF	Large Early Release Frequency
LEU	Low Enriched Uranium
LF	Latent Fatality
LFR	Lead-cooled Fast Reactor
LLRF	Large Late Release Frequency
LMR	Liquid Metal-cooled Reactor
LPZ	Low Population Zone
LOCA	Loss of Coolant Accidents
LPZ	Low Population Zone
LTC	Long Term Cooling
LWR	Light Water Reactors
MOX	Mixed Oxide
MSR	Molten Salt Reactor
MW_e	Mega-watt Electric
NDE	Non Destructive Examination
NEPA	National Environmental Policy Act
NERI	Nuclear Energy Research Initiative
NGNP	Next Generation Nuclear Plant
NPP	Nuclear Power Plant
NRC	U.S. Nuclear Regulatory Commission
OL	Operating License
PAG	Protective Action Guidelines
PBMR	Pebble Bed Modular Reactor
PCT	Peak Cladding Temperature
PRA	Probabilistic risk assessments
PSAR	Preliminary Safety Analysis Report
Pu	Plutonium
QA	Quality Assurance
QC	Quality Control
QUO	Quantitative Health Objectives
RAW	Risk Achievement Worth
RCCS	Reactor Cavity Cooling System
RCS	Reactor Coolant System
RI/PB	Risk-Informed and Performance-Based
ROP	Reactor Oversight Process
SAMDA	Severe Accident Mitigation Design Alternative
SAR	Safety Analysis Report
SBO	Station Blackout
SCWR	Super Critical Water Reactor
SFR	Sodium-cooled Fast Reactor
SGTR	Steam Generator Tube Rupture
SNM	Special Nuclear Material
SRM	Staff Requirements Memorandum
SSC	Systems, Structures and Components
TEDE	Total Effective Dose Equivalent
TP	Total Population
U.K.	United Kingdom
VHTR	Very High Temperature Reactor

APPENDIX A
SAFETY CHARACTERISTICS OF ADVANCED REACTORS

A. SAFETY CHARACTERISTICS OF ADVANCED REACTORS

A.1 Introduction

The purpose of this appendix is to provide some examples of the variation in safety characteristics found among proposed new advanced reactor designs. In developing a technology-neutral framework, it is important to recognize that the safety approaches to the design employed by new reactors may be fundamentally different than those of light water reactors (LWRs), for which the current regulations were developed. These fundamental differences significantly influence the way in which the protective strategies are used to implement reactor-specific designs. Differences include: the selection of materials for the basic reactor components, methods and procedures for performing various safety functions, safety approaches to the design and arrangement of barriers, and for the protection of the barriers. These differences in strategies yield different numbers and types of systems, structures, and components (SSCs) needed to perform a set of safety functions that may be uniquely characterized for each reactor type. The safety functions may be unique in the sense that they are influenced by the inherent features of the reactor concept and the way these features interact with the barriers to the transport of radionuclides during accidents and event sequences. Indeed, the nature of the accident progression and physical and chemical processes that dictate the resulting source term are greatly influenced by the inherent reactor features as well as the details of the design.

The range of reactor types that are envisioned for the application of this technology-neutral, risk-informed and performance-based framework include advanced LWR and CANDU reactors, modular high temperature gas cooled reactors (HTGRs)[2], liquid metal-cooled reactors (LMRs), and other reactor concepts defined in the Department of Energy's Generation IV Reactor Program which covers various gas, lead, and sodium cooled fast reactors, the molten salt reactor (MSR), super critical water reactor (SCWR) and the very high temperature gas-cooled reactor (VHTR). This set of reactors exhibits fundamentally different characteristics than current LWRs, including different inherent features for the reactor fuel, moderator, and coolant, as well as different strategies for arranging barriers for the containment of radioactive material.

A.2 Differences in Approach to Protective Strategies

The five protective strategies: Physical Protection, Stable Operation, Protective Systems, Barrier Integrity, and Accident Management, establish the high level structure that, if followed, can systematically result in requirements for safe nuclear power plant design, construction, and operation. These protective strategies are generically applicable to all existing and new reactors and map to all elements modeled in nuclear power plant safety assessments. However, the nature of how these strategies are deployed for new reactor technologies is reactor-specific and may depart substantially from current U.S. LWR practice. Table A-1 presents examples of technology specific safety issues which the protective strategies need to address.

[2] A modular HTGR is defined here as a graphite moderated, helium cooled reactor using coated particle fuel, a core outlet helium temperature during normal operation of at least 700°C, and a capability for passive decay heat removal. Examples of modular HTGRs include the MHTGR, GT-MHR, and PBMR.

A. Safety Characteristics

Table A-1 Examples of technology-specific safety issues which the Protective Strategies need to address

Reactor Technology	Protective Strategies				
	Physical Protection	**Stable Operations**	**Protective Systems**	**Barrier Integrity**	**Protective Actions**
• **Gas-Cooled**	• On-line refueling implications for theft or diversion	• High temperature materials behavior and design codes and standards: - cracking - creep - fatigue - effect of coolant impurities - embrittlement	• Plant response to: - reactivity insertions - loss of coolant - loss of power	• Capability to accommodate: - air ingress - water ingress - security related events	• Desire for reduction in EP
		• Fuel performance: - steady state - reactivity transient - decay heat	• EQ	• In-service inspection techniques	• Staffing
		• Ensuring quality of fresh fuel	• Long term behavior of passive systems		• Source Terms
		• Equipment reliability	• Leak before break (i.e., no LB LOCA)		
		• Graphite behavior and design codes and standards: - strength - cracking - shrinkage - swelling	• H_2 production (VHTR)		
• **Water-Cooled**: - ALWR - SCWR		• Materials behavior: - cracking - effect of coolant impurities - fatigue - embrittlement	• Plant response to: - reactivity insertions - loss of coolant - loss of power	• Prevention of RPV rupture: - PTS - other?	• Desire for reduction in EP

Table A-1 Examples of technology-specific safety issues which the Protective Strategies need to address

Reactor Technology	Protective Strategies				
	Physical Protection	Stable Operations	Protective Systems	Barrier Integrity	Protective Actions
		• Fuel performance: - steady state - reactivity transient - decay heat			• Staffing
• **Heavy-Water:** - ACR - APHWR	• On-line refueling implications for theft or diversion	• Pressure tube integrity	• Plant response to: - reactivity insertions - loss of coolant - loss of power • Fuel-coolant/ moderator interaction (callandria over-pressure) • Coolant void coefficient	• Capability to accommodate: - fuel-coolant interaction - security-related events	
• **Sodium-Coded**	• Pool versus loop design	• Materials behavior and design codes and standards: - thermal stress - cracking - carbon transfer - nitriding - creep - fatigue - swelling - embrittlement	• Plant response to: - reactivity insertions - loss of power	• Capability to accommodate: - Na spills - Security related events - Fuel-coolant interaction - Recriticality	• Desire for reduction in EP

Table A-1 Examples of technology-specific safety issues which the Protective Strategies need to address

Reactor Technology	Protective Strategies				
	Physical Protection	Stable Operations	Protective Systems	Barrier Integrity	Protective Actions
		• Fuel performance: - metal fuel - oxide fuel - run beyond clad breach - grid spaces versus wire wrapped fuel pins - reactivity transient - actinide burning	• Sodium/ water reaction	• In-service inspection techniques	• Staffing
		• Prevention of loss of coolant	• Fuel-coolant interaction		• Source terms
		• Flow blockage prevention: - sodium freezing - loose material	• Sodium leak detection: - leak before break (i.e., no LB LOCA)		
			• Sodium spills: - fires - reaction with concrete		
			• Prevention of control-rod hydraulic lifting during refueling		
			• Sodium void coefficient		
			• Sodium activation		

Table A-1 Examples of technology-specific safety issues which the Protective Strategies need to address

Reactor Technology	Protective Strategies				
	Physical Protection	**Stable Operations**	**Protective Systems**	**Barrier Integrity**	**Protective Actions**
• **Lead Cooled**		• Materials behavior and design codes and standards: - thermal stress - cracking - effect of coolant impurities - carbon transfer - nitriding - creep - fatigue - swelling - embrittlement	• Plant response to: - reactivity insertions - loss of power	• Capability to accommodate: - Pb spills - security related events - fuel-coolant interaction - recriticality	• Desire for reduction in EP
		• Fuel performance: - nitride fuel - metal fuel - actinide burning	• Pb-water reaction	• In-service inspection techniques	• Staffing
		• Prevention of loss of coolout	• Fuel-coolant interaction		• Source Term
		• Flow blockage prevention: - Pb freezing - loose material	• Pb leak detection		
			• Pb spills: - reaction with concrete		
			• Void co-efficient		
			• Po generation		

A.3 Safety Characteristics of the New Advanced Reactors

The safety characteristics of the new reactors can take many forms. They can include:

- Characteristics of inherent properties of core, fuel, moderator, and coolant

- Characteristics of the radioactive material sources (including multiple reactors and non-core related sources)

- Characteristics of radionuclide transport barriers, including:
  - Fuel elements barrier
  - Coolant pressure boundary
  - Reactor building boundary
  - Site selection

- Characteristics of safe stable operating and shutdown states

- Characteristics of the safety functions and success criteria and the design features and SSCs that provide safety functions, including:
  - Inherent safety features
  - Engineered safety feature SSCs
    - Active engineered safety features
    - Passive engineered safety features

The inherent reactor characteristics are fundamental to defining how the reactor behaves in response to disturbances. The inherent reactor characteristics are also those that are fundamental to defining how reactor concepts differ from each other.

The sections below give a brief overview of the safety characteristics of seven new reactor designs to illustrate the variation found in such characteristics. The seven designs are: the pebble bed modular reactor (PBMR), the Advanced CANDU Reactor (ACR) 700, and five Generation IV reactors. The five Gen IV designs are: Very-High-Temperature Reactor (VHTR), Supercritical Water-Cooled Reactor (SCWR), Gas-Cooled Fast Reactor (GFR), Sodium-Cooled Fast Reactor (SFR),and Lead-Cooled Fast Reactor (LFR). With the exception of the sodium-cooled fast reactor, the information on these reactor designs is taken from [INEEL 2004].

A.3.1 Very-High-Temperature Reactor (VHTR)

The VHTR system is a helium-cooled, graphite moderated, thermal neutron spectrum reactor with an outlet temperature of 1000°C or higher. It will be used to produce electricity and hydrogen. It is important to note that the reactor core design has not yet been selected. The final core may be either a prismatic graphite block design, or a pebble bed reactor design. The reactor thermal power (400-600 MWt) and core configuration will be designed to assure passive decay heat removal without fuel damage during accidents.

The VHTR, prismatic or pebble bed, have passive safety features built into their designs. If a fault occurs during reactor operations, the system, at worst, will come to a standstill and merely dissipate heat on a decreasing curve without any core failure or release of radioactivity to the environment. The inherent safety is a result of the design, the materials used, the fuel and the natural physics involved, rather than active engineered safety. Its passive safety features include: particle fuel in a graphite matrix, a low power density, a high surface area to volume thermal transfer geometry, a high heat capacity, a single-phase coolant that is chemically and radiologically inert, and a negative temperature coefficient of reactivity. Based on these passive safety features, an argument is made that there is no event that raises temperatures high enough to damage intact fuel particles. Thus, a significant release of radionuclides is prevented. The inherently safe design is supposed to render the need for safety grade backup systems obsolete.

The VHTR design is based on limiting the peak transient fuel temperature to 1600°C. This is about 400°C below the SiC dissociation temperature, where damage to the integrity of the primary containment layer is certain to occur. The multiple layer TRISO fuel particles are designed to contain fission product gases and trap solid fission products. The graphite surrounding the fuel particles in either design can further serve to trap fission products released from the particles. Graphite has a high capacity for retaining some fission products, but is virtually transparent to others (e.g., noble gases).

The VHTR reactor shutdown system would be similar to many current systems in LWRs, in that it passively can shut the reactor down. Loss of the coolant normally available to hold the scram rods out of the core would allow them to drop into the core. Another concept would use electromagnets to suspend the scram rods above the core. An increased temperature, above normal, in the core raises the electrical resistance in the electromagnets circuits so that insufficient current flows to provide the magnetic field strength needed to suspend the rods.

In order to enable passive decay heat removal, the VHTR core was designed with a low power density and a high surface area to volume geometry. These traits along with the graphite reflector/moderator's high heat capacity allow decay heat to be transferred in a slow, passive manner. The VHTR power density is about 5 to 7 W/cc (or MW/m3). This is quite low compared to typical LWR power densities of about 70 to 100 MW/m3. The VHTR has a tall annular geometry that provides a large surface area for heat transfer. The large volume of graphite in the fuel matrix and in the center and outer reflectors is able to store a lot of heat and release it slowly over the large surface area via conductive and radiative heat transfer.

The reactor cavity cooling system (RCCS) is a passive heat removal system that relies upon both radiation and natural convection heat transfer to remove the decay heat from the reactor. In contrast with typical LWRs, no reliance is placed upon it to protect the fuel from exceeding its maximum design temperature. The main purpose of the RCCS is to protect the reactor cavity wall and the RPV from thermal degradation.

The RCCS includes three independent cooling systems, each capable of absorbing 50% of the rejected heat from the RPV. Each cooling system has 15 water chambers arranged vertically on the reactor cavity wall. Steel shields or cooling panels are erected between the water chambers and the RPV. The cooling systems are low-pressure, closed loop, pump driven, with an internal water-to-water heat exchanger. Heat is transferred to an open water loop to the ultimate heat sink, either a large body of water or the atmosphere. The natural convection flow in the region between the RPV and cooling panels is induced by buoyancy forces in the air as a result of the temperature difference between the RPV and the cooling panels. It is assumed that the cooling panels have enough heat removal capability to maintain the panel surface temperature at approximately 27°C.

The heat transfer from the pebbles is dominated by convection during nominal operation of the reactor. However, during an accident when the flow in the core decreases to near zero, the heat generated by the pebbles is removed by conduction and radiation through the pebbles to the graphite reflector. In the prismatic design, with fuel compacts in holes of the graphite blocks, conduction would play an even larger role in the heat transfer from fueled to moderator/reflector regions.

A.3.2 Supercritical Water-Cooled Reactor (SCWR)

The SCWR is basically an LWR that is operating at higher pressure and temperature with a direct once-through cycle. Operating above the critical pressure eliminates coolant boiling, so the coolant remains single-phase throughout the system. As with current LWRs, the SCWR will require high pressure and low pressure injection systems that are primarily active in nature to address loss-of-cooling-accident (LOCA) events and removal of decay heat after reactor shutdown. Transients involving a total loss of feedwater pose a serious challenge to the reactor.

The SCWR would be considered to have passive structural fuel barriers (fuel cladding) (i.e., no signal inputs, external power, moving parts or moving working fluids). However, the remaining safety systems necessary for prevention of fission product release would fall into the active safety category.

While many of the safety characteristics are similar to those related to LWRs, the major difference lies in the large enthalpy rise in the core. As noted by Nuclear Energy Reseach Initiative (NERI) research partner Westinghouse, "The problem with SCWRs versus the LWRs is that their core average enthalpy rise is 10 times higher (typically SCWR core ΔT is more than 220°C versus about 40°C for PWRs, plus there is a change of phase) and that has to be multiplied by the total hot channel factor to determine the limiting cladding temperature under steady-state conditions. On top of this, the temperature rise must be further increased to account for transient/accident conditions." This issue drives the materials requirements higher by orders of magnitude and creates a stiff challenge for the designers.

A.3.3 Gas-Cooled Fast Reactor (GFR)

The GFR is a fast-spectrum reactor with a close relationship with the GT-MHR, the PBMR, and the VHTR. Like thermal-spectrum helium-cooled reactors, the high outlet temperature of the helium coolant makes it possible to produce electricity, hydrogen or process heat with high conversion efficiency. The GFR's fast spectrum makes it possible to utilize available fissile and fertile materials with fuel efficiency several orders of magnitude larger than thermal spectrum reactors. The GFR design is less mature than several other Generation IV concepts and three design options are being considered.

The reference GFR system features a fast-spectrum, helium-cooled reactor and closed fuel cycle. This was chosen as the reference design due to its close relationship with the VHTR, and thus its ability to use as much VHTR material and balance-of-plant technology as possible. Like the thermal-spectrum helium-cooled reactors, the GFR's high outlet temperature of the helium coolant makes it possible to deliver electricity, hydrogen, or process heat with high conversion efficiency. The GFR reference design uses a direct-Brayton cycle helium turbine for electricity and process heat for thermochemical production of hydrogen.

The primary optional design is also a helium-cooled system, but uses an indirect Brayton cycle for power conversion. The secondary system of this alternate design uses supercritical CO_2. This allows for more modest temperatures in the primary circuit (~600 - 650°C), reducing the strict fuel, fuel matrix, and material requirements as compared to the direct cycle, while maintaining high thermal efficiency (~42%). The secondary optional design is a supercritical CO_2 cooled direct Brayton cycle system. The main advantage of this design is the modest outlet temperature in the primary circuit, while maintaining high thermal efficiency (~45%). The modest outlet temperature reduces the requirements on the fuel, fuel matrix/cladding, and materials. It also allows for the use of more standard metal alloys within the core.

While many of the safety characteristics of the GFR are similar to other Generation IV concepts, the high power density of this design results in higher decay heat rates and higher temperature increases in the fuel and core. A combination of passive and active systems is proposed to remove decay heat. A pressure retaining guard containment will maintain coolant density to permit heat removal through natural circulation. An active shutdown cooling system, driven by a passive CO_2 accumulator will transfer reactor heat to the ultimate heat sink. In the GFR, reactivity feedbacks play a more prominent role than in thermal gas reactor designs. An important design objective will be to produce sufficient inherent negative reactivity so that the core power safely adjusts itself to the available heat sink.

A.3.4 Sodium-Cooled Fast Reactor (SFR)

The sodium-cooled fast reactor (SFR) features a fast-spectrum, sodium cooled reactor and a closed fuel cycle for efficient management of actinides and conversion of fertile uranium. The primary mission for the SFR is the management of high-level wastes, and in particular, management of plutonium and other actinides, but also includes electricity production. It offers the most direct path forward toward implementation of an effective actinide management strategy, with 99.9% of the actinides recovered and recycled. Systems that employ a fully closed fuel cycle can reduce repository space and performance requirements, but their costs must be manageable. Fast spectrum reactors have the ability to utilize almost all of the energy in the natural uranium versus the 1% utilized in thermal spectrum systems. SFRs are the most technologically developed of the Generation IV systems, since SFRs have been built and operated in France, Japan, Germany, the U.K., Russia, and the U.S. The SFR system is the nearest-term actinide management system in the Generation IV portfolio, estimated to be deployable by 2020. Based on the actinide management and electricity production missions, the primary focus of the research and development of the SFR is on the recycle technology, economics of the overall system, assurance of passive safety, and accommodation of bounding events. On the reactor side, demonstration of passive safety and improvements in inspection and serviceability will be emphasized.

The fuel cycle employs a full actinide recycle with two major options: One involves intermediate-sized (150 to 500 MWe) sodium-cooled fast reactors with uranium-plutonium-minor-actinide-zirconium metal alloy fuel, supported by a fuel cycle based on pyrometallurgical processing in facilities integrated with the reactor. The second involves medium to large (500 to 1500 MWe) sodium-cooled reactors with mixed uranium-plutonium oxide fuel, supported by a fuel cycle based upon advanced aqueous processing at a centralized location serving a number of reactors. The outlet temperature is about 550 degrees Celsius for both.

The safety characteristics of the SFR involve reliance on passive response, large thermal inertia, large margins to boiling, operation at low pressure, and a decay heat removal system that needs no forced circulation. A large margin to coolant boiling is achieved by design, and this is an important safety feature of these systems, since it assures single phase phenomena. Another major safety feature is that the primary system operates at essentially atmospheric pressure, pressurized only to the extent needed to move fluid. An extensive technology base in nuclear safety has shown that the passive safety characteristics of the SFR have the ability to accommodate all of the classical anticipated transients without scram (ATWS) events without fuel damage.

A negative safety characteristic is that sodium reacts chemically with air, and especially with water. To improve safety, a secondary sodium system is used in the design, which acts as a buffer between the radioactive sodium in the primary system and the steam or water that is contained in the conventional power plant cycle. With this feature, if a sodium-water reaction occurs, it does not involve a radioactive release.

Major research and development needs exist for both the pyroprocess fuel cycle and the advanced aqueous fuel cycle. For the safety of the reactor system, assurance or verification of passive safety needs to be further demonstrated, and some extremely low probability but high consequence accident scenarios need to be investigated. In addition, completion of the fuels database including establishing irradiation performance data for fuels fabricated with the new fuel cycle technologies must be established, and the capability for in-service inspection and repair in sodium technologies must be demonstrated.

A.3.5 Lead-Cooled Fast Reactor (LFR)

The LFR is a small lead or lead bismuth eutectic cooled fast-spectrum reactor. It is envisioned as a factory-built turn-key plant with a closed fuel cycle with a very long life. It would be designed for small grid markets and for developing countries. With small liquid metal fast reactors, it is possible to design for natural circulation of the primary coolant with a conventional steam generator power cycle or direct turbine cycles with either He or supercritical CO_2 and a Brayton power cycle. One of the leading LFR applications being considered is the STAR-LM Reactor. The Secure Transportable Autonomous Reactor-Liquid Metal (STAR-LM) project was undertaken to develop a modular nuclear power plant for electric power production with optional production of desalinated water that meets the requirements of a future sustainable world energy supply architecture optimized for nuclear rather than fossil energy.

The LFR system provides for ambient pressure single-phase primary coolant natural circulation heat transport and removal of core power under all operational and postulated accident conditions. External natural convection-driven passive air-cooling of the guard/containment vessel is always in effect and removes power at decay heat levels. The strong reactivity feedback from the fast neutron spectrum core with transuranic nitride fuel and lead coolant results in passive core power reduction to decay heat while system temperatures remain within structural limits, in the event of loss-of-normal heat removal to the secondary side through the in-reactor lead-to-CO_2 heat exchangers.

From the outset, the design and safety philosophy of STAR-LM has been to eliminate the need for reliance upon any active systems. The LFR system provides for ambient-pressure single-phase primary coolant natural-circulation heat transport and removal of core power under all operational and postulated accident conditions. External natural convection-driven passive air cooling of the guard/containment vessel is always in effect and removes power at decay heat levels.

Although scram systems are provided to insert rods to shut down the reactor neutronically, success of scram is not required to prevent the evolution of adverse power or temperature conditions. The STAR-LM LFR system provides for ambient pressure single-phase primary coolant natural circulation heat transport and removal of core power without scram under all accident conditions. This is a consequence of:

- The high boiling temperature of the lead heavy liquid metal coolant equal to 1740°C that realistically eliminates boiling of the low pressure coolant;

- The chemical inertness of the lead coolant that does not react chemically with carbon dioxide above about 250°C (well below the 327°C Pb melting temperature) and does not react vigorously with air or water;

- Natural circulation heat transport of the lead coolant at power levels in excess of 100% nominal that eliminates the entire class of loss-of-flow accidents;

- Transuranic nitride fuel that is chemically compatible with the lead coolant. The high nitride thermal conductivity together with bonding of the fuel and cladding with molten Pb results in low fuel centerline temperatures and small thermal energy storage in the fuel;

- External natural convection-driven passive air cooling of the guard/containment vessel (surrounding the reactor vessel) that is always in effect and removes decay heat power levels;

- Strong reactivity feedbacks from the fast neutron spectrum core with transuranic nitride fuel and lead coolant. There is no reliance upon the motion of control rods either due to operator action or inherent insertion due to heat up of the control rods or control rod drivelines;

- The system pool configuration and ambient pressure coolant with a reactor vessel and surrounding guard vessel that eliminates loss-of-primary coolant; and

- The high heavy metal coolant density (ƒ´Pb=10400 Kg/m3) that limits void growth and downward penetration following postulated heat exchanger tube rupture such that void is not transported to the core but instead rises benignly to the lead free surface through a deliberate escape channel between the heat exchangers and the vessel wall.

Due to the passive safety features of the reactor, the S-CO_2 gas turbine Brayton cycle secondary side does not need to meet safety grade requirements. In the event of a heat exchanger tube rupture, a blowdown of secondary CO and CO vessel must be provided and activity that is entrained from the lead coolant into the CO_2 must be contained. Thus, a pressure relief system is provided for the primary coolant system. The S-CO secondary circuit incorporates valves to isolate the failed heat exchanger and limit the mass of CO that can enter the primary coolant system.

Following an accident such as a loss-of-heat sink without scram in which the reactor power has passively decreased to a low level of after-heat typical of decay heat levels, it may be enough to simply return to power. Or it may only be required for an operator to ultimately insert the shutdown rod(s) to terminate possible fission power at low after-heat levels and render the core subcritical. Until this action is taken, the reactor would continue to generate power at a low level that is removed by the guard vessel natural convection air-cooling system and transported to the inexhaustible atmosphere heat sink.

The LFR coolant enables the traditional sustainability and fuel cycle benefits of a fast neutron spectrum core. The chemical inertness and high boiling temperature of heavy metal coolants provides passive safety with the prospect of boiling realistically eliminated. The core always remains covered and heat can be transported through natural convection. The design features autonomous load following and as long as the reactor and guard vessels remain intact, heat is removed from the fuel by natural circulation of the liquid metal coolant and from the guard vessel/containment by natural circulation of air.

A.3.6 Advanced CANDU Reactor 700 (ACR-700)

The advanced CANDU reactor (ACR) design is based on the use of modular horizontal fuel channels surrounded by a heavy water moderator, the same feature as in all CANDU® reactors. The major innovation in ACR is the use of slightly enriched uranium fuel, and light water as the coolant, which circulates in the fuel channels. The ACR-700 design described represents a standard two-unit plant with each unit having a gross output of 753 MWe with a new output of approximately 703 MWe.

The safety enhancements made in ACR encompass safety margins, performance and reliability of safety related systems. In particular, the use of the CANFLEX® fuel bundle, with lower linear rating and higher critical heat flux, permits increased operating and safety margins of the reactor. Passive safety features draw from those of the existing CANDU plants (e.g., the two independent shutdown systems), and other passive features are added to strengthen the safety of the plant (e.g., a gravity supply of emergency feedwater to the steam generators).

The reactivity control units are comprised of the in-reactor sensor and actuation portions of reactor regulating and shutdown systems. Reactivity control units include neutron flux measuring devices, reactivity control devices, and safety shutdown systems. Flux detectors are provided in and around the core to measure neutron flux, and reactivity control devices are located in the core to control the nuclear reaction. In-core flux detectors are used to measure the neutron flux in different zones of the core. Fission chamber and ion chamber assemblies mounted in housings on the calandria shell supplement these. The signals from the in-core flux detectors are used to adjust the absorber insertion in the zone control assemblies. Control absorber elements penetrate the core vertically. These are normally parked out of the reactor core and are inserted to control the neutron flux level at times when a greater rate or amount of reactivity control is required than can be provided by the zone control assemblies.

Slow or long-term reactivity variations are controlled by the addition of a neutron-absorbing liquid to the moderator. Control is achieved by varying the concentration of this “neutron absorbent material” in the moderator. For example, the liquid “neutron absorbent material” is used to compensate for the excess reactivity that exists with a full core of fresh fuel at first startup of the reactor. Two independent reactor safety shutdown systems are provided. The safety shutdown systems are independent of the reactor regulating system and are also independent of each other.

The emergency core cooling (ECC) system is designed to supply water to the reactor core to cool the reactor fuel in the event of a LOCA. The design bases events are LOCA events where ECC is required to fill and maintain the heat transport circuit inventory. The ECC function design is accomplished by two sub-systems: 1) the emergency coolant injection (ECI) system, for high-pressure coolant injection after a LOCA, and 2) the long term cooling (LTC) system for long term recirculation/recovery after a LOCA. The LTC system is also used for long term cooling of the reactor after shutdown following other accidents and transients.

The ACR-700 would be considered to have passive structural fuel barriers (fuel cladding) (i.e., no signal inputs, external power, moving parts or moving working fluids). Additional passive safety systems include two independent shutdown systems and a gravity supply of emergency feedwater to the steam generators serve to promote the safety characteristics of this design.

A.3.7 Pebble Bed Modular Reactor (PBMR)

The PBMR is a helium-cooled, graphite-moderated high temperature reactor. The PBMR uses particles of enriched uranium oxide coated with silicon carbide and pyrolytic carbon. The particles are encased in graphite to form a fuel sphere or pebble about the size of a tennis ball. Helium is used as the coolant and energy transfer medium, to drive a closed cycle gas turbine and generator system. The geometry of the fuel region is annular and located around a central graphite column. The latter serves as an additional nuclear reflector.

The thermodynamic cycle used is a Brayton cycle with a water-cooled inter-cooler and precooler. A high efficiency recuperator is used after the power turbine. The helium, cooled in the recuperator, is passed through the pre-cooler, inter-cooler and the low and high-pressure compressors before being returned through the recuperator to the reactor core.

The power taken up by the helium in the core and the power given off in the power turbine is proportional to the helium mass flow rate for the same temperatures in the system. The mass flow rate depends on the pressure, so the power can be adjusted by changing the pressure in the system.

The PBMR has passive safety features built into its design. If a fault occurs during reactor operations, the system, at worst, will come to a standstill and merely dissipate heat on a decreasing curve without any core failure or release of radioactivity to the environment. The inherent safety is a result of the design, the materials used, the fuel and the natural physics involved, rather than active engineered safety. These passive safety features include: particle fuel in a graphite matrix, a low power density, a high surface area to volume thermal transfer geometry, a high heat capacity, a single-phase coolant that is chemically and radiologically inert, and a negative temperature coefficient of reactivity. Based on these passive safety features, an argument is made that there is no credible event that raises temperatures high enough to damage intact fuel particles. Thus, a significant release of radionuclides is prevented.

The PBMR design is based on limiting the peak transient fuel temperature to 1600°C. This is about 400°C below the SiC dissociation temperature, where damage to the integrity of the primary containment layer is certain to occur. The multiple layer TRISO fuel particle was designed to contain fission product gases and trap solid fission products. The graphite surrounding the fuel particles in either design can further serve to trap fission products released from the particles. Graphite has a high capacity for retaining some fission products but is virtually transparent to others (i.e., noble gases).

The PBMR proposes to use a standard control rod drive mechanism for control and hot shutdown via borated control rods moving in the inner portion of the outside reflector. Similar to current systems, cutting power to the control rod drive motors allows the rods to drop by gravity. For cold shutdown, 8 channels in the central reflector can be filled with 1 cm diameter borated graphite spheres. The small spheres are stored in a container in a space underneath the RPV head. On demand, the storage container valve opens and the spheres fall by gravity into holes in the reflector. In the event that the electrical supply to the magnetic valve is interrupted, the valve will fall open. A pneumatic system is used to return spheres to storage in controlled quantities.

In order to enable passive decay heat removal, the PBMR core was designed with a low power density and a high surface area to volume geometry. These traits along with the graphite reflector/moderator’s high heat capacity allow decay heat to be transferred in a slow, passive manner. The PBMR power density is about 5 to 7 W/cc (or MW/m3). This is quite low compared to typical LWR power densities of about 70 to 100 MW/m3.

The RCCS is a passive heat removal system that relies upon both radiation and natural convection heat transfer to remove the decay heat from the reactor. No reliance is placed upon it to protect the fuel from exceeding its maximum design temperature. The main purpose of the RCCS is to protect the reactor cavity wall and the RPV. The heat transfer from the pebbles is dominated by convection during nominal operation of the reactor. However, during an accident when the flow in the core decreases to near zero, the heat generated by the pebbles is removed by conduction and radiation through the pebbles to the graphite reflector.

A.4 References

[INEEL 2004] “Generation IV Advanced Reactor Safety Characteristics Report,” Report Developed for Office of Nuclear Energy, Science and Technology USDOE, Idaho National Engineering and Environmental Laboratory, December 2004.

APPENDIX B
RELATIONSHIP TO 10 CFR

B. RELATIONSHIP TO 10 CFR

B.1 Introduction

This Appendix contains (1) the relationship of the requirements in 10 CFR Part 50 to requirements in other parts of 10 CFR, and (2) the relationship of the requirements of other parts of 10 CFR to the requirements of 10 CFR 50. The requirements that are related span a number of areas ranging from purely administrative to physical security and safeguards, technical criteria, standards for radiation protection, and personnel qualifications and training.

B.2 Relation of 10 CFR 50 Requirements to Other Parts of 10 CFR

The data in Table B-1 shows the linkages of 10 CFR 50 requirements to other parts of 10 CFR and the content of the link. The content of the link describes how the requirements are related and the initial part that is italicized displays the title of the content, i.e., what is referred to by the description. The abbreviations in Table B-1 are as follows:

SNM	special nuclear material (U-235, U-233, Pu)
CP	construction permit
OL	operating license
PSAR	Preliminary Safety Analysis Report
FSAR	Final Safety Analysis Report

Table B-1 Link of 10 CFR 50 requirements to other portions of 10 CFR.

Part 50 Subpart	Link to Other 10 CFR	Content of Link
50.2 Definitions	Part 100.11	*Definition of basic component* for the purpose of 50.55(e): “capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those in 100.11"
50.2 Definitions	Parts 30 and 70	*Definition of production facility*: exempts facilities designed or used for batch processing of SNM licensed under Parts 30 and 70 but places limits on amounts of U-235/other SNM in each process batch
50.2 Definitions	Part 100.11	*Definition of safety-related SSCs*: “SSCs that are relied upon to remain functional during and following DBAs to assure the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those in 100.11"
50.2 Definitions	Part 40	*Definition of source material* is that defined in Part 40

B. Relationship to 10 CFR

Table B-1 Link of 10 CFR 50 requirements to other portions of 10 CFR.

Part 50 Subpart	Link to Other 10 CFR	Content of Link
50.10(e)(1) and (2) License Requirements	Parts 51.20(b), 51.104(b) and 51.105	*Environmental*: Authorizes applicant for a construction permit for a utilization facility subject to 51.20(b) to prepare site for construction, install support facilities, etc., provided final EIS under Part 51 is completed and findings made under 51.104(b) and 51.105 that proposed site is suitable from radiological health and safety standpoint
50.30 Filing of Applications	Part 2.101	*Admin requirement* that requires docketing of application under Part 2.101 before releasing copies
50.34(a) Content of Applications- Preliminary Safety Analysis Report	Part 100	*PSAR* by applicants for CP under Part 50 or a design certification/COL under Part 52: Safety assessment must pay attention to the site evaluation factors in Part 100; site characteristics must comply with Part 100
50.34(b)(10) and (11) Content of Applications- Final Safety Analysis Report	Part 100	*FSAR*: OL applicants/license holders under Part 50 whose CP was issues before 01/10/97 will comply with (1) earthquake engineering criteria in Section VI of Part 100 Appendix A and (2) reactor site criteria in Part 100 and geologic/seismic criteria in Part 100 App A
50.34(c) Content of Applications - Physical Security	Parts 11 and 73	*Physical security*: OL applicants must include plan that describes how facility meets requirements of Parts 11 and 73
50.34(d) Content of Applications - Safeguards Contingency Plan	Parts 73.50, 73.55, 73.60	*Safeguards contingency*: OL applicants must include a licensee safeguards contingency plan complying with criteria in Part 73 App C
50.34(e) Content of Applications - Unauthorized Disclosure	Part 73.21	*Protection against unauthorized disclosure*: OL applicants who prepare physical security and safeguards contingency plans must comply with Part 73.21 requirements
50.35 Construction permits	Part 100	*CP* may be issued before completion of technical information if there is reasonable assurance that with respect to site criteria in Part 100 the facility can be constructed and operated at proposed location without undue risk to health and safety
50.36a Tech specs on effluents from reactor operation	Part 20.1301	*Compliance with public dose limits* and to keep average annual releases ALARA: Reactor licensees will include tech specs to comply with Part 20.1301 for releases to unrestricted areas under normal operation and keep releases ALARA
50.37 Classified Information	Parts 25 and 95	Restrict *access to classified information* for individuals not approved under Parts 25 and 95

Table B-1 Link of 10 CFR 50 requirements to other portions of 10 CFR.

Part 50 Subpart	Link to Other 10 CFR	Content of Link
50.40 Common standards	Parts 20 and 51	*Standards for issuing licenses*: Reasonable assurance that licensee will comply with Part 20 to protect health and safety and with requirements of Part 51 Subpart A
50.54(l) Conditions of licenses	Part 55	*Operator qualification*: Reactor controls must be handled by licensed operator or senior operator as provided in Part 55 and senior operator must be present/on-call at all times during operation
50.54(p)(1) Conditions of licenses	Part 73	*Maintaining safeguards contingency plan*: Prepare/maintain safeguards contingency plan in accordance with Part 73 App C
50.54(w)(4)(ii)(B) Accident insurance as condition of license	Part 20	*Post-accident procedures*: Clean up and decontamination of surfaces inside auxiliary and fuel-handling buildings to levels consistent with occupational exposure limits in Part 20
50.55(e) Conditions of CPs	Part 21	*Record keeping*: Maintaining records in compliance with 50.55 satisfies CP holders obligations under Part 21. If defect or failure to comply with a substantial safety hazard has been reported previously under Part 21 or Part 73.71 then 50.55(e) requirements are met
50.59 Changes, tests, experiments	Part 54	*Records of changes in facility* must be maintained until the termination of license under Part 50 or Part 54 whichever is later
50.65 Maintenance monitoring	Part 100.11	*Scope*: safety-related SSCs that are relied upon to remain functional during and following DBAs to assure the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those in 100.11and non-safety SSCs
50.66 Thermal annealing of RPVs	Part 20	*Thermal Annealing Operating Plan*: Methods for performing thermal annealing must ensure occupational exposures are ALARA and comply with Part 20.1206
50.67 Accident source term	Part 54	*Applicability*: Applies to holders of renewed licenses under Part 54 whose initial OL was issued before 01/10/97 and who wish to revise their current DBA source term
50.68 Criticality accident requirements	Part 70	*Handling fuel assemblies*: Gives licensees the option of complying with Part 70.24 in detecting an accidental criticality or 50.68(b) in ensuring subcriticality

B. Relationship to 10 CFR

Table B-1 Link of 10 CFR 50 requirements to other portions of 10 CFR.

Part 50 Subpart	Link to Other 10 CFR	Content of Link
50.69 SSC Risk-informed categorization	Parts 21, 54 and 100	*Applicability and scope*: Parts 50 and 54 licensees or applicants for design approval/COL/manufacturing license under Part 52; may voluntarily comply with 50.69 requirements as an alternative to complying with Part 21 or Part 100 App A, Sections VI(a)(1) and (2) for RISC-3 and RISC-4 SSCs
50.73 Licensee Event Reports	Part 20	*Reportable events*: Any airborne release that results in concentrations in unrestricted area greater than 20 times the limits in Part 20 App B, Table 2, Col 1; any liquid release that exceeds 20 times the concentrations of Part 20 App B, Table 2, Col 2 in unrestricted area (except H-3 and dissolved noble gases)
50.74 Change in operator status	Part 55	*Administrative*: Change in operator status must be notified per requirements of Parts 55.31and 55.25
50.75 Decommissioning planning	Part 30	*Administrative*: Guarantee of funds for decommissioning costs may comply with requirements of Part 30 App A, B, and C as alternative to 50.75
50.78 IAEA Safeguards	Part 75	*Administrative*: Each holder of CP shall comply with Parts 75.6 and 75.11 through 75.14 to permit verification by IAEA
50.82 License Termination	Part 20	*Conditions for termination*: Meet dose criteria of Part 20 Subpart E
50.83 Partial release of site or facility for unrestricted use	Parts 20, 51, 100	*Dose and siting criteria*: public dose remains within limits of Part 20 Subpart D; siting criteria of Part 100 continue to be met; surveys demonstrate compliance with Part 20.1402 for unrestricted use areas; compliance with reporting requirements of Parts 20.1402 and 51.53
50.91 License amendment	Part 2	*Administrative*: Exceptions for public comment hearings and state consultations under Part 2 Subpart L; notice for public comment under Part 2.105 and, for emergency situations, under Part 2.106
50.92 Issuance of amendment	Part 2	*Administrative*: Notice under Part 2.105 for amendments involving significant hazards
50.120	Part 55	*Training* of personnel: Comply with Part 55.4
Appendix C Financial qualifications for CP	Parts 2 and 9	*Administrative*: Allows applicants to withhold information from public disclosure per Parts 2.790 and 9.5

B.3 Relationship of Requirements in Other Parts to 10 CFR 50

The data in Table B-2 shows the linkages of the other parts of 10 CFR to 10 CFR 50 requirements and the content of the link. The content of the link describes how the requirements are related and the initial part that is italicized displays the title of the content, i.e., what is referred to by the description.

Table B-2 Link of other portions of 10 CFR to 10 CFR Part 50*

10 CFR Subpart	Part 50 Subpart	Content of Link
10 CFR 1.43(a)(2)	Part 50	*Defines duties of NRR Office*, e.g., procedures for licensing, inspection, etc. of facilities licensed under Part 50
10 CFR 2.4	Part 50.2	*Definition of facility* as that defined in 50.2
10 CFR 2.101(a)(3)(I)	Part 50	*Procedure for issuance, amendment, transfer, or renewal of a license; Filing of applications*; additional copies required by Part 50
10 CFR 2.101(a)(5)	50.21(b)(2) or (3), 50.22, Part 50, 50.30f, 50.34(a), 50.33, 50.34(a)(1), 50.37	*Procedure for issuance, amendment, transfer, or renewal of a license; Filing of application*; completeness of application
10 CFR 2.101(a)(5)(a-1)	50.21(b)(2) or (3), 50.22, Part 50	*Procedure for issuance, amendment, transfer, or renewal of a license; Filing of application;* early site suitability issues for construction permit
10 CFR 2.101(a)(5)(1)	50.34(a)(1), 50.30(f), 50.33(a) through (e), 50.37, Part 50	*Procedure for issuance, amendment, transfer, or renewal of a license; Filing of application*; early site suitability issues for construction permit; content of application
10 CFR 2.101(a)(5)(2)	50.30(f), 50.33, 50.34(a)(1)	*Procedure for issuance, amendment, transfer, or renewal of a license; Filing of application*; early site suitability issues for construction permit; content of application
10 CFR 2.101(a)(5)(3)	50.34a, 50.34(a)	*Procedure for issuance, amendment, transfer, or renewal of a license; Filing of application*; early site suitability issues for construction permit; content of application
10 CFR 2.101(c)(1)	Part 50	*Procedure for issuance, amendment, transfer, or renewal of a license; Filing of application;* information for antitrust review
10 CFR 2.104(a), (b), (c)	50.21(b), 50.35, 50.22, 50.55b	*Hearing on Application*; Notice of Hearing and contents of Notice; administrative
10 CFR 2.105(a)	50.21(b), 50.22, 50.58, 50.91	*Notice of proposed action on application*; administrative

B. Relationship to 10 CFR

Table B-2 Link of other portions of 10 CFR to 10 CFR Part 50*

10 CFR Subpart	Part 50 Subpart	Content of Link
10 CFR 2.106(a)	50.21(b), 50.22	*Notice of issuance of license or license amendment*; administrative
10 CFR 2.109	50.21(b), 50.22	*Effect of timely renewal application of a license*; administrative
10 CFR 2.202(e)	50.109, Part 50 license	*Procedure for imposing requirements* by order modifying Part 50 license by backfit; administrative
10 CFR 2.310(a)	Part 50	*Selection of hearing procedures*; administrative
10 CFR 2.310(h)	Part 50	*Selection of hearing procedures*; administrative
10 CFR 2.328	50.21(b), 50.22	*Selection of hearing procedures*; Hearings to be public
10 CFR 2.329	50.21(b), 50.22	*Prehearing conference*; notice of timing; administrative
10 CFR 2.401	50.22	*Notice of hearing on applications* pursuant to Appendix N of Part 52 for construction permits for reactors described in 50.22
10 CFR 2.402	50.22	*Separate hearings* on particular issues
10 CFR 2.501	50.22	*Notice of hearing on applications related to Appendix M of Part 52* to manufacture power reactors of type described in 50.22
10 CFR 2.600 Part 2 Subpart F	50.21(b), 50.22	*Additional procedures applicable* to early partial decisions on site suitability
10 CFR 2.602	50.30(e)	*Filing fees for early review* of site suitability issues
10 CFR 2.603	50.21(b), 50.22, 50.33a	*Docketing of applications* for early review of site suitability
10 CFR 2.605	50.30(f)	*Additional considerations* on site suitability issues
10 CFR 2.606	50.10(e)	*Partial decisions* on site suitability issues
10 CFR 2.1103, Part 2 Subpart K	Part 50	*Hybrid hearing procedures* for expansion of spent fuel storage capacity at nuclear power plants
10 CFR 2.1202	50.92	*Informal hearing procedures* for NRC adjudications; authority/role of NRC staff in licensing actions that involve significant hazards considerations defined in 50.92
10 CFR 2.1301	Part 50	*Public notice* of receipt of a license transfer application
10 CFR 2.1403	50.92	*Expedited proceedings* with oral hearings; authority and role of NRC staff in licensing actions that involve significant hazards considerations defined in 50.92

Table B-2 Link of other portions of 10 CFR to 10 CFR Part 50*

10 CFR Subpart	Part 50 Subpart	Content of Link
10 CFR 8.4	Part 50	*AEC jurisdiction* over nuclear facilities and materials under the Atomic Energy Act
10 CFR 11.7	Part 50	*Criteria and Procedures* for determining eligibility for access to or control over SNM; Definitions
10 CFR 19.2	Part 50	*Notices, Instructions and reports* to workers; Scope of worker inspections and investigations
10 CFR 19.3	Part 50	*Notices, Instructions and reports* to workers; inspection and investigations; purpose
10 CFR 19.20	Part 50	*Notices, Instructions and reports* to workers; inspection and investigations; employee protection
10 CFR 20.1002	Part 50	*Standards for Protection Against Radiation*; General Provisions, scope
10 CFR 20.1003	Part 50	*Standards for Protection Against Radiation*; General Provisions, definitions
10 CFR 20.1101	50.34a	*Standards for Protection Against Radiation*; Radiation Protection Programs
10 CFR 20.1401(a)	Part 50, 50.83	*Standards for Protection Against Radiation*; Radiological Criteria for License Termination; General provisions and scope
10 CFR 20.1401(c)	50.83	*Standards for Protection Against Radiation*; Radiological Criteria for License Termination; General provisions and scope
10 CFR 20.1403(d)	50.82(a)&(b)	*Standards for Protection Against Radiation*; Radiological Criteria for License Termination; Criteria for license termination under restricted conditions
10 CFR 20.1404(a)(4)	50.82 (a)&(b)	*Standards for Protection Against Radiation;* Radiological Criteria for License Termination; Alternate criteria for license termination
10 CFR 20.2004	Part 50 App I, 50.34, 50.34(a), 50.71, 50.59	*Treatment or disposal* of radioactively contaminated waste oils by incineration
10 CFR 20.2201	50.73, 50.72	*Reports of thefts or loss* of nuclear material at a nuclear power plant
10 CFR 20.2202	50.72	*Notification of incidents* that exceed specified dose guidelines to individuals

B. Relationship to 10 CFR

Table B-2 Link of other portions of 10 CFR to 10 CFR Part 50*

10 CFR Subpart	Part 50 Subpart	Content of Link
10 CFR 20.2203	50.73	*Reports of exposures, radiation levels, and concentrations* of radioactive materials at operating power plants exceeding constraints or limits
10 CFR 20.2206	50.21(b), 50.22, 50.2	*Reports of individual monitoring* of power plant operators
10 CFR 21.2	50.23, 50.55(e), 50.72, 50.73, Part 50	*Scope of reporting of defects and noncompliance* by persons licensed to construct or operate a power plant
10 CFR 21.3	Part 50, 50.34(a), 50.67, App B,	*Reporting of Defects and Noncompliance*: Definitions
10 CFR 21.21	Part 50	*Notification of failure to comply* or existence of a defect and its evaluation
10 CFR 25.5	Part 50	*Access Authorization for Licensee Personnel*: Definitions
10 CFR 25.17	Part 50	*Approval for processing applicants* for license authorization
10 CFR 30.4	Part 50	*Domestic Licensing* of Byproduct Material: Definitions of Production and Utilization Facility
10 CFR 30.50	50.72	*Reporting Requirements*
10 CFR 40.60	50.72	*Domestic Licensing* of Source Material: Reporting Requirements
10 CFR 51.20	Part 50	*Licensing and Regulatory actions* requiring environmental impact statements
10 CFR 51.22	Part 50	*Licensing and regulatory actions* eligible for categorical exclusion or not requiring environmental review
10 CFR 51.50	50.36b	*Environmental Protection Regulations* for Domestic Licensing and related regulatory functions; Environmental report–construction permit stage
10 CFR 51.53	50.82	*Post-operating license* stage environmental review
10 CFR 51.54	50.4	*Manufacturing license* environmental report
10 CFR 51.101	50.10(c)	*NEPA Procedure* - Limitations on Actions
10 CFR 51.106	50.57(c)	*Public hearings* in proceedings for issuance of operating licenses
10 CFR 52.3	50.2	*Early site permits*; Definitions

Table B-2 Link of other portions of 10 CFR to 10 CFR Part 50*

10 CFR Subpart	Part 50 Subpart	Content of Link
10 CFR 52.13, Part 52 Subpart A	Part 50	*Relationship of application of construction permit* under Part 50 to application for early site permit under Part 52, Subpart A
10 CFR 52.15	50.30, 50.4	*Filing of applications for an early site permit* under Part 52, Subpart A
10 CFR 52.17	50.33, 50.34, 50.47, 50.10	*Contents of applications* for early site permit
10 CFR 52.18	Part 50	*Standards for review* of applications
10 CFR 52.25	50.10	*Extent of activities* permitted under early site permit
10 CFR 52.37	50.100	*Early site permit* is a construction permit for purposes of compliance with 50.100
10 CFR 52.39	50.109	*Finality* of early site permit determinations
10 CFR 52.45, Subpart B	50.4, 50.30(a), 50.30(b)	*Standard Design Certifications*: Filing of applications and filing requirements
10 CFR 52.47	Part 50 and Appendices, 50.34	*Standard Design Certifications*; Contents of applications
10 CFR 52.48	Part 50 and Appendices	*Standards* for review of applications
10 CFR 52.51	Part 50	*Administrative review* of applications
10 CFR 52.63	50.109, 50.12, 50.59	*Finality* of standard design certifications
10 CFR 52.75, Subpart C	50.4, 50.30, 50.38	*Combined Licenses*; Filing of applications
10 CFR 52.77	50.33	*Contents of applications*; general information
10 CFR 52.78	50.120	*Contents of applications*; training and qualification of power plant personnel
10 CFR 52.79	50.10, 50.30, 50.34	*Contents of applications*; technical information
10 CFR 52.81	Part 50	*Standards* for review of applications
10 CFR 52.83	Part 50, 50.51, 50.55(a), (b), (d), 50.58	*Applicability* of Part 50 provisions
10 CFR 52.91	50.10	*Authorization* to conduct site activities
10 CFR 52.93	50.12	*Exemptions and variances*

B. Relationship to 10 CFR

Table B-2 Link of other portions of 10 CFR to 10 CFR Part 50*

10 CFR Subpart	Part 50 Subpart	Content of Link
10 CFR 52.97	50.40, 50.42, 50.43, 50.47, 50.50, 50.91	*Issuance* of combined licenses
10 CFR 52.99	50.70, 50.71	*Inspection* during construction
10 CFR 52, Appendix A, *II*	50.2, 50.34, 50.36, 50.36a	*ABWR design certification*; Definitions
10 CFR 52, Appendix A, *IV*	50.36, 50.36a, Part 50	*ABWR design certification; additional requirements* and restrictions
10 CFR 52, Appendix A, *V*	Part 50, 50.34	*ABWR design certification; applicable regulations* (identifies exemptions from specific portions of 50.34)
10 CFR 52, Appendix A, *VIII*	50.12, 50.90, 50.109	*ABWR design certification; processes* for changes and departures
10 CFR 52, Appendix A, *X*	50.4, 50.71(e)	*ABWR design certification; records* and reporting
10 CFR 52, Appendix B, *II*	50.2, 50.34, 50.36, 50.36a	*System 80*+ design certification; Definitions
10 CFR 52, Appendix B, *IV*	50.36, 50.36a, Part 50	*System 80*+ design certification; additional requirements and restrictions
10 CFR 52, Appendix B, *V*	Part 50, 50.34, Appendix J	*System 80*+ design certification; applicable regulations (identifies exemptions from specific portions of 50.34 and Part 50 Appendix J)
10 CFR 52, Appendix B, *VIII*	50.12(a), 50.90, 50.109	*System 80*+ design certification; processes for changes and departures
10 CFR 52, Appendix B, *X*	50.4, 50.71(e)	*System 80*+ design certification; records and reporting
10 CFR 52, Appendix C, *II*	50.2, 50.34, 50.36, 50.36a	*AP 600 design certification*; Definitions
10 CFR 52, Appendix C, *IV*	50.36, 50.36a, Part 50	*AP 600 design certification*; additional requirements and restrictions
10 CFR 52, Appendix C, *V*	Part 50, 50.34, 50.55a, 50.62, GDC 17, GDC 19	*AP 600 design certification*; applicable regulations (identifies exemptions from specific portions of 50.34, 50.55a, 50.62 and Part 50 Appendix A, GDC 17 and GDC 19)
10 CFR 52, Appendix C, *VIII*	50.12(a), 50.90, 50.109	*AP 600 design certification*; processes for changes and departures
10 CFR 52, Appendix C, *X*	50.4, 50.71(e)	*AP 600 design certification*; records and reporting

Table B-2 Link of other portions of 10 CFR to 10 CFR Part 50*

10 CFR Subpart	Part 50 Subpart	Content of Link
10 CFR 52, Appendix D, *II*	50.2, 50.34, 50.36, 50.36a	*AP 1000 design certification*; Definitions
10 CFR 52, Appendix D, *IV*	50.36, 50.36a, Part 50	*AP 1000 design certification*; additional requirements and restrictions
10 CFR 52, Appendix D, *V*	Part 50, 50.34(f), 50.62(c), GDC 17	*AP 1000 design certification*; applicable regulations (identifies exemptions from specific portions of 50.34, 50.62 and Part 50 Appendix A, GDC 17)
10 CFR 52, Appendix D, *VIII*	50.12(a), 50.90, 50.109	*AP 1000 design certification*; processes for changes and departures
10 CFR 52, Appendix D, *X*	50.4, 50.59, 50.71(e)	*AP 1000 design certification*; records and reporting
10 CFR 52, Appendix M	50.4, 50.10, 50.12, 50.22, 50.23, 50.30, 50.33, 50.34, 50.35, 50.40, 50.45, 50.55, 50.56, 50.57, 50.58, Part 50 Appendices C, E, H, J	*Standardization of Design; Manufacture* of Power Reactors; Construction and Operation of Power Reactors Manufactured Pursuant to Commission License
10 CFR 52, Appendix N	50.4, 50.10, 50.33, 50.33a, 50.34, 50.34a, 50.58, Part 50	*Standardization of Power Plant Design*; Licenses to construct and operate power reactors of duplicate design at multiple sites
10 CFR 52, Appendix O	50.4, 50.22, 50.30, 50.33, 50.34, 50.34a, 50.54f	*Standardization of Design*; Staff Review of Standard Designs
10 CFR 52, Appendix Q	50.4, 50.21, 50.22, 50.30, 50.33, 50.34, 50.4	*Pre-Application* Early Review of Site Suitability Issues
10 CFR 54.3	Part 50, 50.2, 50.21, 50.22, 50.71	*Requirements for Operating License Renewal*; definitions
10 CFR 54.4	50.34, 50.48, 50.49, 50.61, 50.62, 50.63, 50.67	*Requirements for Operating License Renewal*; scope
10 CFR 54.7	50.4	*Requirements for Operating License Renewal*; written communications
10 CFR 54.15	50.12	*Requirements for Operating License Renewal*; specific exemptions

B. Relationship to 10 CFR

Table B-2 Link of other portions of 10 CFR to 10 CFR Part 50*

10 CFR Subpart	Part 50 Subpart	Content of Link
10 CFR 54.17	50.4, 50.30, 50.33	*Requirements for Operating License Renewal*; filing of application
10 CFR 54.19	50.33	*Requirements for Operating License Renewal*; content of application - general information
10 CFR 54.21	50.12	*Requirements for Operating License Renewal*; content of application - technical information
10 CFR 54.33	50.36b, 50.54	*Requirements for Operating License Renewal*; continuation of current licensing basis (CLB) and conditions of renewed license
10 CFR 54.35	Part 50	*Requirements for Operating License Renewal*; requirements during term of renewed license
10 CFR 54.37	50.71(e)	*Requirements for Operating License Renewal*; additional records and record-keeping requirements
10 CFR 55.1	Part 50	*Operators' Licenses*; purpose
10 CFR 55.2	Part 50	*Operators' Licenses*; scope
10 CFR 55.4	Part 50	*Operators' Licenses*; definitions
10 CFR 55.5	Part 50	*Operators' Licenses*; communications
10 CFR 55.25	50.74(c)	*Operators' Licenses*; incapacity due to disability or illness
10 CFR 60.152, Subpart G	Part 50, Appendix B	*Disposal of HLW in Geologic Repositories*; implementation of quality assurance program
10 CFR 63.73, Subpart D	50.55(e)	*Disposal of HLW at Yucca Mountain*; records, reports, tests and inspections: reports of deficiencies
10 CFR 70.20a, Subpart C	Part 50	*Domestic Licensing of Special Nuclear Material*; general licenses: license to possess SNM for transport
10 CFR 70.22, Subpart D	Part 50, Part 50 Appendix B	*Domestic Licensing of Special Nuclear Material*; License applications: contents of applications
10 CFR 70.23, Subpart D	Part 50, Appendix B	*Domestic Licensing of Special Nuclear Material*; License applications: requirements for the approval of applications
10 CFR 70.24, Subpart D	50.68, Part 50	*Domestic Licensing of Special Nuclear Material*; License applications: criticality accident requirements
10 CFR 70.32, Subpart E	Part 50, 50.90	*Domestic Licensing of Special Nuclear Material*; conditions of licenses

Table B-2 Link of other portions of 10 CFR to 10 CFR Part 50*

10 CFR Subpart	Part 50 Subpart	Content of Link
10 CFR 70.50 Subpart G	50.72	*Domestic Licensing of Special Nuclear Material*; SNM control, records, reports and inspections: reporting requirements
10 CFR 71.101	Part 50 Appendix B	*Packaging and Transport of Radioactive Material*; quality assurance requirements
10 CFR 72.3	Part 50	*Licensing Requirements for Independent Storage* of Spent Fuel, high level waster (HLW), and greater than Class C (GTCC) waste; definition of independent spent fuel storage installation (ISFSI)
10 CFR 72.30	50.75, Part 50	*Licensing Requirements for Independent Storage of Spent Fuel*, HLW, and GTCC waste; financial assurance and record keeping for decommissioning
10 CFR 72.32	50.47	*Licensing Requirements for Independent Storage of Spent Fuel*, HLW, and GTCC waste; emergency plan
10 CFR 72.40	Part 50	*Licensing Requirements for Independent Storage of Spent Fuel*, HLW, and GTCC waste; issuance of license
10 CFR 72.75	Part 50	*Licensing Requirements for Independent Storage of Spent Fuel*, HLW, and GTCC waste; reporting requirements for specific events and conditions
10 CFR 72.140	Part 50 Appendix B	*Licensing Requirements for Independent Storage of Spent Fuel*, HLW, and GTCC waste; QA requirements
10 CFR 72.184	Part 50	*Licensing Requirements for Independent Storage of Spent Fuel*, HLW, and GTCC waste; safeguards contingency plan
10 CFR 72.210	Part 50	*Licensing Requirements for Independent Storage of Spent Fuel*, HLW, and GTCC waste; general license for storage of spent fuel at power reactor sites
10 CFR 72.212	50.59	*Licensing Requirements for Independent Storage of Spent Fuel*, HLW, and GTCC waste; conditions of general license
10 CFR 72.218	50.54, 50.82	*Licensing Requirements for Independent Storage of Spent Fuel*, HLW, and GTCC waste; termination of licenses
10 CFR 73.1	Part 50	*Physical Protection of Plants and Materials*; purpose and scope
10 CFR 73.2	Part 50	*Physical Protection of Plants and Materials*; definitions
10 CFR 73.20	Part 50	*Physical Protection of Plants and Materials*; general performance objectives and requirements

B. Relationship to 10 CFR

Table B-2 Link of other portions of 10 CFR to 10 CFR Part 50*

10 CFR Subpart	Part 50 Subpart	Content of Link
10 CFR 73.50	Part 50	*Physical Protection of Plants and Materials*; requirements for physical protection of licensed activities
10 CFR 73.55	50.21, 50.22, 50.54, 50.72, 50.90, 50.109	*Physical Protection of Plants and Materials*; requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage
10 CFR 73.56	50.21, 50.22, 50.54, 50.90	*Physical Protection of Plants and Materials*; personnel access authorization for power plants
10 CFR 73.57	Part 50	*Physical Protection of Plants and Materials*; requirements for criminal history checks of individuals granted unescorted access to a nuclear power facility or access to safeguards information by licensees
10 CFR 73.67	Part 50	*Physical Protection of Plants and Materials*; licensee fixed-site and in-transit requirements for SNM of moderate and low strategic significance
10 CFR 73.71	50.72, 50.73	*Physical Protection of Plants and Materials*; reporting of safeguards events
10 CFR 73, Appendix B	Part 50	*Physical Protection of Plants and Materials*; general criteria for security personnel: definitions
10 CFR 73, Appendix C	Part 50 Appendix E	*Physical Protection of Plants and Materials*; licensee safeguards contingency plans
10 CFR 74.13	50.21, 50.22	*Material Control and Accounting of SNM*; Material Status Reports
10 CFR 74.31	Part 50	*Material Control and Accounting of SNM*; Nuclear material control and accounting for special nuclear material of low strategic significance
10 CFR 74.41	Part 50	*Material Control and Accounting of SNM*; SNM of moderate strategic significance
10 CFR 74.51	Part 50	*Material Control and Accounting of SNM*; formula quantities of strategic SNM: control and accounting for strategic SNM
10 CFR 75.2	50.78	*Safeguards on Nuclear Material* - Implementation of US/IAEA Agreement; Scope
10 CFR 75.4	50.2	*Safeguards on Nuclear Material* - Implementation of US/IAEA Agreement; definitions
10 CFR 95.5	Part 50	*Security Clearance and Safeguarding* of National Security Information and Restricted Data; definitions
10 CFR 100.1	Part 50	*Reactor Site Criteria*; purpose

Table B-2 Link of other portions of 10 CFR to 10 CFR Part 50*

10 CFR Subpart	Part 50 Subpart	Content of Link
10 CFR 100.2	Part 50	*Reactor Site Criteria*; scope
10 CFR 100.3	50.2, 50.21, 50.22, Appendix S	*Reactor Site Criteria*; definitions
10 CFR 100.21	50.34	*Reactor Site Criteria*; non-seismic siting criteria
10 CFR 100.23	50.10, Appendix S	*Reactor Site Criteria*; geologic and seismic siting criteria
10 CFR 100, Appendix A	Part 50 GDC 2, 50.10	*Reactor Site Criteria*; seismic and geologic siting criteria for power plants
10 CFR 140.2	Part 50	*Financial Protection Requirements and Indemnity Agreements*; scope
10 CFR 140.3	50.21	*Financial Protection Requirements and Indemnity Agreements*; definitions
10 CFR 140.10	Part 50	*Financial Protection Requirements and Indemnity Agreements*; provisions applicable only to applicants and licensees other than Federal Agencies and Non-Profit Educational Institutions; scope
10 CFR 140.11	Part 50	*Financial Protection Requirements and Indemnity Agreements*; amounts of financial protection for certain reactors
10 CFR 140.12	Part 50	*Financial Protection Requirements and Indemnity Agreements*; amounts of financial protection required for other reactors
10 CFR 140.13	Part 50	*Financial Protection Requirements and Indemnity Agreements*; amount of financial protection required of certain holders of construction permits
10 CFR 140.20	Part 50	*Financial Protection Requirements and Indemnity Agreements*; indemnity agreements and liens
10 CFR 140.51	Part 50	*Financial Protection Requirements and Indemnity Agreements*; provisions applicable only to Federal Agencies; scope
10 CFR 140.52	Part 50	*Financial Protection Requirements and Indemnity Agreements*; provisions applicable only to Federal Agencies; indemnity agreements
10 CFR 140.72	Part 50	*Financial Protection Requirements and Indemnity Agreements*; provisions applicable only to nonprofit educational institutions; indemnity agreements

B. Relationship to 10 CFR

Table B-2 Link of other portions of 10 CFR to 10 CFR Part 50*

10 CFR Subpart	Part 50 Subpart	Content of Link
10 CFR 150.15	Part 50	*Exemptions and continued regulatory authority* in agreement states and in offshore waters under Section 274, persons not exempt from regulation for storage of GTCC waste
10 CFR 170.2	Part 50	*Fees for Regulatory Services*; scope
10 CFR 170.3	Parts 50, 50.21, 50.22, 50.71	*Fees for Regulatory Services*; definitions
10 CFR 170.12	50.71	*Fees for Regulatory Services*; payment of fees
10 CFR 170.21	50.12	*Fees for Regulatory Services*; schedule of fees
10 CFR 170.41	Part 50	*Fees for Regulatory Services*; failure by applicant or licensee to pay fees
10 CFR 171.3	Part 50	*Annual Fees for Reactor Licensees*; scope
10 CFR 171.5	50.21, 50.22, 50.57	*Annual Fees for Reactor Licensees*; definitions
10 CFR 171.15	Part 50	*Annual Fees for Reactor Licensees*; annual fees for reactors licenses and independent spent fuel storage licenses
10 CFR 171.17	Part 50	*Annual Fees for Reactor Licensees*; proration of annual fees

* Where the second column mentions Part 50, it pertains to the entire part.

APPENDIX C
PROGRAMMATIC, POLICY AND TECHNICAL ISSUES

C. PROGRAMMATIC, POLICY AND TECHNICAL ISSUES

C.1 Introduction

The purpose of this appendix is to identify and discuss the programmatic, policy and open technical issues associated with the Framework. Since the Framework is only the first step in the development of a risk-informed and performance-based approach for future plant licensing (see Chapter 1), there remain additional issues needing resolution to complete development and implement the approach. Discussed below are the programmatic (i.e., directional), policy and open technical issues that the authors have identified and recommend be resolved as part of Framework implementation. It should also be noted that in the process of implementing the Framework, additional issues may arise that need to be resolved.

C.2 Programmatic Issues

The method by which the Framework is to be implemented will be a key factor in setting the direction for and in estimating the resources and schedule for additional work. Specifically, two fundamental issues should be resolved prior to implementation of the Framework. These are:

(a) should the Framework be implemented by rule-making or on a case-by-case basis, and

(b) should the Framework be implemented in a technology-specific or technology-neutral fashion?

The Framework takes no approach on the above issues, but rather identifies them as over-arching issues needing resolution prior to any significant additional work on implementation. Each of these issues have options for their resolution and are discussed below:

C.2.1 Rule-making versus Case-by-Case Implementation

The Framework has been written such that it can be implemented either through rule-making or on a design-specific, case-by-case basis, without rule-making.

Design-specific option, the requirements would be documented in the staff's Safety Evaluation Report or Final Design Approval and Design Certification. Rulemaking option, the requirements could be accomplished via modification to, or supplementing 10 CFR 50, or adding a stand alone new part to 10 CFR. The Framework has been written to support a stand alone implementation but could also support implementation via modification to 10 CFR 50.

In SECY-07-0101 [NRC 2007a], the staff recommended deferring a decision on rule-making until after development of the licensing strategy for Next Generation Nuclear Plants (NGNP) or receipt of an application of design certification of the Pebble Bed Modular Reactor (PBMR). The Commission, in a September 10, 2007 Staff Requirements Memorandum (SRM), agreed with the staff recommendation and suggested that the Framework be tested on a non-light water reactor (non-LWR), such as the PBMR, prior to a decision on rule-making [NRC 2007b]. Such a test can be done with either technology-specific requirements and guidance or with technology-neutral requirements supplemented by technology-specific implementation guidance. As a result, it may be several years before the issue of rule-making is decided, including whether it should be technology-specific or technology-neutral.

C.2.2 Technology-Neutral versus Technology-Specific Implementation

Resolution of the policy and open technical issues described in Sections C.3 and C.4 below is dependent upon whether or not they are being viewed from a technology-specific or technology-neutral standpoint. The Framework has been written such that it can be implemented using either approach. Specifically, the draft example requirements in Appendix J of the Framework have been written in a technology-neutral fashion and those areas where technology-specific guidance will be necessary are identified. The intent was that as the operational history and experience is accumulated and as new information is accrued, the needed changes would occur in regulatory guides rather than rule makings, to the extent possible.

Appendix J illustrates the content of the technology-specific guidance that will be needed. However, draft example requirements could also be written in a technology-specific fashion, if desired. An additional consideration is that the resolution of policy and open technical issues may be more straight forward if done in a technology-specific fashion, since only the safety issues, technical basis and uncertainties associated with that technology will need to be considered. Also, only one set of implementing guidance will need to be developed, in lieu of multiple sets to cover other technologies. This will likely be more straight forward and quicker than a technology-neutral approach. Therefore, to ensure an effective and efficient program to implement the Framework, a decision on technology-specific versus technology-neutral, including which technology should be made prior to any substantial additional work on Framework implementation.

C.3 Policy Issues

In SECY-03-0047, "Policy Issues Associated with Licensing Non-Light-Water Reactor Designs" [NRC 2003a] seven policy issues were identified as needing resolution to support licensing non-LWR designs. These seven issues are:

- Expectations for Enhanced Safety (i.e., level of safety and integrated risk)
- Defense-in-Depth
- Use of Codes and Standards
- Probabilistic Licensing Basis
- Mechanistic Source Term
- Containment
- Emergency Planning

The staff provided a recommendation on each issue. The staff's recommendations were general in nature (i.e., a concept and direction only) and required additional work to develop the details of how they should be implemented. This additional work (i.e., proposed approaches for implementation) was to be done as part of the Framework. The Commission, in a June 26, 2003 SRM [NRC 2003b], provided direction. The Commission approved the staff's recommendation to:

- pursue a licensing strategy and level of safety consistent with the advanced light water reactor (ALWR) design certification and the expectations for safety stated in the Commissions's 1986 Safety Goal Policy Statement [NRC 1986]

- develop a policy statement on defense-in-depth

- allow the use of
  - — a probabilistic approach for establishing the licensing basis
  - — a mechanistic source term

- make no near term changes to emergency preparedness (EP), but in the longer term, assess changes to EP consistent with the defense-in-depth policy

The Commission did not approve the staff's recommendation for early participation in codes and standards review and development. On the containment issue, the Commission requested that options be developed and submitted for Commission consideration. In addition, the Commission did not approve the staff's recommendation on integrated risk and requested further detail on options and impacts.[3]

The proposed approaches developed in the Framework for implementation also represent major changes from past practices and, at this time do not represent a staff recommendation, as such they are also considered policy issues. Each of these issues is discussed below, along with one additional issue (the development of Security Performance Standards) which was proposed in SECY-05-0120 and approved by the Commission in a September 9, 2005 SRM [NRC 2005a,b].

Accordingly, it is considered important to document the policy issues associated with the Framework so that if, and when, it is decided to use the Framework to support a licensing action or a rule-making, it will be apparent where Commission review and direction will need to be obtained prior to any use of the Framework in regulatory decision making. As such, this appendix serves as a means to document those issues of a policy nature (i.e., those that change or establish fundamental Commission regulatory practices or technical approaches) that have been incorporated into the Framework. The discussion of the policy issues contained in this appendix consists of a summary of each issue followed by a summary of the approach the Framework takes on each issue and where this is discussed in the Framework document. However, it needs to be recognized that, if in resolving the issue, the Commission direction is different from that taken in the Framework, the Framework will have to be modified accordingly.

The remainder of this section is organized by policy issue as follows:

- Defense-in-Depth
- Level of Safety
- Integrated Risk
- Probabilistic Licensing Basis
- Source Term
- Containment
- Emergency Planning
- Security Performance Standards

[3]The Commission in a later SRM directed the staff to seek further ACRS input and to seek stakeholder input in an Advanced Notice for Proposed Rulemaking.

C.3.1 Defense-in-Depth

In SECY-03-0047, "Policy Issues Related to Licensing Non-Light-Water Reactor Designs," the staff recommended the Commission take the following actions:

- Approve the development of a policy statement or description (e.g., white paper) on defense-in-depth for nuclear power plants to describe:

  — the objectives of defense-in-depth (philosophy)
  — the scope of defense-in-depth (design, operation, etc.)
  — the elements of defense-in-depth (high level principles and guidelines)

  The policy statement or description would be technology neutral and risk-informed and would be useful in providing consistency in other regulatory programs (e.g., Regulatory Analysis Guidelines).

- Develop the policy statement/description through a process involving stakeholder review, input, and participation.

In the SRM, dated June 26, 2003, the Commission approved the staff recommendation. As part of implementing the staff recommendation, a definition of defense-in-depth and principles for its application have been developed as part of the Framework (see Chapter 4). They have also been applied in the Framework to identify potential requirements that are needed for defense-in-depth purposes (see Chapter 8 and Appendix G).

The definition developed for defense-in-depth is: Defense-in-depth is an element of NRC's safety philosophy that is used to address uncertainty by employing successive measures, including safety margins, to prevent or mitigate damage if a malfunction, accident, naturally or intentional caused event occurs.

The principles associated with defense-in-depth included in Chapter 4 of the Framework are:

- consider intentional as well as inadvertent events;

- include accident prevention and mitigation capability;

- ensure key safety functions are not dependent upon a single element of design, construction, maintenance or operation;

- consider uncertainties in equipment and human performance and provide appropriate safety margin;

- provide alternative capability to prevent unacceptable releases of radioactive material; and

- site plants at locations that facilitate protection of public health and safety.

As stated in its definition, the purpose of defense-in-depth is primarily to account for uncertainties. As such, safety margin is a key element of defense-in-depth as well as the application of the deterministic principles which are primarily intended to account for completeness uncertainties.

In addition to the above principles, the Framework has also been developed using an overall defense-in-depth structure in the form of protective strategies(see Chapter 5). These protective strategies represent lines of defense for protecting public health and safety, the environment and common defense and security. Each design must incorporate these protective strategies which consist of:

- physical protection,
- stable operation,
- protective systems,
- barriers integrity, and
- protective actions.

The balance among the protective strategies and how they are accomplished may vary from design to design, but none can be eliminated.

The defense-in-depth structure definition, purpose and principles developed in the Framework are also intended to form the basis for a policy statement on defense-in-depth (as discussed in SECY-07-0101) . In the SRM -07-0101, dated September 10, 2007, the Commission directed that "the staff should develop a draft policy statement on defense-in-depth for future plants for Commission consideration."

C.3.2 Level of Safety

In using a risk-informed and performance-based approach for the development of requirements, a fundamental issue is what level of safety should the requirements aim to achieve. This issue was discussed in SECY-03-0047, SECY-04-0157 [NRC 2004a], and in SECY-05-0130 [NRC 2005c]. In these SECYs, it was noted that the Framework proposes to use the quantitative health objectives (QHOs) from the Commission's 1986 Safety Goal Policy Statement as the overall level of individual risk to the public which the requirements should achieve (i.e., must achieve at least this level of safety). Accordingly, the risk-risk-informed and performance-based requirements are intended to be consistent with the QHOs. This issue is discussed in Chapter 3 of the Framework document where the QHOs are proposed as the top level criterion which the reactor designs would have to meet.

The use of the Commission's Safety Goals in this fashion represents a change from the Commission's previous guidance (contained in a June 15, 1990, SRM [NRC 1990]) which stated that the safety goals represent "how safe is safe enough" (i.e., a level of safety where additional regulation is not warranted). Accordingly, the proposed approach in the Framework will need Commission review and approval. The Commission directed the staff to consider ACRS views and solicit stakeholder input in an advance notice of proposed rulemaking (ANPR).

Both ACRS and stakeholder views are discussed in detail in Appendix L.

C.3.3 Integrated Risk

Traditionally, plant risk information has been calculated and used on an individual reactor basis, regardless of the number of reactors on a given site. However, with the possibility of future reactors being of a modular nature (i.e., several small reactors co-located to produce the power output of one large reactor) and with the potential for future reactors to be put on existing sites which already contain one or more reactors, the need to consider the integrated (i.e., cumulative) risk from all reactors on a site has been raised.

This issue is also related to how the Commission interprets the application of its Safety Goal Policy (i.e., to a site or to an individual reactor) and was discussed in SECY-03-0047, SECY-04-0103 [NRC 2004b], and SECY-05-0130. In these SECYs, it was noted that the Framework proposes to require that the integrated risk from all future reactors on a site be used in assessing whether or not the risk criteria proposed in the Framework (e.g., QHOs) are met (the risk from existing reactors on a site would not have to be considered). This approach to integrated risk could impact whether or not a particular site is suitable for multiple reactors. Integrated risk is discussed in Chapter 6 of the Framework.

The consideration of integrated risk on a site would be a major change in Commission practice from what is currently done. Accordingly, the proposed approach in the Framework will need Commission review and approval. The Commission directed the staff to consider ACRS views and solicit stakeholder input in an ANPR.

Both ACRS and stakeholder views are discussed in detail in Appendix L.

C.3.4 Probabilistic Licensing Basis

In SECY-03-0047, the staff recommended the use of a probabilistic approach for the development of the licensing basis (e.g., selection of events to be considered in the design, safety classification). The Commission approved the staff's recommendation in a June 26, 2003, SRM. However, the details of how this was to be done were yet to be developed. The Framework (in Chapter 6) proposes an approach for implementing this recommendation. This approach relies on a full scope design-specific probabilistic risk assessment (PRA) and uses a probabilistic approach for:

- establishing frequency ranges to select and categorize events which need to be considered in the licensing basis;
- selecting licensing-basis events (LBEs) and establishing plant design features;
- establishing LBE acceptance criteria as a function of the frequency of event scenarios;
- classifying certain structures, systems and components (SSCs) as safety significant;
- replacing the single failure criterion, where practical; and
- establishing security performance standards.

Because the licensing basis is risk-derived, the approach also requires (1) maintaining the PRA up to date over the life of the reactor, and (2) continually reassessing plant risk, licensing basis events, safety classification, etc., using actual operating experience to determine if the plant licensing basis remains valid. This will involve feeding the results from the updated PRA back into plant operation and, where the licensing basis is affected, reporting the results to NRC.

The event sequences to be considered in the design are those with a frequency of 10^{-7}/ry or larger. As such, some of the low frequency event sequences may include severe accidents. This would represent an extension of the licensing basis from one defined by traditional anticipated operational occurrences and DBAs, to one including accidents beyond traditional DBAs.

In addition, the probabilistic licensing basis uses a frequency-consequence (F-C) curve (Figure 6-2 in Chapter 6) to define dose limits as a function of event scenario frequency. The purpose of this curve is to ensure that event scenarios with high frequency have low consequences and lower frequency event scenarios can have higher consequences, thus providing some measure of risk neutrality. The basis for the F-C curve is described in Chapter 6 and issues associated with its implementation are discussed in Section C.4.2. The dose limits defined by the F-C curve are to be calculated consistent with the times and distances (i.e., either at the exclusion area boundary (EAB) for low doses or, for higher doses, the worst 2-hour dose at the EAB and the dose at the outer edge of the low-population zone (LPZ) for the duration of the event). This was done to be consistent with existing criteria (e.g., 10 CFR 20, siting) as much as possible. Some deterministic (i.e., defense-in-depth) acceptance criteria have also been specified for use in conjunction with the F-C curve. These are shown in Chapter 6 (Table 6-3) and address fuel damage, safety system availability and barrier integrity.

To accomplish the above, the Framework has been developed using what has been called a fully "risk-informed" (or a "risk-derived") approach. This approach differs from the current "risk-informed" activities in that it uses risk information as the underlying basis for the requirements and design specific risk information for their implementation, supplemented with deterministic engineering judgement (e.g., defense-in-depth, good engineering practices), whereas, the current risk-informed approach uses risk information to supplement deterministic requirements.

It is recognized that the Framework approach to the use of risk information represents a transition from the way risk information is currently used in regulatory decision-making to an approach that relies more on the use of risk information to develop requirements.[4]

The reason this approach was taken in the Framework is that it facilitates the integration of safety, security and preparedness by having a common measure (risk) with which to compare and assess the impact of each on the others. As such, it can be the means with which to implement a unified concept for protecting public health and safety, the environment and the common defense and security. It also helps ensure coherence among design, construction, maintenance, operation, security and inspection.

This approach will require a design specific full scope, Level 3 PRA. As such, requirements and guidance on PRA quality, use and information submitted for review will be necessary for implementation of this approach. The extent of the risk information to be considered as part of the licensing basis also needs to be determined. In addition, the PRA is to be maintained up-to-date over the life of the plant and used to ensure that the plant's licensing basis remains up-to-date.

Chapter 6 of the Framework discusses the key elements associated with this approach and its implementation. A discussion on PRA scope and technical acceptability needed to implement this approach and its corresponding probabilistic licensing basis is contained in Chapter 7.

These uses of risk information and the expanded scope of the licensing basis should be reviewed and approved by the Commission since they establish frequency criteria for what must be considered in the design and what can be excluded, they establish acceptance criteria for events

[4] In the current regulatory approach, risk information and insights are used to supplement the deterministic-based structure. In the Framework, the regulatory structure is established from the start integrating both deterministic and probabilistic information and insights.

which must be considered in the design and they represent a major change in current licensing practices.

C.3.5 Source Term

In SECY-03-0047, the staff proposed and the Commission approved (in a June 26, 2003 SRM) the use of design specific and event specific mechanistic source terms (i.e., source terms that can vary depending upon the specific accident scenario, reactor design and reactor technology being evaluated) in lieu of a traditional single large deterministic source term (for LWRs representative of a core melt accident) for the assessment of dose and comparison against regulatory dose criteria (e.g., siting dose criteria, dose in the control room following an accident) and for assessment of equipment performance (e.g., equipment qualification). In Chapter 6, the Framework proposes implementation of a mechanistic source term approach and defines the conditions under which the use of design specific and event specific mechanistic source terms can be justified and used in licensing. These conditions include:

- having sufficient experimental data to confirm the source term (e.g., quantity and form of radio-nuclides, timing of release); and

- accounting for uncertainties in the source term determination (i.e., use 95% confidence level)

However, for siting purposes, a modified design specific mechanistic source term is proposed (for comparison against the current siting dose criteria) which would be representative of a larger release of radio-nuclides from an accident, that was otherwise excluded from the design basis for the reactor. This larger source term would provide margin to account for uncertainties and would also be used to establish the radiological containment functional capability for the design, as described in Section C.3.6.

Accordingly, due to the implications of using design specific and event specific mechanistic source terms in licensing, the technical basis for, the method for determining, and the uses of such source terms in licensing are considered a policy issue requiring Commission review and approval.

C.3.6 Containment

In SECY-03-0047, the staff requested Commission direction on the use of a traditional containment design on future non-LWRs. In a June 26, 2003, SRM, the Commission directed that the staff develop containment functional performance requirements and provide the Commission with options for consideration. The staff, in SECY-04-0103, provided the Commission with a status report on the containment options evaluation and, in SECY-07-0101, the staff summarized stakeholder views on the Framework's proposed criteria for a radiological containment functional capability. Stakeholder input is discussed in detail in Appendix L.

Appendix G of the Framework proposes to establish a design specific radiological containment functional capability consistent with current siting criteria and based upon a technology-neutral set of performance criteria contained in the Framework. Other containment functions (e.g., shielding, heat removal) would also be design specific, but their resolution would be by traditional engineering solutions, and thus, are not considered of a policy nature. The technology-neutral set of radiological containment functional criteria contained in the Framework are intended to result in a barrier, separate from the fuel and the reactor coolant system pressure boundary, that can perform a radiological containment function by limiting the dose to that specified in 10 CFR 50.34(a)(i)(ii)(D) (i.e., the worst 2 hour dose at the site boundary need not exceed 25 rem TEDE and the dose for the duration of the accident at the outer edge of the low population need not exceed 25 rem TEDE) using the modified design specific mechanistic source term for siting discussed in Section C.3.5.

The radiological containment function would need to be maintained for all frequent and infrequent events and for those rare events where credit is taken for its performance. In addition, security considerations also need to be factored into the containment functional capability design. The application of the radiological containment functional criteria would be design specific and would likely result in different containment designs (e.g., closure time, pressure retaining capability) that are consistent with the characteristics of the technology and design being reviewed. As such, these criteria would not be prescriptive or deterministic and may not always require a traditional pressure retaining low leakage containment, although for an LWR, it is expected this would still be the case. However, the proposed radiological containment functional criteria still remain a policy issue needing Commission review and approval.

C.3.7 Emergency Planning

In SECY-03-0047, the staff raised the issue "under what conditions can the emergency planning zone be reduced, including a reduction to the site exclusion area boundary?" The staff recommended that "no change to emergency preparedness requirements be made at this time. . . . .If approved by the Commission, the role of emergency preparedness in defense-in-depth would be addressed as part of the development of a policy or description of defense-in-depth. . . .In the longer term, if and when a need for change in emergency preparedness requirements is identified, that policy or description would serve as guidance in assessing the proposed change."

The Framework has evaluated existing EP requirements contained in 10 CFR 50.47 and 10 CFR 50, Appendix E in light of the defense-in-depth recommendations discussed in Section C.3.1. The defense-in-depth recommendations include retaining EP as a defense-in-depth measure, regardless of the plant design. In Appendix G, the Framework proposes an approach of retaining the 10 CFR 50.47 and 10 CFR 50, Appendix E requirements, but adding a provision that would allow future applicants to propose adjustments to current EP requirements based upon plant specific characteristics (e.g., timing of release, magnitude of release, plant risk). This approach would recognize that different plant characteristics may result in different EP needs and would permit applicants to propose appropriate adjustments (e.g., EPZ size, protective actions). Defense-in-depth and security would be key considerations in reviewing such proposals. In addition, other factors would need to be considered in reviewing proposed changes to EP requirements.

This would include factors such as:

- other Federal agency, state and local authority input and acceptance;
- the range of accidents that should be considered;
- operating experience; and
- security related events.

Finally, it should be noted that the Framework would require EP to consider accident scenarios down to a frequency of 10^{-7}/ry, but no lower.

Since this approach recognizes that changes to current EP requirements may be warranted for future designs (especially non-LWRs), which would represent a departure from past precedent and current practice in a highly visible area key to protecting public health and safety, Commission review and approval of this approach is warranted as a policy issue. It is anticipated, however, that for future LWRs, current EP requirements would likely remain unchanged.

C.3.8 Security Performance Standards

The staff, as part of SECY-05-0120, "Security Design Expectations for New Reactor Licensing Activities," proposed to develop security performance standards for Generation IV and other future reactor concepts as part of the technology-neutral Framework. The Commission approved the staff recommendation in a September 9, 2005, SRM.

Section 6.7 of the Framework includes proposed security performance standards developed in a risk-informed fashion. The proposed standards are technology-neutral and would require each applicant to prepare a security assessment that demonstrates high assurance of protection of public health and safety by:

(1) reducing vulnerabilities to DBTs and a limited set of events outside the DBT;

(2) ensuring that the plant design, operation and security provide multiple layers of defense against each security related threat that could endanger public health and safety, the environment or the common defense and security;

(3) ensuring that the plant design, operation and security provide both prevention and mitigation measures for each security related threat that could endanger public health and safety, the environment or common defense and security; and

(4) for plant designs that use Pu or HEU fuel, ensuring sufficient material control and accounting to detect the theft or diversion of significant amounts of material.

Specific acceptance criteria have been developed and proposed in the Framework for each of the above performance standards. In addition, the security assessment would need to be kept up to date over the life of the plant.

A portion of the acceptance criteria are based upon risk information (i.e., the early and latent fatality QHOs from the Commission's 1986 Safety Goal Policy Statement) and use conditional risk (the change in risk conditional upon the threat occurring and the assumed plant damage resulting from the threat) as the acceptance criteria. This would involve using a design specific PRA to assess security and its conditional risk. Since the proposed security performance standard's acceptance criteria are based upon the use of the QHOs, conditional risk information and defense-in-depth considerations and require a limited assessment of events outside the DBT, this represents a major change in how the adequacy of security is determined and, thus, is considered a policy issue needing Commission review and approval. In addition, the Commission's 1986 Safety Goal Policy Statement indicated that the risk from sabotage did not need to be included in the QHO calculations. Thus, a change in scope for the QHO calculations is also part of this issue.

C.4 Open Technical Areas

Several technical issues remain open with respect to the scope and completeness of the Framework. These technical issues have been identified by the authors and stakeholders. However, they do not affect the validity of the technical information contained in the Framework, but rather are issues to be dealt with as part of its implementation. Summarized below are the major open technical issues, their relevance to the Framework and actions needed to further their resolution. More detailed information may be found in Appendix L where the issue is discussed by the stakeholders. The major open technical issues discussed are:

- Complementary Cumulative Distribution Function (CCDF)
- Frequency-Consequence (F-C) Curve
- Fuel Handling and Storage
- Environmental Protection
- Framework Testing
- Security Frequency-Consequence Curve
- Design Codes and Standards
- PRA Standards and Use of the PRA
- Subsidiary Risk Objectives
- Importance Measures
- Completeness Check Findings
- NRC Reactor Oversight Program

C.4.1 Complementary Cumulative Distribution Function (CCDF)

Several stakeholders (including ACRS) commented on the Framework criteria for using risk information to establish the licensing basis for the design. These criteria are described in Chapter 6 of the Framework and include event scenario categorization, a process for selecting design specific licensing basis events, a frequency-consequence (F-C) curve for establishing the acceptable dose for the PRA event scenarios and licensing basis events, and the use of the QHOs from the Commission's Safety Goal Policy as a measure of acceptable overall risk from the plant. A major stakeholder comment was that a CCDF curve was also needed for completeness and as a complement to the F-C curve. Specifically, the CCDF curve could:

(1) complement the F-C curve discussed in Chapter 6 (Section 6.2.1) by ensuring that the risk from high frequency events is low (whereas the F-C curve ensures that the consequences from high frequency events is low),

(2) provide insight into the design specific distribution of risk, thus identifying areas where there may be a concentration of risk which may indicate a design deficiency,

(3) provide the basis for quantitatively establishing the desired relation between accident prevention and mitigation, and

(4) provide a criterion for assessing the integrated effect of safety, security and preparedness on risk.

The authors consider that the suggestion to include a CCDF curve as an additional acceptance criterion in Chapter 6 warrants additional study and should be pursued as part of implementing the Framework. However, it needs to be noted that the development of a CCDF will involve a number of considerations. These include:

- maintaining consistency with the QHOs and the F-C curve

- the use of dose versus individual risk as the consequence measure; if individual risk is chosen, it is likely two curves would be needed (one for early fatalities and one for latent fatalities)

- should the CCDF curve be
  - — site specific or generic
  - — address site risk (i.e., integrated risk) or individual plant risk

It should be noted that the ACRS, in their letter of September 26, 2007, also commented on this issue. This is discussed further in Appendix L.

C.4.2 Frequency Consequence (F-C) Curve

In Figure 6-2 (Chapter 6 of the Framework), a frequency-consequence curve is proposed that defines the dose criteria as a function of event scenario frequency. Wherever possible, the dose values chosen are the same as existing dose criteria (e.g., 10 CFR 20, 10 CFR 50). Frequency values are assigned to the dose values so as to ensure that for frequent event scenarios, the dose limits are low and, as the frequency of event scenarios gets lower, the dose limits can be higher. However, some existing dose criteria are expressed as annual dose limits (e.g., 10 CFR 20), which generally apply to event scenarios associated with anticipated operational occurrences and some are expressed as per event scenario dose limits which are generally associated with postulated accidents. The F-C curve contains both types of dose criteria. Accordingly, there are two issues associated with the F-C curve that need to be addressed as part of Framework implementation. These are:

(1) Are the event scenario frequencies and dose criteria associated with the F-C curve appropriate and practical, and

(2) Should the F-C curve be split into two curves: one with annual dose limits and one with per event scenario dose limits? It should be noted that in calculating whether or not the Commission's Safety Goal Policy QHOs are met, only the risk from accidents is to be considered. Accordingly, a two curve approach would help to clarify which event scenarios (i.e., those that correspond to per event scenario dose limits) are to be considered in the QHO calculations.

Finally, stakeholders (e.g., the Electric Power Research Institute (EPRI)) have commented on the F-C curve included in the Framework and these comments should be taken into consideration in the resolution of this issue. Appendix L provides a summary of stakeholder comments.

C.4.3 Fuel Handling and Storage

The current scope of the Framework does not include fuel handling and storage. This is due to focusing initial Framework efforts on the development of criteria for reactor safety. The use of risk information to develop criteria for the safety of fuel handling and storage will require different considerations than those for reactor safety. Therefore, the criteria developed for reactor safety may not be directly applicable to fuel handling and storage and need to be assessed for such an application. In addition, the risk information needed to assess the safety of fuel handling and storage needs to be defined and consistent with any criteria developed. Therefore, it is suggested that this issue be addressed as part of implementing the Framework.

C.4.4 Environmental Protection

The authors, in the initial stage of developing the Framework, assessed whether or not the level of safety used in the Framework (see Section C.3.2) also provides reasonable protection to the environment, such that separate goals and criteria on environmental protection are not needed. The basic approach taken in the assessment was to show that the risk to the environment was no greater than the risk to the public, using the 10 CFR 140 extraordinary nuclear occurrence (ENO) dose and land contamination criteria as the threshold for an unacceptable environmental impact. However, additional work is needed in this area during implementation, since no conclusion has been reached at this stage of Framework development and is not addressed in the Framework.

C.4.5 Framework Testing

The authors performed some limited testing of the Framework using a current licensed operating plants. The authors, and several stakeholders believe that the Framework needs to be tested against an actual advanced reactor design prior to any use of the document to support regulatory decision-making. The Commission, in an SRM dated September 10, 2007, directed that the Framework be tested on an actual design, and directed that the PBMR be used for this purpose. The limited test of the process described in Chapter 6 of the Framework for the selection of licensing-basis events (LBEs) was made, using an operating LWR, and is documented in Appendix E of the Framework. This limited test demonstrated the feasibility of the LBE selection process contained in the Framework. However, the other aspects of the Framework have not been tested. The authors agree with the stakeholder views on pilot testing the Framework prior to its use in any regulatory decision-making and will proceed in accordance with Commission direction. Such testing will require a design with a full scope PRA to be available, preferably with the cooperation of a designer.

C.4.6 Security Frequency-Consequence Curve

Framework Section 6.7 contains proposed security performance standards. As part of the security performance standards, a frequency-consequence (F-C) curve for assessing security related events is proposed (see Figure 6-3 in Chapter 6 of the Framework). This F-C curve uses a qualitative scale for frequency and the early and latent fatality QHOs for consequence. The qualitative nature of the frequency scale is due to the fact that the frequency of security related events is not known. The F-C curve has also been developed on the basis of conditional risk (i.e., assuming the initiating event has occurred). The technical issue is whether or not the consequence scale should be based upon dose, the same as that used in assessing other event scenarios in the PRA and safety analysis (see Figure 6-2 in Chapter 6 of the Framework), in lieu of early and latent fatalities.

The practicality and usefulness of using dose in lieu of early and latent fatalities in the security performance standards needs to be assessed as part of implementation of the Framework.

C.4.7 Design Codes and Standards

10 CFR 50 includes in its regulations requirements to design and build certain critical reactor components and systems according to specific consensus design codes and standards. The

specific consensus design codes and standards are identified in 10 CFR 50.55a and include the American Society of Mechanical Engineers (ASME) and Institute of Electrical and Electronics Engineers (IEEE) standards. However, the codes and standards identified in 10 CFR 50.55a are specific to LWR technology (e.g., LWR materials, temperatures, etc.). Before these codes and standards were included in 10 CFR 50, they required NRC review and endorsement. In some cases, NRC staff participated in the development of the codes and standards.

The example draft requirements in Appendix J include a requirement that future designs use consensus design codes and standards, as much as possible, for safety significant SSCs. Future reactor designs (especially non-LWRs) will likely need to develop and use consensus design codes and standards different than those currently in 10 CFR 50.55a. These will need to be developed to be applicable to the materials and conditions of the new designs and receive NRC review and endorsement.

Such development, review and acceptance of these codes and standards by NRC will require a long lead time. Therefore, to support Framework implementation, work on this open technical issue should begin early in the implementation process.

C.4.8 PRA Standards and Use of the PRA

Since a design specific PRA will play a central role in establishing the licensing basis for the design (and demonstrating that the risk-risk-informed and performance-based requirements have been met), the scope, depth and quality of a PRA acceptable for use in licensing needs to be defined. This will help ensure consistency and confidence for such use as well as defensability in any challenge to the PRA. For LWRs, the ASME and American Nuclear Society (ANS) have developed (or are developing) consensus standards for PRA quality. These standards are written for application to LWRs (e.g., data, risk metrics) and assume LWR data and systems are what need to be modeled; and at this time, they do not fully cover Level 2 and 3 analysis. However, ASME is working on developing a standard to support a PRA using an approach akin to that in the Framework.

Future designs using the Framework will need to do a PRA and that PRA will need to cover Level 1, 2 and 3 analysis. In addition, for non-LWRs, different data, systems and risk metrics will be needed. Therefore, to use the Framework, acceptable standards for non-LWR PRAs, as well as for LWR Level 2 and 3 PRAs, will need to be developed, reviewed and accepted by NRC. It is preferable that such standards be developed in a consensus fashion so as to ensure broad acceptance. Since this activity will require substantial time (i.e., the LWR PRA standards development took approximately five years) and resources, work needs to begin early so as to support use of the Framework by the NRC staff and licensing activities by applicants. In some cases, it may be reasonable to use methods other than a full PRA to assess the risk in a particular area (e.g., seismic). If this is to be allowed, standards for such analysis will also be needed.

In addition to and in support of PRA standards development, identification of how the PRA is to be used is essential. Such identification will shape what the PRA standards and the licensing requirements (and/or their supporting guidance) need to address. For example, the PRA can support plant design, construction and operation in the following areas:

- meeting those design requirements that rely on risk information (e.g., LBE selection);

- selection of those portions of the plant that have the largest safety significance for inspection, monitoring, testing and NDE during construction and operation;

- ensuring that procedures, training programs and technical specifications cover the most safety significant SSCs and human actions;

- determining allowable SSC outage times and plant configurations based upon risk;

- determining when is the optimum time (i.e., operation, refueling, shutdown) and duration for maintenance;

- determining plant staffing needs;

- determining plant aging program priorities;

- determining reliability assurance program scope and goals; and

- supporting the development of an NRC reactor oversight program that includes measures that focus on performance.

In addition to the above, during the life of the plant the PRA needs to be maintained to reflect operating experience and changes in plant configuration, operation and equipment availability, reliability and performance. Since the plant licensing basis is, to a large extent, dependent upon risk information, the risk information from the updated PRA needs to be fed back into the licensing analysis to ensure that the plant licensing basis remains valid. This would entail updating items such as LBE selection, safety classification and overall plant risk. Where the updated risk information indicates a change in the plant licensing basis is warranted, a process needs to be established to ensure the appropriate changes are made. This would include specifying how often the PRA needs to be updated, what needs to be reported to NRC and NRC review criteria.

During Framework implementation, each of these areas needs to be examined to ensure the requirements, their supporting guidance and the standards for the PRA describe what needs to be done to accomplish the above.

C.4.9 Subsidiary Risk Objectives

For current LWRs, risk objectives, (i.e., core damage frequency (CDF) and large early release frequency (LERF)) subsidiary to the Commission’s Safety Goal QHOs have been developed to focus more directly on reactor design by specifying accident prevention and mitigation goals and eliminating the need to do the source term portion of a Level 2 PRA analysis and the entire Level 3 analysis. These LWR subsidiary risk objectives are more conservative than the QHOs and were developed by working backward from Level 3 PRA information. This then established what risk metrics the plant design would have to meet to ensure the QHOs are met, accounting for the differences in source term, meteorology, population and EP among current LWR sites. A more detailed description of how CDF and LERF were derived is contained in Appendix B.

For non-LWRs, or LWRs substantially different than current LWRs, the current CDF and LERF subsidiary risk objectives may not be appropriate. This can be due to different reactor technologies (e.g., HTGR source term magnitude and timing are very different from LWR source terms) or EP different than that currently required by 10 CFR 50. In addition, little, if any, Level 3 PRA information exists for non-LWRs. Therefore, in implementing the Framework, it needs to be determined whether or not subsidiary risk objectives addressing accident prevention and mitigation are desired and, if so, should they be developed to be consistent with the reactor technology and EP being used or generically? One possibility in this regard is to use the CCDF curve, discussed in Section C.4.1, to define cumulative risk values associated with accident prevention and mitigation, building upon the philosophy used in establishing the LWR subsidiary risk objectives. These would then be independent of reactor technology, but would likely require Level 3 PRA information. However, it needs to be emphasized that developing generic subsidiary risk objectives is difficult due to the differences in reactor technology. ACRS, in their letter of September 26, 2007 has also commented on this issue, which is discussed further in Appendix L.

C.4.10 Importance Measures

The use of importance measures (e.g., risk achievement worth) as part of application of the PRA can provide insights useful to design. The Framework, in several places, discusses the use of performance measures. Examples include:

- special treatment requirements for SSCs classified as “safety significant”
- inspection programs
- surveillance and monitoring programs
- maintenance programs

However, for non-LWRs the risk metrics to be used need to be developed and tested. Accordingly, as part of implementation of the Framework the use of and methods for importance measures need to be developed.

C.4.11 Completeness Check Findings

In Appendix K a check was made on the completeness of the Framework by comparing the topics (in Chapter 8) and draft example requirements (in Appendix J) against other documents containing reactor design and operational requirements. Specifically, the Framework was compared against the following documents:

- 10 CFR 50 (NRC-reactor licensing requirements)
- IAEA NS-R-1 (design requirements) [IAEA 2000a]
- IAEA NS-R-2 (operational requirements) [IAEA 2000b]
- NEI 02-02 (draft risk-informed, performance-based Framework for licensing) [NEI 2002]
- UK HS&E document on safety assessment principles (HSE 2006]

The purpose of the comparison was to see if the above documents contained any topics or requirements not covered in the Framework. The results of this comparison are discussed in Appendix K. The comparison identified three design requirements contained in IAEA NS-R-1 that are not in the Framework. These are:

- the design should provide for automatic safety actions in the initial stage of accidents;
- the design should have escape routes for operating personnel; and
- the fuel assemblies should be designed to permit inspection.

In implementing the Framework, each of these should be evaluated to see if they should be specifically addressed, in some fashion, in the Framework

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C.4.12 NRC Reactor Oversight Process

It is recognized that the approach to licensing discussed in the Framework represents a significant departure from the current licensing approach, due to its extensive use of risk information. In addition, non-LWR reactor designs will have different requirements, systems, characteristics, measures of safety and performance than LWRs. All of these will need to be considered in NRC's reactor oversight process (ROP).

It is assumed that the current basic approach to reactor oversight (using measures of performance, along with deterministic considerations to identify important SSCs to be monitored and to gauge the significance of any irregularities in performance) will be maintained. Therefore, a design specific PRA can be very useful in developing a reactor oversight program by:

- identifying the most important SSCs;
- identifying the expected performance (e.g., reliability, availability, capacity) of the SSCs;
- identifying what parameters should be monitored to assess performance; and
- setting thresholds for performance based upon risk and safety margins.

These factors will help tie the ROP directly to the requirements, thus making the ROP easier to implement. Given the above, implementation of the Framework needs to consider development of an appropriate reactor oversight program.

C.5 References

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[IAEA 2000b] International Atomic Energy Agency, "Safety of Nuclear Power Plants: Operation Safety Requirements," Safety Standards Series No. NS-R-2, October, 2000.

[NEI 2002] Nuclear Energy Research Institute, "A Risk-Informed, Performance-Based Regulatory Framework For Power Reactors," NEI 02-02, May 2002.

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[NRC 2003b] U.S. Nuclear Regulatory Commission, "Staff Requirements - SECY-03-0047 - Policy Issues Related to Licensing Non-light-water Reactor Designs," June 26, 2003.

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[NRC 2004b] U.S. Nuclear Regulatory Commission, “Status of Response to the June 26, 2003, Staff Requirements Memorandum on Policy Issues Related to Licensing Non-light Water Reactor Designs,” June 23, 2004.

[NRC 2005a] U.S. Nuclear Regulatory Commission, “Security Design Expectations for New Reactor Licensing Activities,” SECY-05-0120, July 6, 2005.

[NRC 2005b] U.S. Nuclear Regulatory Commission, “Staff Requirements - SECY-05-0120 - Security Design Expectations for New Reactor Licensing Activities,” September 9, 2005.

[NRC 2005c] U.S. Nuclear Regulatory Commission, “Policy Issues Related to New Plant Licensing and Status of the Technology-neutral Framework for New Plant Licensing,” July 21, 2005.

[NRC 2007a] U.S. Nuclear Regulatory Commission, SECY-07-0101, “Staff Recommendations Regarding a Risk-Informed and Performance-Based Revision to 10 CFR Part 50,” June 14, 2007.

[NRC 2007b] U.S. Nuclear Regulatory Commission, “Staff Requirements - SECY-07-0101 - Staff Recommendations Regarding a Risk-informed and Performance-based Revision to 10 CFR Part 50,” September 10, 2007.

APPENDIX D
DERIVATION OF RISK SURROGATES FOR LWRS

D. DERIVATION OF RISK SURROGATES FOR LWRS

D.1 Introduction

The purpose of this appendix is to demonstrate that a core damage frequency (CDF) of 10^{-4} /year and a large early release frequency (LERF) of 10^{-5} /year are acceptable surrogates to the latent and early quantitative health objectives (QHO) for the current generation of light water reactors (LWRs).

The following are definitions of the QHOs as stated in the Safety Goal Policy Statement:

- "The risk to an average individual[5] in the vicinity of a nuclear power plant of prompt fatalities[6] that might result from reactor accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accident to which members of the U.S. population are generally exposed."

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[5] The Safety Goal Policy further states that the average individual in the vicinity of the plant is defined as the average individual biologically (in terms of age and other risk factors) and who resides within a mile from the plant site boundary. This means the dose conversion factors (DCFs) that translate exposure to dose (and hence risk) are for an average adult person (i.e., infant DCFs, etc. are not evaluated). In addition the average individual risk is found by accumulating the estimated individual risks and dividing by the number of individuals residing in the vicinity of the plant. (The statement also states that if there are no individuals residing within a mile of the plant boundary, an individual should, for evaluation purposes, be assumed to reside 1 mile from the site boundary).

[6] An accident that results in the release of a large quantity of radionuclides to the environment can result in acute doses to specific organs (e.g., red blood marrow, lungs, lower large intestine, etc.) in individuals in the vicinity of the plant. These acute doses can result in prompt (or early) health effects, fatalities and injuries. Doses that accumulate during the first week after the accidental release are usually considered when calculating these early health effects. The possible pathways for acute doses are: inhalation, cloudshine, groundshine, resuspension inhalation, and skin deposition. Cloudshine and inhalation are calculated for the time the individual is exposed to the cloud. Groundshine and resuspension inhalation doses for early exposure are usually limited to one week after the release. The doses accumulated during this early phase can be significantly influenced by by emergency countermeasures such as evacuation and sheltering of the affected population. Early fatality is generally calculated using a 2-parameter hazard function. A organ dose threshold is incorporated into the hazard function such that below the threshold the hazard is zero. (For example, the default value of the threshold for acute dose to red marrow is 150 rem [NRC 1990a] An early fatality is defined as one that results in death within 1 year of exposure.

- "The risk to the population in the area of nuclear power plant of cancer fatalities[7] that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1%) of the sum of cancer fatality risks resulting from all other causes."

Using risk surrogates to determine a plant's risk as compared to the QHOs is, in many cases, desirable over determining the actual risk of the plant. The risk of a plant is determined from a full-scope probabilistic risk assessment (PRA) which involves: (1) calculating the likelihood of all possible accident sequences leading to core damage, (2) determining whether or not the containment will be breached, (3) calculating the quantity of radionuclides that are released to the environment, and (4) calculating the consequences to the surrounding population.

As the calculations advance from determining the frequency of the accident sequences to estimating the off-site consequences, the calculations become more time consuming, complex and the results become more uncertain. In addition, many regulatory applications require the associated change in risk to be estimated in order to make a risk-informed decision. To perform a full scope PRA to calculate the change in risk associated with every risk-informed regulatory decision would be time consuming and impractical. Consequently, the possibility of using simple risk surrogates that could be compared to the QHOs was explored. It was determined that calculating the frequency of accident sequences leading to core damage and calculating the corresponding containment performance was sufficient information to be able to define surrogates that could be compared to the two QHOs.

For the current fleet of LWRs, defining these risk surrogates was possible. This possibility was because of the extensive severe accident research and the numerous PRAs that have been performed for these types of reactors. This research and large number of PRAs has characterized the radionuclide release and corresponding off-site consequences for a wide range of severe accidents and containment failure modes. The results of this research and calculations provide the basis for defining the risk surrogates as discussed in this appendix.

The following two numerical objectives have currently been adopted as surrogates for the two QHOs:

- A CDF of $<10^{-4}$ per year as a surrogate for the latent cancer QHO
- A LERF of $<10^{-5}$ per year as a surrogate for the early fatality QHO.

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[7] Lifetime 50-year committed doses can result in latent cancer fatalities. These doses occur during the early exposure phase (within one week of the release) from the early pathways, i.e. cloudshine, groundshine, inhalation, and resuspension inhalation, and the long-term phase from the long-term pathways that include groundshine, resuspension inhalation, and ingestion (from contaminated food and water). Just as early exposure can be limited by protective actions such as evacuation during the early phase, chronic exposure during the long-term phase can also be limited by actions such as population relocation, interdiction of contaminated land for habitation if it cannot be decontaminated in a cost-effective manner (within a 30-year period), food and crop disposal, and interdiction of farmland. A piecewise linear dose-response model is generally used to estimate cancer fatalities. A dose and dose rate reduction factor is used at low dose rates (<0.1 Gy per hour) and for low doses (< 0.2 Gy) to estimate cancer fatalities based on the recommendations of the International Commission on Radiation Protection in their ICRP 60 report. Up to 20 organs are included for estimation of latent cancers (e.g., lungs, red bone marrow, small intestine, lower large intestine, stomach, bladder wall, thyroid, bone surface, breast, gonads, etc.)

The following discussiong demonstrates how the above two numerical objectives were derived from the QHOs.

D.2 Surrogate for the Early QHO

The individual risk of a prompt fatality from all "other accidents to which members of the U.S. population are generally exposed," such as fatal automobile accidents, etc., is about $5x10^{-4}$ per year. The safety goal criteria of one-tenth of one percent of this figure implies that the individual risk of prompt fatality from a reactor accident should be less than $5x10^{-7}$ per reactor year (ry); i.e.:

$$(1/10 * 1\% * 5x10^{-4}) = 5x10^{-7}$$

The "vicinity" of a nuclear power plant is understood to be a distance extending to 1 mile from the plant site boundary. The individual early risk (IER) is determined by dividing the number of prompt or early fatalities (societal risk) to 1 mile due to all nuclear power plant accidents, weighted by the frequency of each accident, by the total population to 1 mile and summing over all accidents. This relationship is shown by Equation 1.

$$IER = \sum_1^N [(EFn * LERFn) / TP(1)] \qquad \text{Equation 1}$$

Where: EF_n = number of early fatalities within 1 mile conditional on the occurrence of accident sequence "n"

$LERF_n$ = frequency/ry of a large early release capable of causing early fatalities for accident sequence "n"

TP(1) = total population to 1 mile

The number of early fatalities (EFn) expected to occur for a certain population (TP(1)) given an accident is expressed as follows:

$$EFn = CPEFn * TP(1) \qquad \text{Equation 2}$$

Where: CPEFn = conditional probability of an individual becoming a prompt (or early) fatality (CPEF) for an accident sequence "n"

Therefore, the conditional probability of early fatality (CPEF) is:

$$CPEFn = EFn / TP(1) \qquad \text{Equation 3}$$

Consequently, the individual risk is (combining Equations 1 and 3):

$$IER = \sum_{1}^{N} CPEFn * LERFn \qquad \text{Equation 4}$$

It can be shown that if a plant's LERF is 10^{-5} per year or less, the early fatality QHO is generally met. This acceptance can be demonstrated numerically using the results of probabilistic consequence assessments carried out in Level 3 PRAs as follows:

(1) assuming that one accident sequence "n" dominates the early fatality risk and the LERF

(2) assuming the accident sequence dominating the risk is the worst case scenario:

- a large opening in the containment which occurs early in the accident sequence
- an unscrubbed release that also occurs early before effective evacuation of the surrounding population

(3) using results from NUREG-1150 [NRC 1990a] for the Surry PRA (Table 4.3-1) [NRC 1990b]

- the largest CPEF (within 1 mile) for internal initiators is $3x10^{-2}$.

  This conditional risk value corresponds to a large opening in containment and a very large release that is assumed to occur early before effective evacuation of the surrounding population. The definition of an early release is based on no effective evacuation. Consideration of when or if the vessel is breached as a result of the core melt is not directly pertinent to the definition for early release. Therefore, a "late release" is one where there is effective evacuation. It is consistent with the worst case assumptions for accident scenario "n".

Using the above value of CPEF and assuming a LERF goal of 10^{-5} per year, an estimate of the individual early risk can be made using Equation 4:

$$IER_y = (3x10^{-2}) * (10^{-5}) = 3x10^{-7}/\text{year}$$

The IER corresponding to a LERF = 10^{-5} per year is less than the early fatality QHO of $5x10^{-7}$ per year by a factor of about two. Using a LERF goal of 10^{-5} per year will thus generally ensure that the early fatality QHO is met. Therefore a LERF of 10^{-5}/year is an acceptable surrogate for the early fatality QHO.

D.3 Surrogate for the Latent QHO

The risk to the population from cancer "resulting from all other causes" is taken to be the cancer fatality rate in the U.S. which is about 1 in 500 or $2x10^{-3}$ per year. The safety goal criteria of one-tenth of one percent of this figure implies that the risk of fatal cancer to the population in the area near a nuclear power plant due to its operation should be limited to $2x10^{-6}$/ry; i.e.:

$$1/10 * 1\% * 2x10^{-3} = 2x10^{-6}$$

The "area" is understood to be an annulus of 10-mile radius from the plant site boundary. The cancer risk is also determined on the basis of an average individual risk, i.e., by evaluating the number of latent cancers (societal risk) due to all accidents to a distance of 10 miles from the plant site boundary, weighted by the frequency of the accident, dividing by the total population to 10 miles, and summing over all accidents. This implies:

$$ILR = \sum_{1}^{M} [(LFm * LLRFm) / TP(10)] \qquad \text{Equation 5}$$

Where: LF_m = number of latent cancer fatalities within 10 miles conditional on the occurrence of accident sequence "m"

$LLRF_m$ = frequency/ry of a release leading to a dose to an offsite individual

TP(10) = total population to 10 miles

The number of latent fatalities (LFm) expected to occur for a certain population (TP(10)) given an accident is expressed as follows:

$$LFn = CPLFm * TP(10) \qquad \text{Equation 6}$$

Where: CPLFm = conditional probability of an individual becoming a latent fatality (CPLF) for an accident sequence "m"

Therefore, the conditional probability of latent fatality (CPLF) is:

$$CPLFn = LFn / TP(10) \qquad \text{Equation 7}$$

Consequently, the individual latent risk is (combining Equations 5 and 7):

$$ILR = \sum_{1}^{N} CPLFm * LLRFm \qquad \text{Equation 8}$$

It can be shown that if a plant's CDF is 10^{-4} per year or less, the latent fatality QHO is generally met. This acceptance can be demonstrated numerically using the results of probabilistic consequence assessments carried out in Level 3 PRAs as follows:

(1) assuming that one accident sequence "m" dominates the latent fatality risk and the LLRF

(2) assuming the accident sequence dominating the risk is the worst case scenario:

- a large opening in the containment
- an unscrubbed release that occurs after effective evacuation of the surrounding population (i.e. no early fatalities occur)

(3) assuming that the accident occurs in an open containment, the conditional probability of large late release (CLLRPm) is 1.0; that is:

$$LLRFm = CDFm * CLLRPm \qquad \text{Equation 9}$$

$$LLRFm = CDFm * 1.0$$

Therefore, Equation 8 becomes:

$$ILRm = CPLFm * CDFm \qquad \text{Equation 10}$$

(4) using results from NUREG-1150 (Table 4.3-1) for the Surry PRA

- the largest CPLF (within 10 mile) for internal initiators is $4x10^{-3}$.

  The calculated CPLF values are very uncertain and therefore the approach adopted was to select a conservative estimate of CPLF. A CPLF value was therefore selected from the high consequence-low frequency part of the uncertainty range. This CPLF value corresponds to a large opening in containment and a very large release. It is therefore consistent with the worst case assumptions for accident scenario "m".

Using the above value of CPLF and assuming a CDF goal of 10^{-4} per year, an estimate of the individual latent risk can be made using Equation 10:

$ILRm = (4x10^{-3}) * (10^{-4}) = 4x10^{-7}$/year

The ILR corresponding to a CDF = 10^{-4} per year is less than the latent cancer QHO of $2x10^{-6}$ per year by a factor of about five. Using a CDF goal of 10^{-4} per year will thus generally ensure that the latent cancer QHO is met. Therefore a CDF of 10^{-4}/year is an acceptable surrogate for the latent cancer QHO.

D.4 References

[NRC 1990a] USNRC, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, December 1990.

[NRC 1990b] USNRC, “Evaluation of Severe Accident Risks: Surry, Unit 1,” NUREG/CR-4551, Vol. 3, October 1990.

APPENDIX E
EXAMPLE APPLICATION OF PROBABILISTIC APPROACH

E. EXAMPLE APPLICATION OF PROBABILISTIC APPROACH

E.1 Introduction

This appendix provides an example of the probabilistic selection process for licensing basis events (LBEs) and the selection of safety significant systems, structures and components (SSCs) as described in Chapter 6. The term ‘LBEs’ is used in the Framework to indicate those accidents considered in the safety analysis of the plant that must meet deterministic criteria in addition to meeting the frequency-consequence curve. The term ‘safety significant’ is used in the Framework to designate those systems requiring special treatment.

In the risk-informed approach used in the Framework, there are probabilistically selected LBEs and one deterministic LBE. The probabilistic LBEs are selected from probabilistic risk assessment (PRA) sequences. These probabilistically selected LBEs not only include sequences that involve a radionuclide release and lead to a dose at the site boundary and at one mile (see Section E.1.2 for explanation as to why this dose distance and duration is different from Chapter 6), but may also include sequences that do not involve any release of radionuclides. The process for identifying these probabilistically selected LBEs is included in this appendix. The deterministic LBE is considered for defense-in-depth purposes, as discussed in Subsection 6.4.3. An example of the selection of this deterministic event is not included in this appendix.

Those SSCs whose functionality plays a role in meeting the acceptance criteria imposed on the LBEs define the set of safety-significant SSCs. The SSCs of interest are those that influence the frequency or consequence of LBEs or both. The process of selecting these SSCs is also included in this appendix.

E.1.1 Results

The Framework selection process establishes a comprehensive set of licensing basis events that account for the frequency and severity of the events; and a comprehensive list of safety functions and their associated SSCs. The process identified LBEs with multiple failures and common cause failures and, in some cases, the events included the total loss of safety functions and containment failure. The selection process resulted in the identification of station blackout events (SBO) and anticipated transients without scram (ATWS) events as LBEs. The LBE identification process did exclude some rare event combinations that are currently considered as DBAs.

When the example results are compared against the Framework’s acceptance criteria, six LBEs are identified as exceeding the F-C curve and two events are identified as not meeting the deterministic potential requirements. These results are consistent with the Commission’s Policy Statement on “Regulation of Advanced Nuclear Power Plants,” which contains the expectation that advanced reactors will provide enhanced margins of safety. The results show that the Framework will enhance margins in that some events that are currently acceptable will not be acceptable in future reactors. The Framework selection process also results in a reduced emphasis on some rare event combinations and an increased focus on the most risk-significant events.

E.1.2 Differences Between Appendix E and Chapter 6 Guidance

The example included in this appendix was developed consistent with an intermediate version of Chapter 6 and as such does not reflect the final published version. This section discusses the differences and the impact of these differences.

Differences

The dose distance and duration for rare events used in the example is the dose at one mile for 24 hours. The dose distance and duration for rare events stated in the final published version of Chapter 6 is the dose at the worse two-hour dose at the exclusion area boundary (EAB) and the dose at the low population zone (LPZ) for the duration of the event dose.

Reason for Change

In the Framework's frequency-consequence (F-C) curve, the LBE and PRA event scenario dose limits specified by the curve are applied at various distances from the plant as shown in Figure 6-1. Specifically, the dose limits for the frequent events apply at the EAB and the dose limits for the infrequent and rare events apply at the EAB (worst dose in any two-hour period) and at the outer edge of the LPZ for the duration of the event, as defined in 10 CFR 100. The distances and durations chosen for the infrequent and rare event dose limits correspond to those currently used for siting determinations, as specified in 10 CFR 100 and 10 CFR 50.34(a)(1)(i)(D). This was done to tie the F-C curve, as much as possible, directly to existing dose criteria, including the distances at which they are applied. However, it is recognized that the dose limits in the rare event category exceed those specified in 10 CFR 50 and 10 CFR 100 for siting (i.e., 25 rem, TEDE at the EAB, worst two-hours, and 25 rem TEDE at the LPZ duration of the event). Because of this, earlier drafts of the Framework proposed a different distance and duration for assessing compliance with the rare event dose limits specified on the F-C curve (i.e., a 24-hour duration at one mile from the plant). This distance and duration was proposed to be consistent with the distance and duration generally used in calculating the early fatality QHO, since the dose limits in the lower frequency range of the rare event category are large enough for early fatalities to be predicted and was used in the example rare event LBE dose projections contained in Appendix E. However, the one mile/24 hour distance and duration are not part of the current regulations. Accordingly, it was decided to modify the rare event distance and duration to be consistent with current requirements, since a fundamental ground rule in the Framework development is to be compatible with other parts of 10 CFR and to use existing criteria, wherever reasonable. Therefore, the rare event LBE dose projections shown in Appendix E would need to be adjusted from a one mile/24 hour dose to an EAB - worse two hour dose/LPZ - duration of the event dose in order to match the potential requirements of Chapter 6.

Impact of the Change

Although the 24-hour dose at one mile and the 2-hour "worst case" dose at the site boundary will consist of very different pathway contributors, the total dose from each of these conditions is likely to be approximately the same.

For the 24-hour dose to a stationary recipient, the major contributor to total dose is the groundshine (dose due to radioactive material deposited on the ground) pathway which likely contributes more than half of the total dose. For the worst 2-hour dose to a stationary recipient the groundshine contribution is likely to be a minor contributor and the other pathways of inhalation and cloudshine will dominate. The magnitude of the inhalation and cloudshine doses will be considerably greater (probably by a factor of roughly 3 to 4) at the site boundary than at 1 mile. Hence the composition of the two dose conditions will be quite different.

The conclusion that total dose will be approximately the same (i.e., 24 hour 1 mile versus 2 hour site boundary) considers (1) insights from NUREG/CR-6094, which contains MACCS calculations of relative contributions from the groundshine, cloudshine, and inhalation pathways to the centerline dose to non-evacuee populations at various distances, including within 0 to 0.25 miles of the site boundary and 0.5 to 1.25 miles from the site boundary, from a various of severe accident source terms and gives a very rough estimate on which comparisons between the 24-hour dose at 1 mile and the hour dose at the site boundary, and (2) the source terms used to derive the unmitigated doses in this appendix are very conservative and hence the dose estimates based on these source terms are also likely very conservative.

E.2 Process

This section provides an overview of the LBE selection process, the process for selecting the dose duration and distance for the identified sequences, the process for evaluating the potential defense-in-depth requirements and the selection process for safety-significant SSCs.

E.2.1 LBE Selection Process

The LBE selection process is described in Chapter 6. This process assumes that the PRA used to support the LBE selection process is capable of evaluating event sequence doses and that the PRA includes those event sequences that would normally be considered to be success sequences (i.e., non-core damage sequences). The selection process includes the following steps.

1. Modify the PRA to credit only those mitigating functions that are considered to be safety significant.

2. Determine the point estimate frequency for each resulting event sequence from the quantification of the modified PRA.

3. For sequences with point estimate frequencies equal to or greater than $1\text{x}10^{-8}$ per year, determine the mean and 95^{th} percentile frequency.

4. Identify all PRA event sequences with a 95^{th} percentile frequency $\geq 1\text{x}10^{-7}$ per year. Event sequences with 95^{th} percentile frequencies less than $1\text{x}10^{-7}$ per year are excluded from further consideration.

5. Group the PRA event sequences with a 95^{th} frequency percentile $\geq 1\text{x}10^{-7}$ per year into event classes.

6. Select an event sequence from the event class that represents the bounding consequence.

7. Establish the LBE's frequency for a given event class.

8. Bin each LBE into one of three frequencies ranges: Frequent, Infrequent or Rare.

9. Determine the total weighted annual frequencies for all events equal to or greater than $1\text{x}10^{-2}$ and $1\text{x}10^{-3}$.

10. Verify that the selected LBEs meet the deterministic and probabilistic acceptance criteria.

Each of these steps is described in further detail in subsequent sections of this appendix.

E.2.2 Selection of Dose Distance and Duration

As stated in Chapter 6, the dose limits shown on the frequency-consequence curve are based on, and derived from, current regulatory requirements in Part 20, 50 and 100. However, these regulatory requirements reflect a variety of radiological exposure characteristics including variations associated with the distances from which dose is measured and variations in exposure duration characteristics. To reflect these variations, each event sequence is evaluated against one of three dose categories.

The first category is concern with the annual dose to an individual in an unrestricted area. Unrestricted area can be interpreted as the "Exclusion Area Boundary" or "EAB" which is defined as the boundary of the area surrounding the reactor where the reactor licensee has the authority to determine all activities, including exclusion or removal of personnel and property. Therefore the characteristics of the dose in the first category is the annual total effective dose equivalent (TEDE) to a receptor at the EAB.

The second category is associated with the dose that results from an event. 10 CFR 50.34(a)(1) characterizes the dose as the total radiation dose received to an individual at any point on the EAB for any two hours following the onset of a postulated fission product release. This time frame is often referred to as the worst two hours. 10 CFR 50.34 also includes an additional dose limit of 25 rem for an individual located at any point on the outer boundary of the low population zone (LPZ) who is exposed to the radioactive cloud resulting for a postulated fission product release (during the entire period of its passage). The LPZ is defined as the area immediately surrounding the exclusion area which there is a reasonable probability that appropriate protective measures can be taken for the residents in this area in the event of a serious accident. Within the Framework, the additional LPZ requirement is addressed separately from the F-C Curve and is therefore not included in the dose category characteristic.

The third category is associated with rare events and is assigned a 24 hour dose duration at 1 mile from the EAB.

These dose categories are summarized in Table E-1.

Table E-1 LBE dose categories

Cat.	Frequency	Characteristics
1	$\geq 10^{-3}$/ry	annual dose to a receptor at the EAB does not exceed the F-C Curve
2	< 10^{-3}/ry to $\geq 10^{-5}$/ry	the worst two-hour dose at the EAB does not exceed the F-C Curve[8]
3	<10^{-5}/ry to $\geq 10^{-7}$/ry	the 24 hour dose at 1 mile from the EAB

[8] The dose and duration criteria contained in Appendix E is based on an interim version of the Framework document. See Section E1.2 for additional discussion.

Therefore, it is necessary to relate each event sequence to the appropriate consequence measure based on its frequency. The initial assignment of the dose consequences is performed in Step 3. The annual dose determination is performed in Step 9.

E.2.3 Selection of Defense-in-Depth Requirements

The Framework uses three frequency categories to establish the potential defense-in-depth deterministic requirements as shown in Table 6-3 of the main report and summarized below in Table E-2.

Table E-2 LBE frequency categories

Category	Frequency	Deterministic LBE Criteria
frequent	$\geq 10^{-2}$/ry	- no barrier failure - no impact on fuel integrity or lifetime and safety analysis assumptions - redundant means for reactor shutdown remain functional - redundant means for decay heat removal remain functional
infrequent	$< 10^{-2}$/ry to $\geq 10^{-5}$/ry	- at least one barrier remains - a coolable geometry is maintained - at least one means of reactor shutdown remains functional - at least one means of decay heat removal remains functional
rare	$<10^{-5}$/ry to $\geq 10^{-7}$/ry	- none
• applies to all internal and external events • events with mean frequency <10-7/ry do not have to be considered in the design for licensing purposes		

It is necessary to know the frequency category of an event sequence in order to establish the applicable defense-in-depth requirement. The categorization process is performed in Step 8 of the LBE selection process.

E.2.4 Safety-Significant SSCs Selection

The determination of safety-significant SSCs is an integral part of the LBE selection process. The SSCs of interest are those that influence the frequency or consequence of the LBE's or both. All functions included in the PRA have the potential to influence the frequency of LBE sequences and many influence the consequences. Therefore, any function and the associated SSCs included in the PRA used to develop the set of LBEs is safety significant unless it has been set to 1.0, indicating guaranteed failure. The identification process is performed in Step 1 of the LBE selection process.

E.3 Example Plant

The example used in this appendix is a currently licensed pressurized water reactor (PWR) plant that was selected based on the availability of Level 2 PRA models. The plant is one of the three for which a SPAR (Standardized Plant Analysis Risk) Level 2/LERF model had been developed. Due to model limitations, the example is limited to at-power internal events related to the reactor core, excluding flooding and internal fires. These limitations are related solely to the scope limitations of this study, as it is expected that in actual practice, a fully developed PRA could be used to develop a complete set of LBEs. The required full-scope PRA model would include external events (seismic, high winds, etc.) and all modes of operation (hot standby, cold shutdown, refueling, etc.), as described in Chapter 7.

The selected Level 2/LERF model was modified for this example to facilitate the consequence analysis (the determination of the dose at the site boundary and at one mile). Seven designators were added to the existing end states (to allow characterization of both LERF and non-LERF end states), which contained six designators to enable unique consequence LERF end states to be determined. In this example, the consequence analysis was performed for all sequences with a point estimate frequency of $1\text{x}10^{-8}$ per year or greater.

A simple parametric approach to the consequence analysis was developed to permit representative doses to be assigned based on a limited set of MACCS2 calculations. For this purpose, NUREG-1465 release fractions from the core were adjusted to values that are representative of 95^{th} percentile from a quantitative uncertainty analysis.

The limited set of MACCS2 computations was then performed to obtain representative 95^{th} percentile doses without credit for radionuclide retention by plant features. Finally, representative dose reduction factors were applied to adjust these dose estimates to account for sequence-specific dose reduction by containment, containment engineered safety features, and other plant features. The resulting doses from the consequence analysis were then incorporated into the PRA model so that the LBEs can be selected based on both frequencies and consequences of the event sequences.

E.3.1 Initiating Events

This example uses a simplified set of initiating events that is consistent with those contained in the SPAR models. The initiating events identified in Table E-3 are included.

Table E-3 Initiating events

Initiating Event	Description	Frequency
IE-LDCA	Loss of One DC Bus	$2.5\text{x}10^{-3}$
IE-LLOCA	Large Break Loss of Coolant Accident (LOCA)	$5.0\text{x}10^{-6}$
IE-LOCCW-A	Loss of Component Cooling Water	$2.0\text{x}10^{-4}$
IE-LOESW	Loss of Essential Service Water (Essential Reactor Cooling Water)	$4.0\text{x}10^{-4}$

Table E-3 Initiating events

Initiating Event	Description	Frequency
IE-LOOP	Loss of Offsite Power	$3.3x10^{-2}$
IE-MLOCA	Medium Break LOCA	$4.0x10^{-5}$
IE-SGTR	Steam Generator Tube Rupture	$4.0x10^{-3}$
IE-SLOCA	Small Break LOCA	$4.0x10^{-4}$
IE-TRANS	Transient	$7.0x10^{-1}$
IE-RHR-DIS-V	Residual Heat Removal Discharge Interfacing System LOCA (ISLOCA)	$2.3x10^{-9}$
IE-RHR-HL-V	Residual Heat Removal Hot Leg ISLOCA	$8.9x10^{-10}$
IE-RHR-SUC-V	Residual Heat Removal Suction ISLOCA	$7.7x10^{-7}$
IE-SI-CLDIS-V	Safety Injection Cold Leg Discharge ISLOCA	$7.8x10^{-12}$

E.3.2 Event Sequences

The event sequences used in this example represent the response of the plant in terms of an initiating event followed by a combination of system, function, and operation failures or successes, that leads to an end state. This end state can be successful mitigation of the challenge, resulting in no core damage or release, or can be more severe, including core damage and release of radionuclides. There are two key issues that warrant discussion with respect to the construction of the event sequences: the design of the top events and the design of the sequence end states.

E.3.2.1 Event Sequence Top Events

In the Framework approach, the LBEs are sequences selected from the PRA at the ‘systemic’ level in terms of front-line systems that provide the needed safety functions. The specific level of detail for these ‘front-line’ systems for different technologies could be determined in the technology-specific regulatory guides.

Table E-4 shows the top events used in the front-line event trees that are questioned directly as a result of an initiating event for this PWR example. Note that additional event trees are often questioned, resulting in additional top events (not shown).

Table E-4 Event sequence top events

Top Event	Description	LODCA	LLOCA	LOCCW-A	LOESW	LOOP	MLOCA	SGTR	SLOCA	TRANS	RHR-DIS-V	RHR-HL-V	RHR-SUC-V	SI-CLDIS-V
ACC	RCS Accumulators Re-flood on Demand		Y				Y							
AFW	Auxiliary Feedwater System Operates on Demand	Y		Y	Y	Y		Y	Y	Y				
COOL DOWN	Various RCS Cooldown Actions	Y			Y				Y	Y				
DEPRES	Various RCS Depressurization Actions							Y						
EPS	Emergency Onsite Power Available Following LOOP					Y								
FAB	Feed and Bleed Operates on Demand (Non-safety-related, Set to 1.0 in this example)	Y			Y	Y		Y	Y	Y				
HPI	High Pressure Injection Operates on Demand	Y			Y	Y	Y		Y	Y				
HPR	High Pressure Recirculation Operates on Demand	Y			Y	Y	Y		Y	Y				
LPI	Low Pressure Injection Operates on Demand		Y		Y									
LPR	Low Pressure Recirculation Operates on Demand		Y		Y			Y						
MFW	Main Feedwater Operates Following a Reactor Trip (Non-safety-related, Set to 1.0 in this example)			Y	Y			Y	Y	Y				
OPR-02H, OPR-06H	Operator Recovers Offsite Power is 2 or 6 Hours					Y								
OPR-Detects	Operator Detects V-Sequence										Y	Y	Y	Y
OPR-ISOL	Operator Isolates V-Sequence										Y	Y	Y	Y

Table E-4 Event sequence top events

Top Event	Description	LODCA	LLOCA	LOCCW-A	LOESW	LOOP	MLOCA	SGTR	SLOCA	TRANS	RHR-DIS-V	RHR-HL-V	RHR-SUC-V	SI-CLDIS-V
PORV	Power Operated Relief Valves Close on Demand	Y		Y		Y				Y				
PZR	Operator Depressurizes RCS					Y								
RCP Seals	Reactor Coolant Pump Seals Maintain Pressure Integrity				Y	Y								
RHR	Residual Heat Removal Operates on Demand	Y				Y		Y	Y	Y				
RPS	Reactor Protection System Operates on Demand	Y		Y	Y	Y	Y	Y	Y	Y				
SSC	Secondary Side Cooling					Y								
SG-ISOL	Operator Isolates Affected SG							Y						

In addition to the reactivity control, heat removal and, pressure and inventory functions identified above, top events addressing containment-related functions are also included.

Table E-5 shows ten different types of top events that are used in the example PRA to model accident progression subsequent to core damage.

Table E-5 Containment related top events

Top Event	Description
CIF	Containment Isolation
RCSDEP-LATE	No Late RCS Depressurization
SGDEP-LATE	No Late Secondary Depressurization
ISGTR	No Induced Steam Generator Tube Rupture
H2	No Containment Failure due to Hydrogen Burn
PREVB-INVREC	In Vessel Recovery before Vessel Breach
RCSPIPE-MELT	No Melt of Surge Line, Hot Legs

Table E-5 Containment related top events

Top Event	Description
DCH	No Containment Failure due to Direct Containment Heating (DCH) with Hydrogen Burn
CMTSTF	No Containment Melt-through via Seal Table Failure
LER	No Large Early Release

Of these top events, ISGTR and LER are each further classified so that different failure probabilities can be applied depending on the specific event sequences modeled in the containment event trees (CETs). For instance, the failure probabilities for induced steam generator tube rupture depend on specific accident conditions, such as RCS condition (i.e., RCS intact, seal LOCA, or stuck-open relief valve), RCS depressurization, steam generator depressurization, and flaws in steam generator tubing; hence, situation-specific top events for ISGTR are used for induced steam generator tube rupture events. On the other hand, the LER top event is further classified based on the accident type (e.g., SBO isolation failure, non-SBO isolation failure, SGTR, ISGTR, etc.) and condition (e.g., RCS pressure, secondary pressure, etc.), so that the appropriate split fractions for large early release can be applied depending on the specific circumstances.

E.3.2.2 Event Sequence End States

As stated in Chapter 7, a key mission of the PRA analysis is to generate a complete set of accident sequences. These sequences are the foundation for many of the PRA's Framework applications and are a direct input into the determination of the proposed design's level of safety. They include a spectrum of releases from minor to major, and sequences that address conditions less than the core damage sequences of the current reactors and conditions similar to current reactor core damage sequences.

In this PWR example, both core damage and non-core damage sequences are included.

E.3.3 Dose End States

For event sequences with the 95th percentile frequency larger than $1x10^{-7}$ per year, Chapter 6 of the Framework requires the dose (duration and location specific to each dose category) to meet the frequency-consequence curve. In this example, the one mile 24 hour consequence analysis was performed for all core damage sequences with a point estimate frequency of $1x10^{-8}$ per year or greater. A separate evaluation was performed for the one core damage sequence that has a 95th percentile frequency greater than $1x10^{-5}$ per year (i.e., Dose Category 2 as shown in Table E.1). Event sequences that do not result in core damage are set to an end state of <1 mREM. This end state was selected in order to recognize that there is a potential for radionuclide release due to activity in the reactor coolant system that results from normal operation. Additional analysis would be needed to determine the actual boundary dose levels for these non-core-damage events.

E.3.4 Nomenclature

The PWR example model is constructed using the SAPHIRE code and is a small event tree, fault tree linked modeled. Each initiating event has a dedicated front-line event tree. The end states for these fault trees either terminate within this initial event tree (e.g., LOOP 01: Loss of offsite power with all functions successful) or transfer to one or more additional event trees that address additional functional requirements (e.g., LOOP 18-06-11-01: Loss of offsite power with station blackout (1^{st} tree Sequence 18), Stage two failure of the RCP seals with no LOOP recovery (2^{nd} tree, Sequence 06), H2 combustion resulting in containment failure (3^{rd} tree, Sequence 11), and a mapping tree that assigns the end state to a boundary dose (4^{th} tree, Sequence 01)).

E.4 Example: Identification of LBEs

Following the steps identified in Section E.2, the identification of the LBEs and safety significant SSCs for the example PWR is described below.

Step 1 Modify the PRA to only credit those mitigating functions that are to be considered safety significant.

The term 'safety significant' is used in the Framework to designate those systems needing special treatment. The type of special treatment varies dependent on the function the SSC needs to fulfill. As stated in Chapter 6, the treatment ensures that the SSC will perform reliably (as postulated in the PRA) under the conditions (temperature, pressure, radiation, etc.) assumed to prevail in the event scenarios for which the SSC's successful function is credited in the risk analysis. As a minimum, credited SSCs will be required to have a reliability performance goal.

It is the designer's decision as to what SSCs will be considered safety-significant as long as the Framework's acceptance criteria are met. This determination could be accomplished through an iterative approach, where the impact on the selection of LBEs is evaluated with a proposed set of safety significant SSCs, then re-assessed with another set of safety significant SSCs, until the desire set of LBEs and other design objectives are achieved.

As the example used in this appendix is an analysis of a currently licensed PWR, the function of main feedwater providing adequate flow post trip and the function of performing feed and bleed were set to 1.0, or guaranteed failure, because these functions are typically considered to be non-safety-related. For new reactors, all SSCs could be included in the scope of the licensing basis PRA. However, this would require, as a minimum, reliability performance goals for those credited functions and potentially other special treatment requirements.

It should be noted that functions that have an adverse effect on plant risk as a result of miss operation or malfunction associated with an anticipated response cannot be removed through this classification process. These adverse effects need to be included in the PRA and, at a minimum, monitored to ensure actual performance is consistent with their reliability and availability goals. To illustrate this point, the feed and bleed classification example is further examined. As stated earlier in this section, the feed and bleed function is typically considered to be non-safety-related and therefore it would be set to guaranteed failure. The Framework approach for non-safety significant components is to remove credit for its function from the PRA. In the case of feed and bleed, this approach removes credit for the operator action to initiate feed and bleed and removes credit for the function of several plant components including the power operated relief valves (PORVs) to manually operate on demand. Although the classification implies that the function of manually opening the PORVs is not credited, it does not mean that the function of the PORVs reseating after an open demand (e.g., opening to relieve reactor coolant system (RCS) pressure following a pressure transient) can be eliminated. If the PORVs can open as a result of a transient for a function other than that needed to support the feed and bleed then their ability to re-close in order to re-establish the RCS pressure boundary is an anticipated response. Failure of the PORVs to re-close is a potential adverse impact of having installed PORVs and is a safety significant function that needs to be included in the PRA regardless of the feed and bleed safety classification. If the sole purpose of the PORVs was to support feed and bleed (not the case for the current fleet of PWRs) then the removal of both the open and close function might be appropriate. But in the current example, the anticipated response of the PORVs needs to be included. The above discussion illustrates that care is required when removing non safety significant functions from the PRA. Functions that add to the reliability, redundancy or diversity can be removed if the miss operation of these functions does not impact the anticipated response of other credited systems and the required acceptance criteria (e.g., frequency-consequence curve limits, defense-in-depth requirements, etc.) can be maintained. However, the adverse impact of equipment cannot be removed if the miss operation of the equipment would exasperate the response beyond that due to the lost of the credit for the removed function.

As stated earlier, those SSCs whose functionality plays a role in meeting the acceptance criteria imposed on the LBEs define the set of safety significant SSCs. The SSCs of interest are those that influence the frequency or consequence of the LBEs, or both. All functions included in the PRA have the potential to influence the frequency of LBE sequences and many influence the consequences. Therefore, any function and the associated SSCs included in the PRA used to develop the set of LBEs is safety significant unless it has been set to 1.0 or guaranteed failure. As stated above, the designer can remove mitigation functions from the PRA in order to reduce the set of safety significant SSCs. However, the resulting PRA must meet the F-C curve and the defense-in-depth deterministic requirements.

Note that in this example only the main feedwater and the feed and bleed functions were set to guaranteed failure. It is likely that there are other non-safety-related functions included within the example PRA, but these were not explicitly identified and removed from the model for this appendix.

Step 2 Determine the point estimate frequency for each resulting event sequence from the quantification of the modified PRA.

This step establishes the complete set of event sequences that will be processed to determine the LBEs. An quantification truncation limit of $1x10^{-15}$ per year was used. In this example, the 13 initiating events produce a total of 1,536 sequences. Table E-6 summarizes the results.

Table E-6 Accident sequences

Initiating Event	Number of Sequences	Number of Sequences point estimate > $1x10^{-8}$	Number of Sequences 95th > $1x10^{-7}$
IE-LDCA	64	9	7
IE-LLOCA	10	1	1
IE-LOCCW-A	141	5	3
IE-LOESW	190	6	6
IE-LOOP	829	47	24
IE-MLOCA	13	2	2
IE-SGTR	68	15	13
IE-SLOCA	84	4	4
IE-TRANS	121	18	16
IE-V-RHR-DIS	4	0	0
IE-V-RHR-HLDIS	4	0	0
IE-V-RHR-SUC	4	3	3
IE-V-SI-CLDIS	4	0	0
Total	**1,536**	**110**	**79**

The process used to reduce the number of sequences from 1536 to 110 to 79 is further described in Steps 3 and 4 below.

Step 3 For sequences with point estimate frequencies equal to or greater than $1x10^{-8}$, determine the mean and 95th percentile frequency.

The frequency used to determine whether an event sequence remains within scope of the LBE selection process is based the 95th percentile. Therefore, the mean and 95th percentile are determined in this step.

In the example, an uncertainty analysis is performed on the 110 sequences that were determined to be in scope by Step 2. Of these sequences, 79 sequences have a 95th percentile equal to or larger than $1x10^{-7}$ per year. The 31 sequences that are screened (those sequences less than $1x10^{-7}$) are shaded in Table E-7.

Note that the characterization of the dose (exposure time and distance) associated with the sequence end state is dependent on the 95th percentile frequency of the sequence. In this example, the 1 mile 24 hour dose was determined for all core damage sequences with a mean frequency greater than $1x10^{-8}$ per year. These are indicated by the term “1 mile” in Table E-7. One core damage event sequence, LOESW 04-01-01, has a 95th percentile frequency greater than $1x10^{-5}$ per year and is therefore considered to be in the Infrequent category and requires an assessment of the worst 2-hour dose at the exclusion area boundary. This dose is annotated by the term "EAB" in Table E-7.

Table E-7 Accident sequences for sequences with a point estimate > $1x10^{-8}$/yr

Initiating Event	Sequence	Description	Point Estimate (per year)	Mean (per year)	95th (per year)	Mean Dose (REM)	95th Dose (REM)
LDCA	01	Loss of a DC bus with all remaining systems successful	$2.5x10^{-3}$	$2.51x10^{-3}$	$1.0x10^{-2}$	<1mR	<1mR
LDCA	10-01-01-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	$4.1x10^{-8}$	$3.8x10^{-8}$	$1.6x10^{-7}$	1 mile 0.6R	1 mile 1.2R
LDCA	10-01-03-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	$3.6x10^{-8}$	$3.28x10^{-8}$	$1.4x10^{-7}$	1 mile 0.6R	1 mile 1.2R
LDCA	10-01-06-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	$4.2x10^{-8}$	$3.9x10^{-8}$	$1.7x10^{-7}$	1 mile 0.6R	1 mile 1.2R
LDCA	10-01-07-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	$3.8x10^{-6}$	$3.5x10^{-8}$	$1.5x10^{-7}$	1 mile 0.6R	1 mile 1.2R
LDCA	10-02-01-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	$3.8x10^{-8}$	$3.5x10^{-8}$	$1.5x10^{-7}$	1 mile 0.6R	1 mile 1.2R
LDCA	10-02-02-01	Loss of a DC bus with no secondary heat removal and induced SGTR	$1.8x10^{-8}$	$1.6x10^{-8}$	$7.2x10^{-8}$	1 mile 100R	1 mile 356R
LDCA	10-02-03-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	$4.1x10^{-8}$	$3.8x10^{-8}$	$1.7x10^{-7}$	1 mile 0.6R	1 mile 1.2R
LDCA	10-02-04-01	Loss of a DC bus with no secondary heat removal and induced SGTR	$1.5x10^{-8}$	$1.3x10^{-8}$	$5.8x10^{-8}$	1 mile 100R	1 mile 356R
LLOCA	01	LLOCA with all systems successful	$5.0x10^{-6}$	$5.1x10^{-6}$	$1.9x10^{-5}$	<1mR	<1mR

Table E-7 Accident sequences for sequences with a point estimate > $1x10^{-8}$/yr

Initiating Event	Sequence	Description	Point Estimate (per year)	Mean (per year)	95^{th} (per year)	Mean Dose (REM)	95^{th} Dose (REM)
LOCCW-A	01	Loss of Component Cooling with RCP seal failure	$2.0x10^{-4}$	$2.0x10^{-4}$	$9.6x10^{-4}$	<1mR	<1mR
LOCCW-A	02	Loss of Component Cooling with RCP seal failure	$4.8x10^{-7}$	$4.4x10^{-7}$	$1.8x10^{-6}$	<1mR	<1mR
LOCCW-A	07	Loss of Component Cooling with failure to cooldown	$2.0x10^{-7}$	$2.0x10^{-7}$	$1.0x10^{-6}$	<1mR	<1mR
LOESW	01	Loss of Essential Reactor Cooling Water with RCPs remaining intact	$4.0x10^{-4}$	$4.1x10^{-4}$	$1.92x10^{-3}$	<1mR	<1mR
LOESW	02	Loss of Essential Reactor Cooling with RCP seal failure	$7.6x10^{-5}$	$8.1x10^{-5}$	$4.1x10^{-4}$	<1mR	<1mR
LOESW	03-01-01	Loss of Essential Reactor Cooling with RCP seal failure. Although ERCW is recovered, low pressure recirculation fails.	$2.6x10^{-8}$	$2.9x10^{-8}$	$1.28x10^{-7}$	1 mile 0.4R	1 mile 0.5R
LOESW	04-01-01	Loss of Essential Reactor Cooling with RCP Seal failure. With failure to recover ERCW, low pressure recirculation fails.	$2.6x10^{-5}$	$2.5x10^{-5}$	$1.2x10^{-4}$	EAB NA 1 mile 0.4R	EAB 7R 1 mile 0.5R
LOESW	06-01-01	Loss of Essential Reactor Cooling with RCP Seal failure. Low pressure injection fails, Essential Reactor Cooling is recovered but high pressure recirculation fails.	$1.3x10^{-8}$	$1.5x10^{-8}$	$6.1x10^{-8}$	1 mile 0.4R	1 mile 0.5R
LOESW	09	Loss of Essential Reactor Cooling with failure to cooldown	$4.0x10^{-7}$	$3.9x10^{-7}$	$2.0x10^{-6}$	<1mR	<1mR

Table E-7 Accident sequences for sequences with a point estimate > $1x10^{-8}$/yr

Initiating Event	Sequence	Description	Point Estimate (per year)	Mean (per year)	95th (per year)	Mean Dose (REM)	95th Dose (REM)
LOESW	10	Loss of Essential Reactor Cooling with ERCW recovery and RCP seal failure	$7.6x10^{-8}$	$7.8x10^{-9}$	$3.3x10^{-7}$	<1mR	<1mR
LOESW	13-01-01	Loss of Essential Reactor Cooling with RCP Seal failure. RCS cooldown fails and cooling water is not recovered.	$2.6x10^{-8}$	$2.5x10^{-8}$	$7.7x10^{-8}$	1 mile 0.6R	1 mile 1.2R
LOOP	01	LOOP with all systems successful, 2 hour recovery, no inventory challenge	$3.3x10^{-2}$	$3.3x10^{-2}$	$8.5x10^{-2}$	<1mR	<1mR
LOOP	02-01	LOOP with RCP seal failure	$1.6x10^{-6}$	$2.4x10^{-6}$	$9.4x10^{-6}$	<1mR	<1mR
LOOP	02-02-01	LOOP with RCP seal failure	$2.6x10^{-7}$	$2.6x10^{-7}$	$1.0x10^{-6}$	<1mR	<1mR
LOOP	02-03	LOOP with RCP seal failure	$1.5x10^{-7}$	$1.1x10^{-7}$	$4.7x10^{-7}$	<1mR	<1mR
LOOP	02-04-01-01	LOOP with RCP seal failure and failure of high pressure recirculation	$1.0x10^{-8}$	$8.3x10^{-9}$	$2.4x10^{-8}$	1 mile 0.4R	1 mile 0.5R
LOOP	02-06-01	LOOP, 2 hour recovery, inventory challenged (PORVs fail to close) and RCS depressurization to low pressure injection fails	$1.3x10^{-8}$	$1.8x10^{-8}$	$6.7x10^{-8}$	<1mR	<1mR
LOOP	03	LOOP, 2 hour recovery, inventory challenged (PORVs fail to close)	$1.2x10^{-7}$	$1.7x10^{-7}$	$6.0x10^{-7}$	<1mR	<1mR
LOOP	10	LOOP, 2 hr recovery fails, PORVs fail to close, high pressure recirc successful	$7.2x10^{-8}$	$6.6x10^{-8}$	$2.6x10^{-7}$	<1mR	<1mR
LOOP	17-01-01-01	LOOP with AFW failure	$2.4x10^{-8}$	$2.6x10^{-8}$	$1.1x10^{-7}$	1 mile 256R	1 mile 927R

Table E-7 Accident sequences for sequences with a point estimate > 1x10^{-8}/yr

Initiating Event	Sequence	Description	Point Estimate (per year)	Mean (per year)	95th (per year)	Mean Dose (REM)	95th Dose (REM)
LOOP	17-01-03-01	LOOP with AFW failure	2.1x10^{-8}	2.3x10^{-8}	9.3x10^{-8}	1 mile 256R	1 mile 927R
LOOP	17-01-06-01	LOOP with AFW failure	2.5x10^{-8}	2.7x10^{-8}	1.1x10^{-7}	1 mile 256R	1 mile 927R
LOOP	17-01-07-01	LOOP with AFW failure	2.2x10^{-8}	2.4x10^{-8}	1.0x10^{-8}	1 mile 256R	1 mile 927R
LOOP	17-03-01-01	LOOP with AFW failure	2.2x10^{-8}	2.4x10^{-8}	9.9x10^{-8}	1 mile 256R	1 mile 927R
LOOP	17-03-02	LOOP with AFW failure	1.0x10^{-8}	1.2x10^{-8}	4.8x10^{-8}	1 mile 256R	1 mile 927R
LOOP	17-03-03-01	LOOP with AFW failure	2.4x10^{-8}	2.6x10^{-8}	1.1x10^{-7}	1 mile 256R	1 mile 927R
LOOP	18-01	SBO with secondary heat removal, power recovery and RCP seal integrity maintained	9.8x10^{-6}	1.4x10^{-5}	5.5x10^{-5}	<1mR	<1mR
LOOP	18-02	SBO with secondary heat removal, power recovery and RCP seal integrity maintained	2.8x10^{-6}	3.9x10^{-6}	1.5x10^{-5}	<1mR	<1mR
LOOP	18-03-05-01	SBO with battery depletion	2.8x10^{-8}	3.9x10^{-8}	1.5x10^{-7}	1 mile 376R	1 mile 1060R
LOOP	18-03-06-01	SBO with battery depletion	6.9x10^{-7}	9.7x10^{-7}	3.8x10^{-6}	1 mile 376R	1 mile 1060R
LOOP	18-03-10-01	SBO with battery depletion	2.8x10^{-8}	3.9x10^{-8}	1.5x10^{-7}	1 mile 376R	1 mile 1060R
LOOP	18-03-11-01	SBO with battery depletion	6.9x10^{-7}	9.7x10^{-7}	3.8x10^{-6}	1 mile 376R	1 mile 1060R
LOOP	18-04-01	SBO with secondary heat removal, RCP seal failure and power recovery	2.4x10^{-6}	2.2x10^{-6}	1.0x10^{-5}	<1mR	<1mR
LOOP	18-04-07-01 -01	SBO with secondary heat removal, RCP seal failure and power recovery. Both high and low pressure injection fail.	1.9x10^{-8}	1.4x10^{-8}	4.7x10^{-8}	1 mile 376R	1 mile 1060R

Table E-7 Accident sequences for sequences with a point estimate > 1x10^{-8}/yr

Initiating Event	Sequence	Description	Point Estimate (per year)	Mean (per year)	95th (per year)	Mean Dose (REM)	95th Dose (REM)
LOOP	18-05	SBO with secondary heat removal, RCP seal failure and power recovery	7.1x10^{-7}	6.4x10^{-7}	2.5x10^{-6}	<1mR	<1mR
LOOP	18-06-06-01	SBO with secondary heat removal, RCP seal failure and no power recovery	1.7x10^{-7}	1.8x10^{-7}	7.0x10^{-7}	1 mile 376R	1 mile 1060R
LOOP	18-06-11-01	SBO with secondary heat removal, RCP seal failure and no power recovery	1.7x10^{-7}	1,8x10^{-7}	7.0x10^{-7}	1 mile 376R	1 mile 1060R
LOOP	18-07-01	SBO with secondary heat removal, RCP seal failure and power recovery	1.2x10^{-7}	1.7x10^{-7}	6.5x10^{-7}	<1mR	<1mR
LOOP	18-08	SBO with secondary heat removal, RCP seal failure and power recovery	3.5x10^{-8}	4.7x10^{-8}	1.8x10^{-7}	<1mR	<1mR
LOOP	18-10-01	SBO with secondary heat removal, RCP seal failure and power recovery	2.5x10^{-8}	2.1x10^{-8}	7.4x10^{-8}	<1mR	<1mR
LOOP	18-11	SBO with secondary heat removal, RCP seal failure and EDG recovery	1.5x10^{-8}	1.4x10^{-8}	4.1x10^{-8}	<1mR	<1mR
LOOP	18-40-01	SBO with secondary heat removal, PORV fails to re-close and power recovery	1.6x10^{-8}	1.8x10^{-8}	7.4x10^{-8}	<1mR	<1mR
LOOP	18-41	SBO with secondary heat removal, PORV fails to re-close and EDG recovery	1.8x10^{-8}	2.8x10^{-8}	8.1x10^{-8}	<1mR	<1mR
LOOP	18-42-05-01	SBO with secondary heat removal, PORV fails to re-close, no power recovery, containment failure due to H2	1.5x10^{-8}	1.8x10^{-8}	6.5x10^{-8}	1 mile 376R	1 mile 1060R

Table E-7 Accident sequences for sequences with a point estimate > 1x10^{-8}/yr

Initiating Event	Sequence	Description	Point Estimate (per year)	Mean (per year)	95th (per year)	Mean Dose (REM)	95th Dose (REM)
LOOP	18-43-03-01 -01-01	SBO with secondary heat removal, PORV fails to re-close, no power recovery, containment failure due to seal table	1.8x10^{-8}	2.6x10^{-8}	9.9x10^{-8}	1 mile 256R	1 mile 927R
LOOP	18-43-03-01 -03-01	SBO without secondary heat removal	1.6x10^{-8}	2.2x10^{-8}	8.6x10^{-8}	1 mile 256R	1 mile 927R
LOOP	18-43-03-01 -06-01	SBO with failure of secondary heat removal, RCP seal failure and no power recovery	1.8x10^{-8}	2.6x10^{-8}	1.0x10^{-7}	1 mile 256R	1 mile 927R
LOOP	18-43-03-01 -07-01	SBO without secondary heat removal	1.7x10^{-8}	2.4x10^{-8}	9.2x10^{-8}	1 mile 256R	1 mile 927R
LOOP	18-43-03-03 -03-01	SBO without secondary heat removal	1.8x10^{-8}	2.6x10^{-8}	1.0x10^{-7}	1 mile 256R	1 mile 927R
LOOP	18-44	SBO with failure of secondary heat removal, RCP seal failure and power recovery within 1 hour	1.4x10^{-7}	1.7x10^{-7}	6.5x10^{-7}	<1mR	<1mR
LOOP	18-45-01-06 -01	SBO without secondary heat removal	1.6x10^{-8}	2.1x10^{-8}	8.6x10^{-8}	1 mile 256R	1 mile 927R
LOOP	18-45-01-13 -01	SBO without secondary heat removal	1.4x10^{-8}	1.9x10^{-8}	7.5x10^{-8}	1 mile 256R	1 mile 927R
LOOP	18-45-01-20 -01	SBO without secondary heat removal	1.7x10^{-8}	2.2x10^{-8}	8.8x10^{-8}	1 mile 256R	1 mile 927R
LOOP	18-45-01-25 -01	SBO without secondary heat removal	1.5x10^{-8}	2.0x10^{-8}	8.0x10^{-8}	1 mile 256R	1 mile 927R
LOOP	18-45-02-06 -01	SBO without secondary heat removal	1.5x10^{-8}	2.0x10^{-8}	8.0x10^{-8}	1 mile 256R	1 mile 927R

Table E-7 Accident sequences for sequences with a point estimate > $1x10^{-8}$/yr

Initiating Event	Sequence	Description	Point Estimate (per year)	Mean (per year)	95th (per year)	Mean Dose (REM)	95th Dose (REM)
LOOP	18-45-02-12 -01	SBO without secondary heat removal	$1.7x10^{-8}$	$2.2x10^{-8}$	$8.7x10^{-8}$	1 mile 256R	1 mile 927R
LOOP	19-08	ATWS with all systems successful (MFW not credited)	$4.0x10^{-8}$	$4.2x10^{-8}$	$1.5x10^{-7}$	<1mR	<1mR
LOOP	19-09	ATWS with failure of PORVs to re-close (MFW not credited)	$1.2x10^{-8}$	$1.2x10^{-8}$	$5.0x10^{-8}$	<1mR	<1mR
MLOCA	01	MLOCA with all systems successful	$4.0x10^{-5}$	$4.1x10^{-5}$	$1.5x10^{-4}$	<1mR	<1mR
MLOCA	02-01-01	MLOCA with high pressure recirculation failure	$1.0x10^{-7}$	$1.0x10^{-7}$	$4.3x10^{-7}$	1 mile 0.6R	1 mile 1.2R
SGTR	01	SGTR with all systems successful	$4.0x10^{-3}$	$4.0x10^{-3}$	$1.6x10^{-2}$	<1mR	<1mR
SGTR	02	SGTR with failure to isolate the ruptured SG	$4.8x10^{-5}$	$5.0x10^{-5}$	$2.4x10^{-4}$	<1mR	<1mR
SGTR	03-01-01	SGTR with failure to isolate and failure of RHR	$9.7x10^{-8}$	$9.5x10^{-8}$	$4.4x10^{-7}$	1 mile 36R	1 mile 88R
SGTR	03-02-01	SGTR with failure to isolate and failure of RHR	$1.2x10^{-7}$	$1.2x10^{-7}$	$5.4x10^{-7}$	1 mile 36R	1 mile 88R
SGTR	04-01-01	SGTR with failure to depressurize to RHR entry condition	$2.2x10^{-8}$	$2.0x10^{-4}$	$9.5x10^{-8}$	1 mile 36R	1 mile 88R
SGTR	04-02-01	SGTR with failure to isolate and failure to depressurize to RHR entry conditions	$2.6x10^{-8}$	$2.5x10^{-8}$	$1.1x10^{-7}$	1 mile 36R	1 mile 88R
SGTR	05-01-01	SGTR with failure to depressurize < SG RV setpoints	$5.1x10^{-8}$	$5.8x10^{-8}$	$2.4x10^{-7}$	1 mile 36R	1 mile 88R
SGTR	05-02-01	SGTR with failure to depressurize to < SG RV setpoints	$6.2x10^{-8}$	$7.1x10^{-8}$	$2.9x10^{-7}$	1 mile 36R	1 mile 88R
SGTR	06	SGTR with failure to depressurize before SG reliefs lift	$4.4x10^{-5}$	$4.4x10^{-5}$	$2.1x10^{-4}$	<1mR	<1mR

Table E-7 Accident sequences for sequences with a point estimate > 1x10^{-8}/yr

Initiating Event	Sequence	Description	Point Estimate (per year)	Mean (per year)	95th (per year)	Mean Dose (REM)	95th Dose (REM)
SGTR	07	SGTR with failure to isolate the ruptured SG and failure to depressurize before SG reliefs lift	5.5x10^{-7}	5.5x10^{-7}	2.4x10^{-6}	<1mR	<1mR
SGTR	08-01-01	SGTR with failure to depressurize before SG reliefs lift, failure to isolate the rupture SG and failure or RHR	1,6x10^{-7}	1.7x10^{-7}	5.8x10^{-7}	1 mile 36R	1 mile 88R
SGTR	11-01-01	SGTR with failure to depressurize before and after SG reliefs lift	4.0x10^{-7}	3.85x10^{-7}	1.8x10^{-6}	1 mile 36R	1 mile 88R
SGTR	11-02-01	SGTR with failure to depressurize before and after SG reliefs lift	4.8x10^{-7}	4.7x10^{-7}	2.2x10^{-6}	1 mile 36R	1 mile 88R
SGTR	12	SGTR with failure of high pressure injection	1.5x10^{-8}	1.6x10^{-8}	6.9x10^{-8}	<1mR	<1mR
SGTR	43-01	SGTR with failure of secondary heat removal	4.1x10^{-7}	4.6x10^{-7}	1.9x10^{-6}	1 mile 105R	1 mile 366R
SLOCA	01	SLOCA with all systems successful	4.0x10^{-4}	4.1x10^{-4}	1.9x10^{-3}	<1mR	<1mR
SLOCA	02	SLOCA with the failure of RHR and successful high pressure recirculation	1.6x10^{-6}	1.6x10^{-6}	7.9x10^{-6}	<1mR	<1mR
SLOCA	04	SLOCA with failure of cooldown and high pressure recirculation	4.0x10^{-7}	3.9x10^{-7}	2.0x10^{-6}	<1mR	<1mR
SLOCA	03-01-01	SLOCA with the failure of RHR and high pressure recirculation	1.8x10^{-7}	1.9x10^{-7}	8.7x10^{-7}	1 mile 0.6R	1 mile 1.2R
TRANS	01	TRANS with all system successful	7.0x10^{-1-1}	7.0x10^{-1}	1.3	<1mR	<1mR
TRANS	02	TRANS with failure PORVs to reseat	5.0x10^{-7}	4.3x10^{-7}	1.4x10^{-6}	<1mR	<1mR

Table E-7 Accident sequences for sequences with a point estimate > 1x10^{-8}/yr

Initiating Event	Sequence	Description	Point Estimate (per year)	Mean (per year)	95th (per year)	Mean Dose (REM)	95th Dose (REM)
TRANS	18-01-01-01	TRANS with failure of secondary heat removal	4.5x10^{-7}	4.7x10^{-7}	2.0x10^{-6}	1 mile 0.6R	1 mile 1.2R
TRANS	18-01-02-01	TRANS with failure of secondary heat removal & induced SGTR	1.1x10^{-8}	1.2x10^{-8}	5.0x10^{-8}	1 mile 100R	1 mile 356R
TRANS	18-01-03-01	TRANS with failure of secondary heat removal	3.9x10^{-7}	4.1x10^{-7}	1.7x10^{-6}	1 mile 0.6R	1 mile 1.2R
TRANS	18-01-04-01	TRANS with failure of secondary heat removal & induced SGTR	6.2x10^{-8}	6.9x10^{-8}	2.9x10^{-7}	1 mile 100R	1 mile 356R
TRANS	18-01-06-01	TRANS with failure of secondary heat removal	4.6x10^{-7}	4.8x10^{-7}	2.0x10^{-6}	1 mile 0.6R	1 mile 1.2R
TRANS	18-01-07-01	TRANS with failure of secondary heat removal	4.1x10^{-7}	4.4x10^{-7}	1.8x10^{-6}	1 mile 0.6R	1 mile 1.2R
TRANS	18-01-08-01	TRANS with failure of secondary heat removal & induced SGTR	3.7x10^{-8}	3.9x10^{-8}	1.6x10^{-7}	1 mile 100R	1 mile 356R
TRANS	18-02-01-01	TRANS with failure of secondary heat removal	4.1x10^{-7}	4.4x10^{-7}	1.8x10^{-6}	1 mile 0.6R	1 mile 1.2R
TRANS	18-02-02-01	TRANS with failure of secondary heat removal & induced SGTR	2.0x10^{-7}	2.2x10^{-7}	9.3x10^{-7}	1 mile 100R	1 mile 356R
TRANS	18-02-03-01	TRANS with failure of secondary heat removal	4.5x10^{-7}	4.7x10^{-7}	2.-x10^{-6}	1 mile 0.6R	1 mile 1.2R
TRANS	18-02-04-01	TRANS with failure of secondary heat removal & induced SGTR	1.6x10^{-7}	1.8x10^{-7}	7.5x10^{-7}	1 mile 100R	1 mile 356R
TRANS	19-08	ATWS with all systems successful (MFW not credited)	1.4x10^{-6}	1.4x10^{-6}	4.8x10^{-6}	<1mR	<1mR

Table E-7 Accident sequences for sequences with a point estimate > 1x10^{-8}/yr

Initiating Event	Sequence	Description	Point Estimate (per year)	Mean (per year)	95th (per year)	Mean Dose (REM)	95th Dose (REM)
TRANS	19-09	ATWS with stuck open PORVs	4.3x10^{-7}	4.2x10^{-7}	1.9x10^{-6}	<1mR	<1mR
TRANS	19-14-01-01	ATWS with failure to emergency borate	2.9x10^{-8}	2.9x10^{-8}	1.3x10^{-7}	1 mile 0.4R	1 mile 0.5R
TRANS	19-16-01-01 -01	ATWS with RCS pressure boundary failure	3.4x10^{-8}	3.4x10^{-8}	1.4x10^{-7}	1 mile 0.4R	1 mile 0.5R
TRANS	19-16-03-01 -01	ATWS with RCS pressure boundary failure	2.2x10^{-8}	2.3x10^{-8}	9.0x10^{-8}	1 mile 0.4R	1 mile 0.5R
V-RHR-SUC	03	RHR Suction ISLOCA with successful mitigation	6.1x10^{-7}	4.0x10^{-6}	8.8x10^{-6}	<1mR	<1mR
V-RHR-SUC	04-01	RHR Suction ISLOCA with failure to isolate	1.2x10^{-8}	9.7x10^{-8}	1.4x10^{-8}	1 mile 998R	1 mile 3548R
V-RHR-SUC	05-01	RHR Suction ISLOCA with failure to diagnose	1.5x10^{-7}	9.9x10^{-7}	1.6x10^{-6}	1 mile 998R	1 mile 3548R

Step 4 Identify all PRA event sequences with a 95th percentile frequency $\geq$ 1x10^{-7} per year.

This step identifies those sequences that are to be included in the event class grouping process. Sequences having a 95th percentile frequency that is less than 1x10^{-7} per year are screened from the process. The remaining in-scope sequences are those that are not shaded in Table E-7. These sequences are shown in Table E-8.

Table E-8 PRA sequences grouping

Initiating Event	Sequence	Description	Event Class	Mean (per year)	95th (per year)	95th Dose (REM)
LDCA	01	Loss of a DC bus with all remaining systems successful	LBE-01	2.51x10^{-3}	1.0x10^{-2}	<1mR
LDCA	10-01-01-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	LBE-02	3.8x10^{-8}	1.6x10^{-7}	1.2R

Table E-8 PRA sequences grouping

Initiating Event	Sequence	Description	Event Class	Mean (per year)	95th (per year)	95th Dose (REM)
LDCA	10-01-03-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	LBE-02	3.28×10^{-8}	1.4×10^{-7}	1.2R
LDCA	10-01-06-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	LBE-02	3.9×10^{-8}	1.7×10^{-7}	1.2R
LDCA	10-01-07-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	LBE-02	3.5×10^{-8}	1.5×10^{-7}	1.2R
LDCA	10-02-01-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	LBE-02	3.5×10^{-8}	1.5×10^{-7}	1.2R
LDCA	10-02-03-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	LBE-02	3.8×10^{-8}	1.7×10^{-7}	1.2R
LLOCA	01	LLOCA with all systems successful	LBE-03	5.1×10^{-6}	1.9×10^{-5}	<1mR
LOCCW-A	01	Loss of Component Cooling with RCP seal failure	LBE-04	2.0×10^{-4}	9.6×10^{-4}	<1mR
LOCCW-A	02	Loss of Component Cooling with RCP seal failure	LBE-05	4.4×10^{-7}	1.8×10^{-6}	<1mR
LOCCW-A	07	Loss of Component Cooling with failure to cooldown	LBE-06	2.0×10^{-7}	1.0×10^{-6}	<1mR
LOESW	01	Loss os Essential Reactor Cooling Water with RCPs remaining intact	LBE-04	4.1×10^{-4}	1.92×10^{-3}	<1mR
LOESW	02	Loss of Essential Reactor Cooling with RCP seal failure	LBE-05	8.1×10^{-5}	4.1×10^{-4}	<1mR
LOESW	03-01-01	Loss of Essential Reactor Cooling with RCP seal failure. Although ERCW is recovered, low pressure recirculation fails.	LBE-07	2.9×10^{-8}	1.28×10^{-7}	7R

Table E-8 PRA sequences grouping

Initiating Event	Sequence	Description	Event Class	Mean (per year)	95th (per year)	95th Dose (REM)
LOESW	04-01-01	Loss of Essential Reactor Cooling with RCP Seal failure. With failure to recover ERCW, low pressure recirculation fails.	LBE-07	$2.5x10^{-5}$	$1.2x10^{-4}$	7R
LOESW	09	Loss of Essential Reactor Cooling with failure to cooldown	LBE-06	$3.9x10^{-7}$	$2.0x10^{-6}$	<1mR
LOESW	10	Loss of Essential Reactor Cooling with ERCW recovery and RCP seal failure	LBE-08	$7.8x10^{-9}$	$3.3x10^{-7}$	<1mR
LOOP	01	LOOP with all systems successful, 2 hour recovery, no inventory challenge	LBE-09	$3.3x10^{-2}$	$8.5x10^{-2}$	<1mR
LOOP	02-01	LOOP with RCP seal failure	LBE-10	$2.4x10^{-6}$	$9.4x10^{-6}$	<1mR
LOOP	02-02-01	LOOP with RCP seal failure	LBE-10	$2.6x10^{-7}$	$1.0x10^{-6}$	<1mR
LOOP	02-03	LOOP with RCP seal failure	LBE-10	$1.1x10^{-7}$	$4.7x10^{-7}$	<1mR
LOOP	03	LOOP, 2 hour recovery, inventory challenged (PORVs fail to close)	LBE-11	$1.7x10^{-7}$	$6.0x10^{-7}$	<1mR
LOOP	10	LOOP, 2 hr recovery fails, PORVs fail to close, high pressure recirc successful	LBE-11	$6.6x10^{-8}$	$2.6x10^{-7}$	<1mR
LOOP	17-01-01-01	LOOP with AFW failure	LBE-12	$2.6x10^{-8}$	$1.1x10^{-7}$	927R
LOOP	17-01-06-01	LOOP with AFW failure	LBE-12	$2.7x10^{-8}$	$1.1x10^{-7}$	927R
LOOP	17-03-03-01	LOOP with AFW failure	LBE-12	$2.6x10^{-8}$	$1.1x10^{-7}$	927R
LOOP	18-01	SBO with secondary heat removal, power recovery and RCP seal integrity maintained	LBE-13	$1.4x10^{-5}$	$5.5x10^{-5}$	<1mR
LOOP	18-02	SBO with secondary heat removal, power recovery and RCP seal integrity maintained	LBE-13	$3.9x10^{-6}$	$1.5x10^{-5}$	<1mR

Table E-8 PRA sequences grouping

Initiating Event	Sequence	Description	Event Class	Mean (per year)	95th (per year)	95th Dose (REM)
LOOP	18-03-05-01	SBO with battery depletion	LBE-14	$3.9x10^{-8}$	$1.5x10^{-7}$	1060R
LOOP	18-03-06-01	SBO with battery depletion	LBE-14	$9.7x10^{-7}$	$3.8x10^{-6}$	1060R
LOOP	18-03-10-01	SBO with battery depletion	LBE-14	$3.9x10^{-8}$	$1.5x10^{-7}$	1060R
LOOP	18-03-11-01	SBO with battery depletion	LBE-14	$9.7x10^{-7}$	$3.8x10^{-6}$	1060R
LOOP	18-04-01	SBO with secondary heat removal, RCP seal failure and power recovery	LBE-15	$2.2x10^{-6}$	$1.0x10^{-5}$	<1mR
LOOP	18-05	SBO with secondary heat removal, RCP seal failure and power recovery	LBE-15	$6.4x10^{-7}$	$2.5x10^{-6}$	<1mR
LOOP	18-06-06-01	SBO with secondary heat removal, RCP seal failure and no power recovery	LBE-16	$1.8x10^{-7}$	$7.0x10^{-7}$	1060R
LOOP	18-06-11-01	SBO with secondary heat removal, RCP seal failure and no power recovery	LBE-16	$1,8x10^{-7}$	$7.0x10^{-7}$	1060R
LOOP	18-07-01	SBO with secondary heat removal, RCP seal failure and power recovery	LBE-15	$1.7x10^{-7}$	$6.5x10^{-7}$	<1mR
LOOP	18-08	SBO with secondary heat removal, RCP seal failure and power recovery	LBE-15	$4.7x10^{-8}$	$1.8x10^{-7}$	<1mR
LOOP	18-43-03-01-06-01	SBO with failure of secondary heat removal, RCP seal failure and no power recovery	LBE-16	$2.6x10^{-8}$	$1.0x10^{-7}$	927R
LOOP	18-44	SBO with failure of secondary heat removal, RCP seal failure and power recovery within 1 hour	LBE-17	$1.7x10^{-7}$	$6.5x10^{-7}$	<1mR

Table E-8 PRA sequences grouping

Initiating Event	Sequence	Description	Event Class	Mean (per year)	95th (per year)	95th Dose (REM)
LOOP	19-08	ATWS with all systems successful (MFW not credited)	LBE-18	$4.2x10^{-8}$	$1.5x10^{-7}$	<1mR
MLOCA	01	MLOCA with all systems successful	LBE-19	$4.1x10^{-5}$	$1.5x10^{-4}$	<1mR
MLOCA	02-01-01	MLOCA with high pressure recirculation failure	LBE-20	$1.0x10^{-7}$	$4.3x10^{-7}$	1.2R
SGTR	01	SGTR with all systems successful	LBE-21	$4.0x10^{-3}$	$1.6x10^{-2}$	<1mR
SGTR	02	SGTR with failure ro isolate the ruptured SG	LBE-22	$5.0x10^{-5}$	$2.4x10^{-4}$	<1mR
SGTR	03-01-01	SGTR with failure to isolate and failure of RHR	LBE-23	$9.5x10^{-8}$	$4.4x10^{-7}$	88R
SGTR	03-02-01	SGTR with failure to isolate and failure of RHR	LBE-23	$1.2x10^{-7}$	$5.4x10^{-7}$	88R
SGTR	04-02-01	SGTR with failure to isolate and failure to depressurize to RHR entry conditions	LBE-23	$2.5x10^{-8}$	$1.1x10^{-7}$	88R
SGTR	05-01-01	SGTR with failure to depressurize < SG RV setpoints	LBE-24	$5.8x10^{-8}$	$2.4x10^{-7}$	88R
SGTR	05-02-01	SGTR with failure to depressurize to < SG RV setpoints	LBE-24	$7.1x10^{-8}$	$2.9x10^{-7}$	88R
SGTR	06	SGTR with failure to depressurize before SG reliefs lift	LBE-25	$4.4x10^{-5}$	$2.1x10^{-4}$	<1mR
SGTR	07	SGTR with failure to isolate the ruptured SG and failure to depressurize before SG reliefs lift	LBE-25	$5.5x10^{-7}$	$2.4x10^{-6}$	<1mR
SGTR	08-01-01	SGTR with failure to depressurize before SG reliefs lift, failure to isolate the rupture SG and failure or RHR	LBE-24	$1.7x10^{-7}$	$5.8x10^{-7}$	88R

Table E-8 PRA sequences grouping

Initiating Event	Sequence	Description	Event Class	Mean (per year)	95th (per year)	95th Dose (REM)
SGTR	11-01-01	SGTR with failure to depressurize before and after SG reliefs lift	LBE-24	3.85×10^{-7}	1.8×10^{-6}	88R
SGTR	11-02-01	SGTR with failure to depressurize before and after SG reliefs lift	LBE-24	4.7×10^{-7}	2.2×10^{-6}	88R
SGTR	43-01	SGTR with failure of secondary heat removal	LBE-26	4.6×10^{-7}	1.9×10^{-6}	366R
SLOCA	01	SLOCA with all systems successful	LBE-27	4.1×10^{-4}	1.9×10^{-7}	<1mR
SLOCA	02	SLOCA with the failure of RHR and successful high pressure recirculation	LBE-28	1.6×10^{-6}	7.9×10^{-6}	<1mR
SLOCA	04	SLOCA with failure of cooldown and high pressure recirculation	LBE-28	3.9×10^{-7}	2.0×10^{-6}	<1mR
SLOCA	03-01-01	SLOCA with the failure of RHR and high pressure recirculation	LBE-29	1.9×10^{-7}	8.7×10^{-7}	1.2R
TRANS	01	TRANS with all system successful	LBE-30	7.0×10^{-1}	1.3	<1mR
TRANS	02	TRANS with failure PORVs to reseat	LBE-27	4.3×10^{-7}	1.4×10^{-6}	<1mR
TRANS	18-01-01-01	TRANS with failure of secondary heat removal	LBE-31	4.7×10^{-7}	2.0×10^{-6}	1.2R
TRANS	18-01-03-01	TRANS with failure of secondary heat removal	LBE-31	4.1×10^{-7}	1.7×10^{-6}	1.2R
TRANS	18-01-04-01	TRANS with failure of secondary heat removal and induced SGTR	LBE-29	6.9×10^{-8}	2.9×10^{-7}	356R
TRANS	18-01-06-01	TRANS with failure of secondary heat removal	LBE-31	4.8×10^{-7}	2.0×10^{-6}	1.2R
TRANS	18-01-07-01	TRANS with failure of secondary heat removal	LBE-31	4.4×10^{-7}	1.8×10^{-6}	1.2R
TRANS	18-01-08-01	TRANS with failure of secondary heat removal and induced SGTR	LBE-29	3.9×10^{-8}	1.6×10^{-7}	356R
TRANS	18-02-01-01	TRANS with failure of secondary heat removal	LBE-31	4.4×10^{-7}	1.8×10^{-6}	1.2R

Table E-8 PRA sequences grouping

Initiating Event	Sequence	Description	Event Class	Mean (per year)	95th (per year)	95th Dose (REM)
TRANS	18-02-02-01	TRANS with failure of secondary heat removal and induced SGTR	LBE-29	2.2×10^{-7}	9.3×10^{-7}	356R
TRANS	18-02-03-01	TRANS with failure of secondary heat removal	LBE-31	4.7×10^{-7}	$2.- \times 10^{-6}$	1.2R
TRANS	18-02-04-01	TRANS with failure of secondary heat removal and induced SGTR	LBE-29	1.8×10^{-7}	7.5×10^{-7}	356R
TRANS	19-08	ATWS with all systems successful (MFW not credited)	LBE-18	1.4×10^{-6}	4.8×10^{-6}	<1mR
TRANS	19-09	ATWS with stuck open PORVs	LBE-18	4.2×10^{-7}	1.9×10^{-6}	<1mR
TRANS	19-14-01-01	ATWS with failure to emergency borate	LBE-32	2.9×10^{-8}	1.3×10^{-7}	0.5R
TRANS	19-16-01-01-01	ATWS with RCS pressure boundary failure	LBE-32	3.4×10^{-8}	1.4×10^{-7}	0.5R
V-RHR-SUC	03	RHR Suction ISLOCA with successful mitigation	LBE-33	4.0×10^{-6}	8.8×10^{-6}	<1mR
V-RHR-SUC	04-01	RHR Suction ISLOCA with failure to isolate	LBE-34	9.7×10^{-8}	1.4×10^{-8}	3548R
V-RHR-SUC	05-01	RHR Suction ISLOCA with failure to diagnose	LBE-34	9.9×10^{-7}	1.6×10^{-6}	3548R

Discussion of the grouping process can be found in Step 5 and Step 6.

Step 5 Group the PRA event sequences with a 95th percentile frequency $\geq 1 \times 10^{-7}$ per year into event classes.

An event class is a group of sequences that displays similar accident behavior or phenomena. As stated in Chapter 6, the goal of the grouping process is to account for all the event sequences with a 95th percentile frequency equal to or greater than 1×10^{-7} per year and to strike a reasonable balance between the number of event classes and the degree of conservatism used in the grouping process. As a result of the grouping process, all sequences equal to or greater than 1×10^{-7} per year are covered by an LBE. Sequences resulting in small doses can be covered with a few 'high' frequency LBEs, representing general event classes, that still satisfy the F-C curve and the associated frequency-range related criteria of Table 6-3 of the main report. Higher dose sequences can be covered with more numerous LBEs representing more detailed event classes, to show that they satisfy the F-C curve and associated criteria. Table E-8 shows the assignment of the PRA sequences to event classes.

Step 6 Select an event sequence from the event class that represents the bounding consequence.

The selected event sequence defines the accident behavior and consequences for the LBE that represent this event class. If several events within the event class have similar consequences, then a bounding event is selected. If there is not a clear bounding event, then the event with the lowest frequency is selected. Note that the frequency of the event class is determined separately from the bounding consequence event. See Step 7. Table E-9 lists the resulting bounding events for the example PWR.

Table E-9 Licensing Basis Events

LBE	Description	Frequency Bases	Consequence Bases	Mean (per year)	95th (per year)	Category	95th Dose
LBE-01	Loss of a DC Bus with all remaining systems successful	LDCA 01	LDCA 01 (1 Event)	$2.5x10^{-3}$	$1.0x10^{-2}$	Frequent	<1mR
LBE-02	Loss of DC with no secondary heat removal, early secondary depressurization and no induced SGTR	LDCA 10-01-06-01	LDCA 10-01-03-01 (6 Events)	$3.9x10^{-8}$	$1.7x10^{-7}$	Rare	1.2R
LBE-03	LLOCA with all systems successful	LLOCA 01	LLOCA 01 (1 Event)	$5.1x10^{-6}$	$1.9x10^{-5}$	Infrequent	<1mR
LBE-04	Loss of Essential Reactor Cooling Water with RCPs intact	LOESW 01	LOESW 01 (2 Events)	$4.1x10^{-4}$	$1.9x10^{-3}$	Infrequent	<1mR
LBE-05	Loss of Essential Reactor Cooling Water with RCP seal failure	LOESW 02	LOESW 02 (2 Events)	$8.1x10^{-5}$	$4.1x10^{-4}$	Infrequent	<1mR
LBE-06	Loss of Essential Reactor Cooling Water with failure to cooldown	LOESW 09	LOESW 09 (2 Events)	$3.9x10^{-7}$	$2.0x10^{-6}$	Rare	<1mR
LBE-07	Loss of Essential Reactor Cooling Water with RCP seal failure. Essential Reactor Cooling is recovered but low pressure recirculation fails	LOESW 04-01-01	LOESW 03-01-01 (2 Events)	$2.5x10^{-5}$	$1.2x10^{-4}$	Infrequent	7R
LBE-08	Loss of Essential Reactor Cooling Water with recovery and RCP seal failure	LOESW 10	LOESW 10 (1 Event)	$7.8x10^{-8}$	$3.3x10^{-7}$	Rare	<1mR
LBE-09	LOOP with all systems successful, 2 hr recovery no inventory challenge	LOOP 01	LOOP 01 (1 Event)	$3.3x10^{-2}$	$8.5x10^{-2}$	Frequent	<1mR

Table E-9 Licensing Basis Events

LBE	Description	Frequency Bases	Consequence Bases	Mean (per year)	95th (per year)	Category	95th Dose
LBE-10	LOOP with RCP seal failure (Bounding LOOP: Stage 2 seal failure and Loop recovery fails)	LOOP 02-01	LOOP 02-03 (3 Events)	2.4×10^{-6}	9.4×10^{-6}	Rare	<1mR
LBE-11	LOOP, 2 hr recovery fails, PORVs fail to close, high pressure recirculation successful	LOOP 03	LOOP 10 (2 Events)	1.7×10^{-7}	6.0×10^{-7}	Rare	<1mR
LBE-12	LOOP with AFW failure (Bounding LOOP: RCP seals intact, early SG depressurization)	LOOP 17-01-06-01	LOOP 17-03-03-01 (3 Events)	2.7×10^{-8}	1.1×10^{-7}	Rare	927R
LBE-13	SBO with secondary heat removal, power recovery and RCP seal integrity maintained	LOOP 18-01	LOOP 18-01 (2 Events)	1.4×10^{-5}	5.5×10^{-5}	Infrequent	<1mR
LBE-14	SBO with battery depletion (Bounding LOOP: no RCS depressurization, vessel breach)	LOOP 18-03-06-01	LOOP 18-03-10-01 (4 Events)	9.7×10^{-7}	3.8×10^{-6}	Rare	1060R
LBE-15	SBO with secondary heat removal, RCP seal failure and power recovery	LOOP 18-04-01	LOOP 18-04-01 (4 Events)	2.2×10^{-6}	1.0×10^{-5}	Infrequent	<1mR
LBE-16	SBO with secondary heat removal, RCP seal failure and no power recovery (Bounding: no RCS depressurization, RCP Stage 2 failure)	LOOP 18-06-11-01	LOOP 18-06-11-01 (3 Events)	1.8×10^{-7}	7.0×10^{-7}	Rare	1060R
LBE-17	SBO with failure of secondary heat removal, RCP seal failure and power recovery within 1 hour	LOOP 18-44	LOOP 18-44 (1 Event)	1.7×10^{-7}	6.5×10^{-7}	Rare	<1mR
LBE-18	ATWS with all systems successful (MFW not credited)	TRANS 19-08	TRANS 19-08 (3 Events)	1.4×10^{-6}	4.8×10^{-6}	Rare	<1mR
LBE-19	MLOCA with all systems successful	MLOCA 01	MLOCA 01 (1 Event)	4.1×10^{-5}	1.5×10^{-4}	Infrequent	<1mR
LBE-20	MLOCA with high pressure recirculation failure	MLOCA 02-01-01	MLOCA 02-01-01 (1 Event)	$1,1 \times 10^{-7}$	4.3×10^{-7}	Rare	1.2R
LBE-21	SGTR with all systems successful	SGTR 01	SGTR 01 (1 Event)	4.0×10^{-3}	1.6×10^{-2}	Frequent	<1mR

Table E-9 Licensing Basis Events

LBE	Description	Frequency Bases	Consequence Bases	Mean (per year)	95th (per year)	Category	95th Dose
LBE-22	SGTR with failure to isolate the ruptured SG	SGTR 02	SGTR 02 (1 Event)	5.0×10^{-5}	2.4×10^{-4}	Infrequent	<1mR
LBE-23	SGTR with failure to isolate and failure of RHR	SGTR 03-02-01	SGTR 03-02-01 (3 Events)	1.2×10^{-7}	5.4×10^{-7}	Rare	88R
LBE-24	SGTR with failure to depressurize before SG reliefs lift, failure to isolate the ruptured SG and failure of RHR	SGTR 11-02-01	SGTR 08-01-01 (5 Events)	4.7×10^{-7}	2.2×10^{-6}	Rare	88R
LBE-25	SGTR with failure to depressurize before SG reliefs lift	SGTR 06	SGTR 06 (2 Events)	4.4×10^{-5}	2.1×10^{-4}	Infrequent	<1mR
LBE-26	SGTR with failure of secondary heat removal	SGTR 43-01	SGTR 43-01 (1 Event)	4.6×10^{-7}	1.9×10^{-6}	Rare	366R
LBE-27	SLOCA with all systems successful	SLOCA 01	SLOCA 01 (2 Events)	4.1×10^{-4}	1.9×10^{-3}	Infrequent	<1mR
LBE-28	SLOCA with the failure of RHR and successful high pressure recirculation (Bounding event: failure of HP recirculation)	SLOCA 02	SLOCA 03-01-01 (2 Events)	1.6×10^{-6}	7.9×10^{-6}	Rare	<1mR
LBE-29	Transient with failure of secondary heat removal and induced SGTR	TRANS 18-02-02-01	TRANS 18-02-02-01 (5 Events)	2.2×10^{-7}	9.3×10^{-7}	Rare	356R
LBE-30	Transient with all systems successful	TRANS 01	TRANS 01 (1 Event)	6.7×10^{-1}	1.2	Frequent	<1mR
LBE-31	Transient with failure of secondary heat removal (Bounding: SG depressurization with induced SGTR)	TRANS 18-01-06-01	TRANS 18-01-03-01 (6 Events)	4.8×10^{-7}	2.0×10^{-6}	Rare	1.2R
LBE-32	ATWS with RCS pressure boundary failure (Bounding: ATWS with failure to emergency borate)	TRANS 19-16-01-01-01	TRANS 19-14-01-01 (2 Events)	3.4×10^{-8}	1.4×10^{-7}	Rare	0.5R
LBE-33	RHR Suction ISLOCA with successful mitigation	V-RHR-SUC 03	Y-RHR-SUC 03 (1 Event)	3.8×10^{-6}	8.8×10^{-6}	Rare	<1mR
LBE-34	RHR Suction ISLOCA with failure to diagnose	V-RHR-SUC 05-01	V-RHR-SUC 05-01 (2 Events)	9.9×10^{-7}	1.5×10^{-6}	Rare	3548R

As can be seen from Table E-9, 34 LBEs have been identified with each representing between one and six event sequences. Eleven LBEs address only a single event sequence. For the remaining 23 sequences, a bounding event was selected to represent the event class.

A discussion on LBE-02 is provided in order to illustrate the selection and grouping process. LBE-02 represents 6 events sequences, each initiated by the loss of a DC bus followed by the failure of auxiliary feedwater. Although feed and bleed is available at the example plant, this function was set to guaranteed failure, as it is not safety-related. For all six events, containment isolation remains intact and an induced steam generator tube rupture is avoided. The six events are differentiated by the status of RCS and secondary system pressure. For the four sequence 10-01 events, the steam generators are initially maintained at normal pressure. For the two sequence 10-02 events, early secondary system depressurization occurs. The additional variations of these sequences is associated with late depressurization of the RCS and secondary systems. The variations are shown in Table E-10.

Table E-10 LBE-02 bounding event selection

Sequence	Early Secondary System Depressurization	Late RCS Depressurization	Late Secondary Systems Depressurization
LDCA 10-01-01-01	No	No	No
LDCA 10-01-03-01	No	No	Yes
LDCA 10-01-06-01	No	Yes	No
LDCA 10-01-07-01	No	Yes	Yes
LDCA 10-02-01-01	Yes	No	No
LDCA 10-02-03-01	Yes	No	Yes

The bounding event sequence, LDC 10-02-03-01, was selected to represent event class LBE-02 because it results in the highest pressure differential across the steam generator tubes for the longest period of time. Although none of these sequences result in a steam generator tube rupture, the bounding event creates the most severe challenge to this condition.

It should also be noted that event grouping does not have to be limited to sequences with the same initiating event. LBE-18 is an example of an event class that crosses between initiating events. LBE-18 represents three anticipated transient without scram (ATWS) events. One of the events is initiated as a result of a loss of offsite power event with the resulting failure of the control rods to insert into the reactor core. The other two sequences are initiated by a transient. These events are shown in Table E-11.

Table E-11 LBE-18 bounding event selection

Initiating Event	Sequence	Description	Dose
LOOP	19-08	ATWS with all systems successful (MFW not credited)	<1mR
TRANS	19-08	ATWS with all systems successful (MFW not credited)	<1mR
TRANS	19-09	ATWS with stuck open PORV (MFW not credited)	<1mR

TRANS 19-09 was selected as the bounding event because, similar to the other events, the ATWS event is mitigated. However, this event has the additional challenge of the stuck open PORV.

Step 7 Establish the LBE's frequency for a given event class.

The frequency of an event class is determined by setting the LBE's mean frequency to the highest mean frequency of the event sequences in the event class and its 95^{th} percentile frequency to the highest 95^{th} percentile frequency of the event sequences in the event class. Note that the mean and 95^{th} percentile frequencies can come from different event sequences. The example results are shown in Table E-9. In the example, the mean and 95^{th} percentile frequency for each LBE come from the same event sequence.

Step 8 Bin each LBE into one of three frequencies ranges: Frequent, Infrequent or Rare.

The potential defense-in-depth requirements are a function of the frequency ranges. This binning is required in order to determine the LBE deterministic requirements. These frequency ranges and their associated potential requirements are shown in Table E-2. Table E-9 shows the results of this binning process.

Step 9 Determine the total weighted annual frequencies for all events equal to or greater than $1x10^{-2}$ and $1x10^{-3}$.

The frequency-consequence limits for events greater than $1x10^{-2}$ and $1x10^{-3}$ are based on annual dose as opposed to the event dose limits that are associated with the other regions of the frequency-consequence curve. Therefore, to determine the expected annual dose, the weighted dose (frequency x dose) of the events equal to or greater than 1E-2 are summed and evaluated against the frequency-consequence curve limit of 5mrem as shown in Table E-12. In a similar fashion, the weighted dose for those events equal to or greater than $1x10^{-3}$ are summed and evaluated against the frequency-consequence limit of 100mrem as shown in Table E-13.

Table E-12 Licensing Basis Events equal or greater than 1E-2 per year

LBE	Sequence	Mean Frequency	95^{th} Frequency	Event Dose (mrem)	Weighted Mean Dose (mrem)	Weighted 95^{th} Dose (mrem)
1	LDCA-01	$2.5x10^{-3}$	$1.0x10^{-2}$	<1	$<2.5x10^{-3}$	$<1.0x10^{-2}$
9	LOOP-01	$3.3x10^{-2}$	$8.5x10^{-2}$	<1	$<3.3x10^{-2}$	$<8.5x10^{-2}$
21	SGTR-01	$4.0x10^{-3}$	$1.6x10^{-2}$	<1	$<4.0x10^{-3}$	$<1.6x10^{-2}$
30	TRANS-01	$7.0x10^{-1}$	1.3	<1	$<7.0x10^{-1}$	<1.3
TOTAL					**$<7.4x10^{-1}$**	**<1.4**

Table E-13 Licensing Basis Events equal or greater than $1x10^{-3}$ per year

LBE	Sequence	Mean Frequency	95^{th} Frequency	Event Dose (mrem)	Weighted Mean Dose (mrem)	Weighted 95^{th} Dose (mrem)
1	LDCA-01	$2.5x10^{-3}$	$1.0x10^{-2}$	<1	$<2.5x10^{-3}$	$<1.0x10^{-2}$
4	LOCCW-A	$2.0x10^{-4}$	$9.6x10^{-4}$	<1	$<2.0x10^{-4}$	$<9.6x10^{-4}$
4	LOESW-01	$4.1x10^{-4}$	$1.9x10^{-3}$	<1	$<4.1x10^{-4}$	$<1.92x10^{-3}$
9	LOOP-01	$3.3x10^{-2}$	$8.5x10^{-2}$	<1	$<3.3x10^{-2}$	$<8.5x10^{-2}$
21	SGTR-01	$4.0x10^{-3}$	$1.6x10^{-2}$	<1	$<4.0x10^{-3}$	$<1.6x10^{-2}$
27	SLOCA-01	$4.1x10^{-4}$	$1.9x10^{-7}$	<1	$<4.1x10^{-4}$	$<1.9x10^{-7}$
27	TRANS-02	$4.3x10^{-7}$	$1.4x10^{-6}$	<1	$4.3x10^{-7}$	$<1.4x10^{-6}$
30	TRANS-01	$7.0x10^{-1}$	1.3	<1	$<7.0x10^{-1}$	<1.3
TOTAL					**$<7.4x10^{-1}$**	**<1.4**

Step 10 Verify that the selected LBEs meet the probabilistic and deterministic probabilistic acceptance criteria.

Figure E-1 shows the 95^{th} percentile dose of the identified LBEs on the F-C curve. The PWR example shows six LBEs exceeding the F-C curve. Figure E-2 shows the mean dose values with four LBEs exceeding the F-C curve.

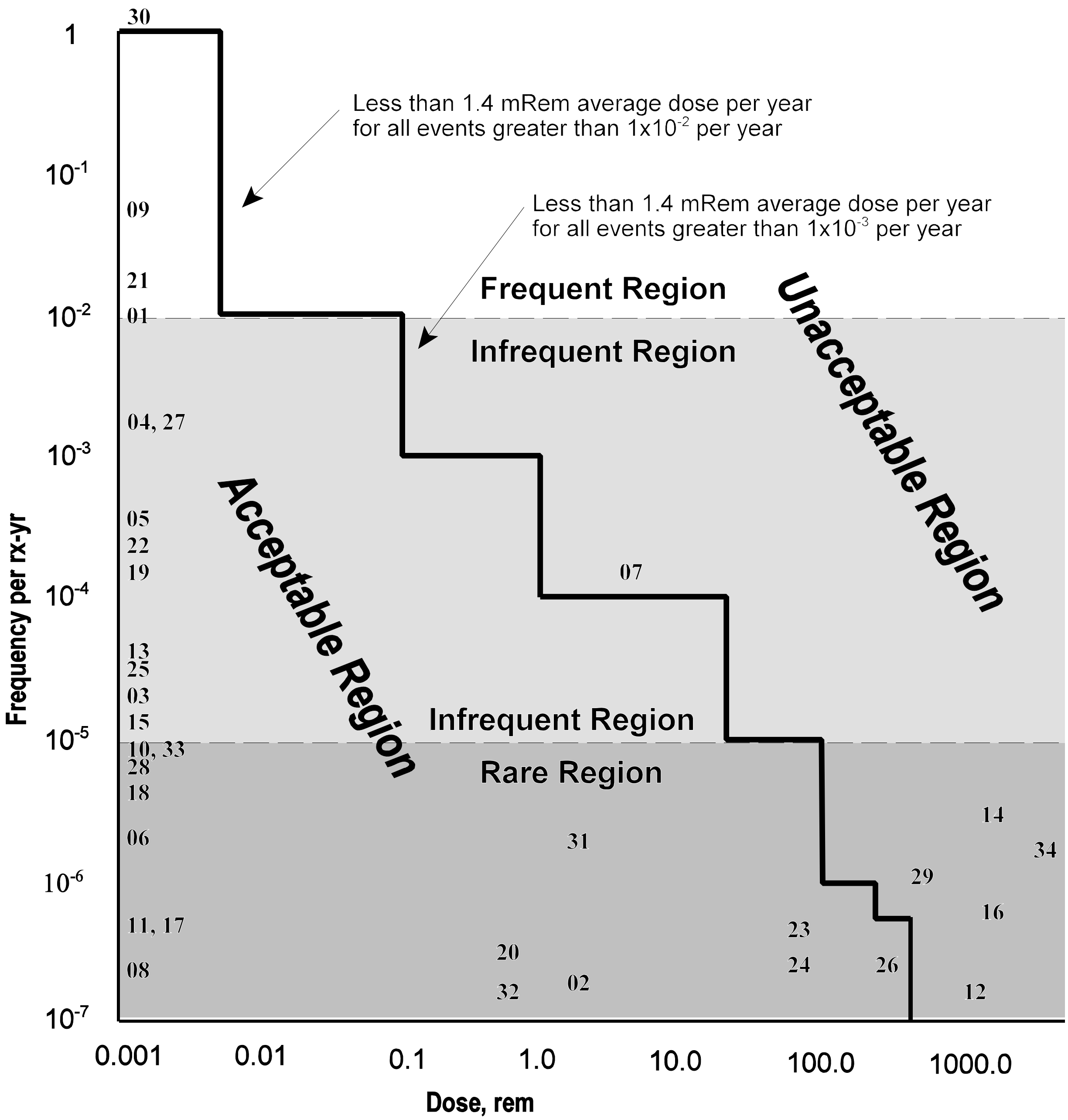

Figure E-1 Frequency-consequence curve with 95th percentile values

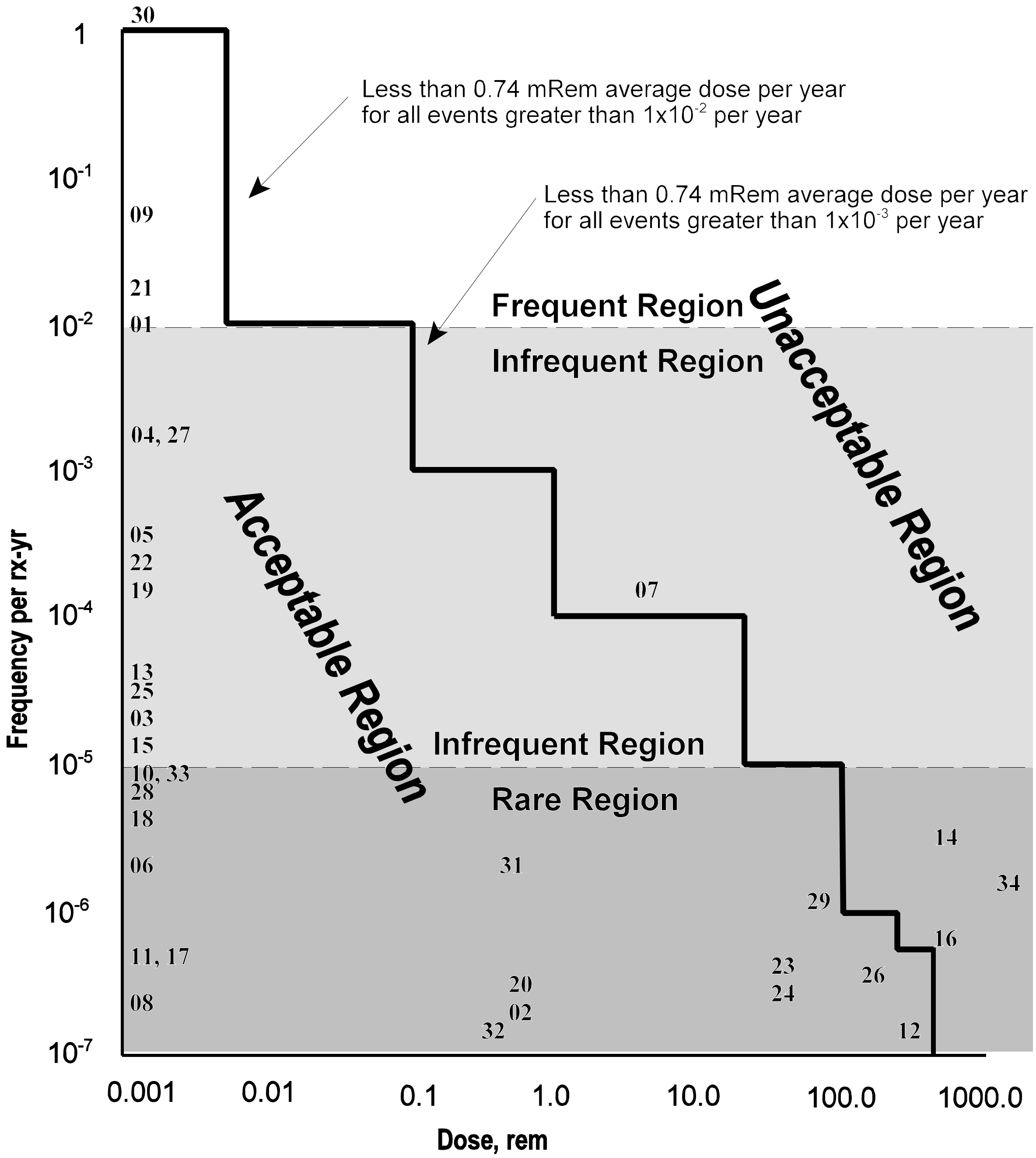

Figure E-2 Frequency-consequence curve with mean values

The Framework has additional deterministic potential requirements for LBEs classified as Frequent or Infrequent. The example in this appendix has four Frequent Events, ten Infrequent LBEs and twenty Rare LBEs. Tables E-14 and E-15 show the deterministic requirements for Frequent and Infrequent LBEs, respectively, and show how the example's LBEs compare with the deterministic requirement.

Table E-14 Deterministic requirements for LBEs categorized as frequent

LBE	Description	No Barrier Failure	No Impact on Safety Assumptions	Redundant Functions (1)	Dose <100mR	Comments
LBE-01	Loss of a DC Bus with all remaining systems successful	MEETS	MEETS	MEETS	MEETS	
LBE-09	LOOP with all systems successful, 2 hr recovery no inventory challenge	MEETS	MEETS	MEETS	MEETS	
LBE-21	SGTR with all systems successful	DOES NOT MEET	MEETS	MEETS	MEETS	The SGTR initiating event fails the RCS and containment boundaries
LBE-30	Transient with all systems successful	MEETS	MEETS	MEETS	MEETS	

(1) This column addresses the acceptance criteria for redundant means of reactor shutdown and decay heat removal.

Table E-15 Deterministic requirements for LBEs categorized as infrequent

LBE	Description	At Least One Barrier Remains	Coolable Geometry Remains	One Means of Reactor S/D & DHR (1)	Dose Meets F-C Curve	Comments
LBE-03	LLOCA with all systems successful	MEETS	MEETS	MEETS	MEETS	
LBE-04	Loss of Essential Reactor Cooling Water with RCPs intact	MEETS	MEETS	MEETS	MEETS	

Table E-15 Deterministic requirements for LBEs categorized as infrequent

LBE	Description	At Least One Barrier Remains	Coolable Geometry Remains	One Means of Reactor S/D & DHR (1)	Dose Meets F-C Curve	Comments
LBE-05	Loss of Essential Reactor Cooling Water with RCP seal failure	MEETS	MEETS	MEETS	MEETS	
LBE-07	Loss of Essential Reactor Cooling Water with RCP seal failure. Essential Reactor Cooling is recovered but low pressure recirculation fails.	MEETS	DOES NOT MEET	DOES NOT MEET	DOES NOT MEET	This event sequence results in core damage and exceeds the F-C curve. The RCS barrier is breached due to RCP seal failure and fuel cladding barrier fails due to failure of low pressure recirculation. Long-term decay heat removal is not achieved due to failure of low pressure recirculation. Containment isolation is achieved and maintained.
LBE-13	SBO with secondary heat removal, power recovery and RCP seal integrity maintained	MEETS	MEETS	MEETS	MEETS	Reactivity control and Auxiliary Feedwater function. Power is recovered prior to battery depletion.
LBE-15	SBO with secondary heat removal, RCP seal failure and power recovery	MEETS	MEETS	MEETS	MEETS	Reactivity control and Auxiliary Feedwater function. Power is recovered prior to battery depletion.
LBE-19	MLOCA with all systems successful	MEETS	MEETS	MEETS	MEETS	
LBE-22	SGTR with failure to isolate the ruptured SG	MEETS	MEETS	MEETS	MEETS	

Table E-15 Deterministic requirements for LBEs categorized as infrequent

LBE	Description	At Least One Barrier Remains	Coolable Geometry Remains	One Means of Reactor S/D & DHR (1)	Dose Meets F-C Curve	Comments
LBE-25	SGTR with failure to depressurize before SG reliefs lift	MEETS	MEETS	MEETS	MEETS	
LBE-27	SLOCA with all systems successful	MEETS	MEETS	MEETS	MEETS	

(1) The column addresses the acceptance criteria for at least one means of reactor shutdown and one means of decay heat removal to remain functional.

E.5 Comparison with Current Design Bases Events

E.5.1 Design Bases Events for Example Plant

This section describes the conditions or design basis events (DBEs) analyzed in the example plant's FSAR Chapter 15 analysis. The development of these original DBEs is consistent with Regulatory 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." The following five conditions, shown in Table E-16, were analyzed in the example plant's FSAR:

Table E-16 DBE condition categories

Condition	Title	Description
1	Normal operation and operational transients	These faults, at worst, result in the reactor shutdown with the plant being capable of returning to operation. By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., Condition III or IV. In addition, Condition II events are not expected to result in fuel rod failures or Reactor Coolant System over pressurization.
2	Faults of moderate frequency	Faults which may occur very infrequently during the life of the plant. They will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of these areas beyond the exclusion radius.
3	Infrequent faults	Faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These are the most drastic which must be designed against and thus, represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR Part 100.

Table E-16 DBE condition categories

Condition	Title	Description
4	Limiting faults	Faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These are the most drastic which must be designed against and thus, represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR Part 100. A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the Emergency Core Cooling System (ECCS) and the containment.
E	Environmental Faults	Faults that provide the limiting events for environmental consequences of an event.

Table E-17 lists the Condition II, III, IV and E events. Condition I events are normal operation and operational transients (e.g., power operation, start up, hot shutdown, cold shutdown, refueling). As stated in the example plant's FSAR, Condition I occurrences occur frequently or regularly, and they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III, and IV). In this regard, analysis of each fault condition described in Table E-17 is generally based on a conservative set of initial conditions corresponding to the most adverse set of conditions which can occur during Condition I operation. An explicit evaluation of each Condition I event is not provided in the FSAR.

Table E-17 Example PWR Chapter 15 events

Event	Title	Description	Cat
1.1	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition	A rod cluster control assembly withdrawal of rod cluster control assemblies resulting in a power excursion. Such a transient could be caused by a malfunction of the reactor control or control rod drive systems. This is the maximum rate of reactivity addition (greater than the boron dilution event).	II
1.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power	Same as D.1.1, except at-power.	II
1.3	Rod Cluster Control Assembly Misalignment	Rod cluster control assembly misalignment includes: a dropped full-length assembly, a dropped full-length assembly bank, and statically misaligned full length assembly.	II
1.4	Uncontrolled Boron Dilution	The Chemical and Volume Control System (CVCS) is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.	II

Table E-17 Example PWR Chapter 15 events

Event	Title	Description	Cat
1.5	Partial Loss of Forced Reactor Coolant Loop	A partial loss of coolant flow accident can result from a mechanical or electrical failure in a reactor coolant pump, or from a fault in the power supply to the pump. If the reactor is at-power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. The necessary protection against a partial loss of coolant flow is provided by the low primary coolant flow reactor trip, which is actuated by two out of three low flow signals in any reactor coolant loop.	II
1.6	Startup of an Inactive Reactor Coolant Loop	Starting of an idle reactor coolant pump without bringing the inactive loop hot leg temperature close to the core inlet temperature would result in the injection of cold water into the core, which causes a rapid reactivity insertion and subsequent power increase.	II
1.7	Loss of External Electrical Load and/or Turbine Trip	Major load loss on the plant can result from loss of external electrical load or from a turbine trip. For either case, off-site power remains available for the continued operation of plant components, such as reactor coolant pumps. The case of loss of all AC power (station blackout) is analyzed in Section D.1.9. Following the loss of generator load, an immediate fast closure of the turbine control valves will occur. For a turbine trip, the reactor would be tripped directly (unless below approximately 50% power) from a signal derived from the turbine autostop oil pressure and turbine stop valves.	II
1.8	Loss of Normal Feedwater	Event assumes that the reactor trips on low-low level in any steam generator and that only one motor driven auxiliary feedwater pump is available one minute after the low-low steam generator level signal is initiated. Secondary system steam relief is achieved through the self-actuated safety valves.	II
1.9	Loss of All Off-Site Power to the Station Auxiliaries	Event assumes that only one motor-driven auxiliary feedwater pump is available one minute after the low-low steam generator level signal is initiated in any steam generator.	II
1.10	Excessive Heat Removal Due to Feedwater System Malfunctions	Excessive feedwater flow could be caused by a full opening of one or more feedwater regulator valves due to a feedwater control system malfunction or an operator error. The feedwater flow from a fully open regulator valve is terminated by the steam generator high-high signal, which closes all feedwater regulator valves and feedwater isolation valves and trips the main feedwater pumps.	II
1.11	Excessive Load Increase Incident	This accident could result from either an administrative violation, such as excessive loading by the operator, or an equipment malfunction in the steam dump control or turbine speed control.	II
1.12	Accidental Depressurization of the Reactor Coolant System (inadvertent opening of pressurizer spray valve)	The most severe core condition resulting from an accidental depressurization of the RCS is associated with an inadvertent opening of a pressurizer safety valve. The reactor will be tripped by one of the following RPS signals: 1) pressurizer low pressure, or 2) overtemperature ΔT.	II

Table E-17 Example PWR Chapter 15 events

Event	Title	Description	Cat
1.13	Accidental Depressurization of Main Steam System (inadvertent opening of a single dump, relief or safety valve)	The most severe core condition resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening of a single steam dump, relief or safety valve. The following systems provide the necessary protection against an accidental depressurization of the main steam system: 1) safety injection system actuation, 2) the overpower reactor trip, and 3) redundant isolation of the main feedwater lines.	II
1.14	Spurious Operation of the Safety Injection System At Power	Following the actuation signal, the suction of the centrifugal charging pump is diverted from the volume control tank to the refueling water storage tank. The valves isolating the injection tank from the charging pumps and the injection header then automatically open. The charging pumps then provide RWST water through the header and injection line and into the cold legs of each loop. The safety injection pumps also start automatically but provide no flow when the RCS is at normal pressure.	II
2.1	Loss of Coolant for Small Rupture Pipes or from Cracks in Large Pipes which Actuate the Emergency Core Cooling System	The analysis shows that the small break LOCA is not limiting with respect to large break LOCA results. The predicted peck cladding temperature is less than 1163F for the pump discharge break, the local and whole-core metal-water reaction percentages are negligible, the hot pin thermal transient is insufficient to cause significant fuel pin deformation and the core remains amenable to cooling.	III
2.2	Minor Secondary System Pipe Breaks	Minor secondary system pipe breaks must be accommodated with the failure of only a small fraction of the fuel elements in the reactor. Since the results of analysis for a major secondary system pipe rupture also meet this criteria, separate analysis form minor secondary system pipe breaks is not required.	III
2.3	Inadvertent Loading of a Fuel Assembly into an Improper Position	Fuel assembly enrichment errors would be prevented by administrative procedures implemented in fabrication. In the event that a single pin or pellet has a higher enrichment than the nominal value, the consequences in terms of reduced DNBR and increased fuel and clad temperatures will be limited to the incorrectly loaded pin or pins.	III
2.4	Complete Loss of Forced Reactor Coolant Flow	The analysis demonstrates that for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the safety analysis limit during the transient and thus, there is no clad damage or release of fission products to the Reactor Coolant System.	III
2.5	Waste Gas Decay Tank Rupture	Refer to Table Entry 4.2.	III
2.6	Single Rod Cluster Control Assembly Withdrawal, At Full Power	For the case of one rod cluster control assembly fully withdrawn, with the reactor in the automatic or the manual control mode and initially operation at full power with Bank D at the insertion limit, an upper bound of the number of fuel rods experiencing a DNBR of less than 1.3 is 5 percent of the total fuel rods in the core.	III
2.7	Steam Line Break Coincident with Rod Withdrawal at Power (SLB c/w RWAP)	Addresses potential unreviewed safety question identified in IE-79-22 entitled “Qualification of Control Systems.” One of the postulated scenarios that was identified was the operation of the non-safety grade automatic rod control system following a steam line break inside or outside of containment.	III

Table E-17 Example PWR Chapter 15 events

Event	Title	Description	Cat
3.1	Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)	Containment Design: (Section 3.8.2.2.2) The containment is designed so that the leakage from the largest credible energy release following a LOCA (DBA), including the calculated energy form metal-water or other chemical reactions that could occur as a consequence of failure of any single active component in the Emergency Cooling system will not result in undue risk to the health and safety of the public, and is designed to limit to below 10 CFR 100 values, the leakage of radioactive products from the containment under such (DBA) conditions. See 15.5.3 for siting criteria.	IV
3.2	Major Secondary System Pipe Rupture	Main Steam Line Break: One S/G blows down (one MSIV fails or break is upsteam of MSIV), one safety injection pump available, MFW isolation occurs, AFW flow is maximized. Main Feedwater Line Break: MFW assumed stopped at time of break, AFW turbine-driven pump assumed failed, AFW motor-driven pump supplies two of four S/Gs.	IV
3.3	Steam Generator Tube Rupture	Analysis assumes that the operator identifies the accident type and terminates break flow to the faulty steam generator within 30 minutes of accident initiation. Included in this 30 minute time period would be an allowance of 5 minutes to trip the reactor and actuate the safety injection system, 10 minutes to identify the accident as a steam generator tube rupture and 15 minutes to isolate the faulty steam generator. The operator is then assumed to initiate RCS cooldown by dumping steam from intact steam generators to condenser. This action is required to establish adequate subcooling to permit reducing RCS pressure. Cases with and without off-site power were evaluated.	IV
3.4	Single Reactor Coolant Pump Locked Rotor	After pump seizure, reactor coolant system flow is reduced and the system heats up and pressurizes. A reactor trip occurs as a consequence of low flow. The neutron flux is rapidly reduced by control rod insertion. Loss of off-site power is assumed to occur simultaneously with the reactor trip.	IV
3.5	Fuel Handling Event	The accident is defined as dropping of a spent fuel assembly onto the spent fuel pit floor resulting in the rupture of the cladding of all the fuel rods in the assembly. See 15.5.6.	IV
3.6	Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)	This accident is defined as the mechanical failure of a controlled mechanism pressure housing resulting in the ejection of a rod cluster control assembly and drive shaft. The consequence of this mechanical failure is a rapid reactivity insertion together with an adverse core power distribution possibly leading to localized fuel rod damage.	IV
4.1	Environmental Consequences of a Postulated Loss of AC Power to the Plant Auxiliaries	The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage form the Reactor Coolant systems to the secondary system in the steam generators. This analysis incorporates assumptions of one percent defective fuel and steam generator leakage prior to the postulated accident for a time sufficient to establish equilibrium specific activity levels in the secondary system.	E

Table E-17 Example PWR Chapter 15 events

Event	Title	Description	Cat
4.2	Environmental Consequences of a Postulated Waste Gas Decay Tank Rupture	RG 1.24 Analysis.	E
4.3	Environmental Consequences of a Postulated Loss of Coolant Accident	RG 1.4 Analysis: For the analysis of this hypothetical case, it is assumed that of the entire core-fission product inventory, 100 percent of the noble gases, 50 percent of the halogens, and 1percent of the solids in the fission product inventory are released to the containment. Of the fission product iodine released to the containment, 50 percent is considered to be available for leakage, while the remaining 50 percent is assumed to condense on the various structural surfaces in the containment. Thus, a total of 100 percent of the noble gas core inventory and 25 percent of the core iodine inventory are assumed to be immediately available for leakage for the primary containment. Of the halogen activity available for release, it is further assumed that 91 percent is in elemental form, 4 percent in methyl form, and 5 percent in particulate form.	E
4.4	Environmental Consequences of a Postulated Steam Line Break	The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the Reactor Coolant systems to the secondary system in the steam generators. This analysis incorporates assumptions of one percent defective fuel and steam generator leakage prior to the postulated accident for a time sufficient to establish equilibrium specific activity levels in the secondary system.	E
4.5	Environmental Consequences of a Postulated Steam Generator Tube Rupture	The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the Reactor Coolant systems to the secondary system in the steam generators. A conservative analysis of the postulated steam generator tube rupture assumes that loss of offsite power and hence, involves the release of steam from the secondary system. This analysis incorporates assumptions of one percent defective fuel and steam generator leakage prior to the postulated accident for a time sufficient to establish equilibrium specific activity levels in the secondary system.	E
4.6	Environmental Consequences of a Postulated Fuel Handling Accident	RG 1.25 Analysis.	E
4.7	Environmental Consequences of a Postulated Rod Ejection Accident	Bounded by Loss of Coolant Accident.	E

E.5.2 Comparison of DBEs and LBEs

The DBEs frequency categories can be loosely compared with the Framework's categories as shown in Table E-18.

Table E-18 DBE and LBE categories

FSAR Category	FSAR Description	Framework Category
II	moderate frequency	frequent
III	infrequent	infrequent
IV	limiting faults	rare

It should be noted that the DBE category is based on the initiating event frequency, while the Framework category is based on the accident sequence frequency. For the frequent category, this difference is not significant, such that there are only four event sequences in the example that fall into this category and none of these sequences have any system failures beyond that of their initiating event. Therefore, their frequency is the initiating event frequency (an approximation that ignores the impact of the success term contribution). For the other categories, this comparison becomes more difficult, such that initiating events that occur in the Framework's frequent category also appear in the infrequent and rare category.

E.5.2.1 Comparison of Events by Category

Moderate Frequency (Category II)/Frequent Category

In the (moderate) frequency category, the events identified by the two methods are similar. As shown in Table E-19, many of the FSAR events are mapped to the Framework's transient initiating event indicating the need for this event to be bounding for all the initiators that are grouped into the transient initiating event category. One event, DB Event 1.12, appears to best map to the infrequent Framework event of small LOCA (Sequence SLOCA 01). Two Framework events, a steam generator tube rupture (Sequence SGTR 01) and the loss of a DC Bus (LDCA-01) are not included as frequent (the Framework equivalent to "moderate frequency") events in the FSAR.

Table E-19 Moderate frequency (Category II) event comparison

FSAR Event	FSAR Title	FSAR Cat	Framework Event	FR Cat
1.1	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition	II	Not addressed by current at-power scope	NA
1.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power	II	In scope of Transient Initiating Event (Sequence TRANS 01)	Freq

Table E-19 Moderate frequency (Category II) event comparison

FSAR Event	FSAR Title	FSAR Cat	Framework Event	FR Cat
1.3	Rod Cluster Control Assembly Misalignment	II	In scope of Transient Initiating Event (Sequence TRANS 01)	Freq
1.4	Uncontrolled Boron Dilution	II	Not addressed by current at-power scope	NA
1.5	Partial Loss of Forced Reactor Coolant Loop	II	In scope of Transient Initiating Event (Sequence TRANS 01)	Freq
1.6	Startup of an Inactive Reactor Coolant Loop	II	Not addressed by current at-power scope	NA
1.7	Loss of External Electrical Load and/or Turbine Trip	II	In scope of Transient Initiating Event (Sequence TRANS 01)	Freq
1.8	Loss of Normal Feedwater	II	In scope of Transient Initiating Event (Sequence TRANS 01)	Freq
1.9	Loss of All Off-Site Power to the Station Auxiliaries	II	In scope of Loss of Offsite Power Event (Sequence LOOP 01)	Freq
1.10	Excessive Heat Removal Due to Feedwater System Malfunctions	II	In scope of Transient Initiating Event (Sequence TRANS 01)	Freq
1.11	Excessive Load Increase Incident	II	In scope of Transient Initiating Event (Sequence TRANS 01)	Freq
1.12	Accidental Depressurization of the Reactor Coolant System (inadvertent opening of pressurizer spray valve)	II	In scope of small LOCA Event (Sequence SLOCA 01)	Infreq
1.13	Accidental Depressurization of Main Steam System (inadvertent opening of a single dump, relief or safety valve)	II	In scope of Transient Initiating Event (Sequence TRANS 01)	Freq
1.14	Spurious Operation of the Safety Injection System At Power	II	In scope of Transient Initiating Event (Sequence TRANS 01)	Freq

<u>Infrequent Category (Category III)</u>

Table E-20 shows the Category III events. There are significant differences between the approaches in this category. First, the Framework example includes small, medium and large LOCA event sequences. For all three initiating events, no degradation of the mitigating systems is assumed (for these events in this category). Small LOCA with failure of residual heat removal is included in the rare event category. The presence of small LOCA in the Framework's frequent and infrequent categories, and only in Category III of the FSAR's approach, highlights the impact of the binning differences between the approaches. Another difference identified in this category is the lack of a main steam line break event in the Framework example. This is due to the exclusion of steam line break events in the SPAR model likely due to the limited contribution these initiators typically have on overall plant risk. It is expected that a fully developed Framework PRA would have these steam line break initiators. Table E-20 provides a list of Category III events with the related LBE.

Table E-20 Infrequent (Category III) event comparison

FSAR Event	FSAR Title	FSAR Cat	Framework Event	FR Cat
2.1	Loss of Coolant for Small Rupture Pipes or from Cracks in Large Pipes which Actuate the Emergency Core Cooling System	III	In scope of small LOCA Event (Sequence SLOCA 01)	Infreq/ Rare
2.2	Minor Secondary System Pipe Breaks	III	No included in scope of SPAR Model.	NA
2.3	Inadvertent Loading of a Fuel Assembly into an Improper Position	III	Not addressed by current at-power scope	NA
2.4	Complete Loss of Forced Reactor Coolant Flow	III	In scope of Transient Initiating Event (Sequence TRANS 01)	Freq
2.5	Waste Gas Decay Tank Rupture	III	Not addressed by current at-power scope	NA
2.6	Single Rod Cluster Control Assembly Withdrawal, At Full Power	III	In scope of Transient Initiating Event (Sequence TRANS 01)	Freq
2.7	Steam Line Break Coincident with Rod Withdrawal at Power (SLB c/w RWAP)	III	Not included in scope of SPAR Model.	NA

Limiting Fault (Category IV)/Rare

There are six limiting fault DBEs identified in example plant's FSAR as shown in Table E-21. One is shutdown related and not addressed by the current selection of at-power LBEs. Both the large break LOCA and main steam line breaks are identified as limiting fault DBEs with the large break LOCA being identified as the limiting event for containment design and siting. In Framework's selection process, only one large break LOCA scenario was identified. Unlike the DBE which considers a simultaneous LOOP and LOCA with a single failure, the large break LOCA LBE does not consider the occurrence of a LOOP event and has all safety functions available.

The SGTR DBE evaluates the mitigation of the rupture with and without a LOOP event. For the LOOP case, the SGTR DBE assumes that a LOOP results in the loss of condenser vacuum and the release of steam to the atmosphere. The DBE analysis appears to be focused on determining the limiting case for mass transfer from the RCS to the secondary system. The analysis assumes one percent defective fuel and steam generator leakage prior to the postulated accident.

The Framework includes six SGTR LBEs. These vary from a sequence with all mitigating systems available to sequences with the failure of residual heat removal or secondary heat removal. There are no Framework events with both a SGTR and a LOOP.

The RCP locked rotor DBE appears to be the limiting RCS pressure transient event with no credit taken for the pressure reducing effect of pressurizer relief valves, pressurizer spray, steam dump or controlled feedwater flow after the plant trip. A similar event was not identified in the Framework LBE process (unless that transient initiating is constructed to bound this event).

The rupture of a control rod drive mechanism is considered the limiting reactivity insertion event and occurs with an adverse core power distribution possibly leading to localized fuel rod damage. This event is not explicitly identified in the Framework LBE process, although it could be considered a specific type of small break LOCA and depending of the design of this initiating event, included in the scope of the SLOCA initiating event. Note that the environmental consequences (dose) of each of the Category IV DBEs are evaluated separately in an environmental consequence section.

Table E-21 Infrequent (Category IV) event comparison

FSAR Event	FSAR Title	FSAR Cat	Framework Event	FR Cat
3.1	Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)	IV	In scope of Large LOCA Event (Sequence LLOCA 01)	Rare
3.2	Major Secondary System Pipe Rupture	IV	No included in scope of SPAR Model.	Rare
3.3	Steam Generator Tube Rupture	IV	In scope of steam generator tube rupture event (Sequence SGTR 01, SGTR 02, SGTR 03-02-01, SGTR 11-02-01, SGTR 06, SGTR 43-01)	Freq/ Infreq/ Rare
3.4	Single Reactor Coolant Pump Locked Rotor	IV	In scope of Transient Initiating Event (Sequence TRANS 01) Note: Assume transient initiating event is constructed to include this event.	Freq
3.5	Fuel Handling Event	IV	Not addressed by current at-power scope	NA
3.6	Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)	IV	In scope of small LOCA Event (Sequence SLOCA 01) Note: The inclusion of this event with the SLOCA event is dependent on the scope of the SLOCA event within the PRA.	Freq

Environmental Consequences of Accidents

The environmental consequence section of example plant's FSAR addresses one Category II event (2.9) that appears to be the limiting Category II event for off-site consequences. It also addresses two shutdown events. These events are not included in the scope of the discussion due to the analysis limitations. The remaining events address the consequences of at-power limiting faults. Both the main steam line break and the rod cluster assembly ejection DBEs were found to be bounded by the large-break LOCA analysis. The large-break LOCA analysis is a RG 1.4 analysis of a hypothetical case that assumes the entire core-fission product inventory, 100 percent of the noble gases, 50 percent of the halogens, and 1 percent of the solids are released to the containment. This analysis is the bounding analysis for siting.

Table E-22 provides a list of environmental events with the related LBE.

Table E-22 Environmental consequences event comparison

FSAR Event	FSAR Title	FSAR Cat	Framework Event	FR Cat
4.1	Environmental Consequences of a Postulated Loss of A.C. Power to the Plant Auxiliaries	E	LOOP Events (01, 02-03, 10, 17-03-03-01, 18-01, 18-03-06-01, 18-04-01, 18-06-11-01 and 18-44)	Freq/ Infreq/ Rare
4.2	Environmental Consequences of a Postulated Waste Gas Decay Tank Rupture	E	Not addressed by current at-power scope	NA
4.3	Environmental Consequences of a Postulated Loss of Coolant Accident	E	Although Sequence LLOCA 01 is identified by the probabilistic LBE selection process, this event is more closely aligned to the deterministic LBE as described in Chapter 6	NA
4.4	Environmental Consequences of a Postulated Steam Line Break	E	Not included in scope of SPAR Model.	NA
4.5	Environmental Consequences of a Postulated Steam Generator Tube Rupture	E	SGTR Events (01, 02, 03-02-01, 11-02-01, 06 and 43-01)	Freq/ Infreq/ Rare
4.6	Environmental Consequences of a Postulated Fuel Handling Accident	E	Not included in scope of SPAR Model.	NA
4.7	Environmental Consequences of a Postulated Rod Ejection Accident	E	Bounded by Loss of Coolant Accident.	NA

E.6 Conclusion

The Framework selection process establishes a comprehensive set of licensing basis events that account for the frequency and severity of the events. In the example, 34 LBEs were identified including four frequent events, 10 infrequent events and 20 rare events. The process identified events with multiple failures and common cause failures and, in some cases, the events included the total loss of safety functions and containment failure. The selection process resulted in the identification of station blackout events (SBO) and anticipated transients without scram (ATWS) events as LBEs.

The identification process did exclude some rare event combinations, such as the coincident LOOP – LOCAs, LOOP – MSLB and LOOP – SGTRs events. For these DBAs, the coincidence occurrences are often used to maximize the release due to the loss of the secondary plant or as a target of the single failure analysis with an emergency diesel generator being failed and therefore, failing all the supported safety equipment. Based on the identified LBEs in this example, there would not be LBEs that require EDGs to support either a medium or large break LOCA.

When the results of the Framework events are compared against the Framework's acceptance criteria, six LBEs are identified as exceeding the F-C curve when using the 95th percentile for both frequency and consequence, and two events are identified as not meeting the deterministic potential requirements. Considering the exclusion of some rare DBA event combinations and a more restrictive performance criteria for the 6 of 34 LBEs that do not satisfy the potential requirements of the F-C curve (and considering the addition of the Framework's deterministic event as described in Chapter 6), the level of safety achieved by the Framework selection process and associated acceptance criteria appears to be exceed that required for current plants. These results are consistent with the Commission's Policy Statement on "Regulation of Advanced Nuclear Power Plants," which contains the expectation that advanced reactors will provide enhanced margins of safety. The results show that the Framework will enhance margins in that some events that are currently acceptable will not be acceptable in future reactors. The Framework process also results in a reduced emphasis on some rare event combinations and an increased focus on the most risk-significant events.

In addition, the selection process for safety significant SSCs results in a comprehensive list of safety functions and their associated SSCs. It includes all SSCs that are credited with reducing the frequency or consequence of a LBE. It also provides full coherence between functions credited in the PRA and the establishment of special treatment requirements.

APPENDIX F
PRA TECHNICAL ACCEPTABILITY

F. PRA TECHNICAL ACCEPTABILITY

F.1 Introduction

Probabilistic risk assessment (PRA) will play a significant role in the licensing of new reactors. Because of this fact, the quality of the PRA used in making licensing decisions will have to be commensurate with the significance of the regulatory decision. The purpose of this Appendix is to identify the high level attributes necessary to ensure the technical acceptability of a PRA used in licensing applications. Although the quality of the PRA has to be commensurate with the specific application, this appendix provides the general attributes necessary for a high quality PRA that will be utilized fully in the licensing process. The required scope of the PRA and the corresponding attributes for each technical element are addressed. Specifically, attributes are provided for all the technical elements of a PRA required to calculate the frequency of accidents, the magnitude of radioactive material released, and the resulting consequences. In addition to delineating the PRA attributes, some unique aspects of new reactors that will impact the PRA are identified.

The attributes focus on a PRA of the reactor core that includes both internal and external events during all modes of operation. A licensee for a new reactor may choose to perform a fully integrated PRA that includes all sources of radioactivity and all accident initiating events during all modes of operation. Alternatively, the licensee may choose to perform separate PRAs for internal and external events, for different sources of radioactivity, and for different operating modes. In either case, the PRAs must reflect the as-built, as-operated plant and the attributes presented in this appendix should be met.

This appendix builds on existing PRA technical characteristics and attributes delineated in Regulatory Guide 1.200 and the High Level Requirements (HLRs) currently identified in the existing PRA standards. The PRA attributes and HLRs provided in these documents were reviewed and modified to make them generic for different reactor types, modes of operation and accident initiators. In addition, some of the attributes from RG 1.200 and HLRs from the PRA standards were generalized to address different accident end states and associated risk metrics. The Supporting Requirements (SRs) in the PRA standards were also reviewed and in some cases, the content of an SR was deemed to contain an important high level attribute that should be included in this appendix.

F.2 Scope of the PRA

The scope of the PRA is defined by the challenges included in the analysis and the level of analysis performed. These are in turn determined by how the PRA will be used in the licensing, construction, and operation of the reactor. Specifically, the scope of an new reactor PRA will be defined by the following:

- how the PRA is used to address licensing, construction, and operation issues;
- the plant operating states that must be included in the resolution of issues;
- the types of initiating events that can disrupt the normal operation of the plant leading to the release of those materials; and
- the risk metrics chosen in the licensing process.

The required scope and level of detail of a PRA will increase during the licensing process and will ultimately be dependent upon how PRA is used in each licensing phase. Section 7.2 identifies some potential PRA applications during the licensing, construction, and operation phases of an new reactor. The applications include identification of Licensing Basis Events (LBEs); identification of systems, structures, and components requiring special treatment and monitoring under programs like the Maintenance Rule; development of operator procedures and training programs, comparison of the PRA results to quantitative goals (i.e, the Quantitative Health Objectives and the Frequency-Consequence Curve provided in Chapter 6); and the use of a risk monitor to control the plant configuration in a risk-informed manner. The increased use of PRA in the licensing process will require that the PRA reflect the as-built and as-operated plant even as the plant is modified during its operating history.

The risk perspectives used in the licensing of new reactors should be based on the total risk connected with the operation of the reactor which includes not only full power operation but also low-power and shutdown conditions. The specification of plant operating states (POSs) is an accepted method to subdivide the plant operating cycle into unique operational states for use in the PRA process. Each POS is a configuration where the plant conditions (e.g., core power level, coolant level, primary temperature, containment status, decay heat removal mechanisms) are relatively constant and are distinct from other configurations that impact the risk parameters evaluated in a PRA. The POSs for new reactor designs may be substantially different from those for current light water reactors (LWRs). For example, a proposed Pebble Bed Modular Reactor (PBMR) design will utilize online refueling which will preclude the need to consider a separate refueling POS. However, consideration of refueling accidents during power operation will have to be considered. The high level attributes for defining POSs for future reactor designs are shown in Table F-1.

Table F-1 Plant operating state and hazardous source identification attributes.

Item	Attribute
POS-1	Use a structured and systematic process to identify the unique plant operation states (POSs) that encompasses all modes of plant operation.
POS-2	Group POSs into classes such that the operation characteristics are similar.
POS-3	Determine the frequency and duration for each POS.
RSI-1	Identify the radioactive and hazardous other sources in the plant that pose a risk to the public or plant operators.

The types of initiating events that can challenge a plant include failure of equipment from internal plant causes such as hardware failures, operator actions, floods or fires, or external causes such as earthquakes, airplane crashes, or high winds. The risk perspective used in the licensing of an new reactor should be based on a consideration of the total risk, which includes both internal and external events. For this reason, the PRA attributes presented in this section address all potential initiators during all modes of operation. The licensee may choose to perform a fully integrated PRA that examines all accident initiators or perform separate PRAs for internal and external events. In either case, the identified PRA attributes are applicable.

Finally, the risk metrics used to help make risk-informed licensing decisions will affect the scope of the PRA. Since the Framework is using a frequency-consequence curve to identify licensing basis events and in classifying systems, structures and components (SSCs), the PRA must evaluate the frequency of accidents, the magnitude of radioactive material released, and the resulting consequences. Additional required risk metrics such as importance measures or surrogates for the QHOs may also affect the required attributes and scope of the PRA. In addition, risk assessment techniques and evaluated metrics may be used to address licensing issues that affect the environment. The PRA attributes presented in this section cover the PRA technical elements necessary for evaluating the risk to the public and the environment.

The PRA technical elements are shown in Table F-2. They are divided into three levels of analysis for purposes of identifying high-level PRA attributes. The first level, Accident Sequence Development, consists of an analysis of the plant design and operation focused on identifying the accident sequences that could lead to a release of radioactive material from the reactor core or other locations, and their frequencies. This level of analysis includes accidents initiated during both internal and external events and during all modes of reactor operation. This level of analysis provides an assessment of the adequacy of the plant design and operation in preventing radioactive material release but does not permit an assessment of the associated risk. For existing LWR cores, a PRA of this level is referred to as a Level 1 PRA.

The second level, Release Analysis, consists of an analysis of the physical processes of the accident, the corresponding response of confinement barriers (including the controlled leakage barrier that provides a fission product containment functional capability), and the transport of the material to the environment. The end point of this level of analysis is the estimation of the inventory of radioactive material released to the environment and the timing of the release. As a result, accident sequences can be categorized with regard to their frequency and severity and time of release. Although an analysis to this level also does not provide an estimate of the risk to the public, it does provide a relative measure of risk that can be useful in risk-informed licensing applications. For existing LWR cores, a PRA that includes both the Accident Sequence Development and Release Analysis technical elements is referred to as a Level 2 PRA.

Table F-2 Technical elements of a PRA.

Level of Analysis	Technical	Element
Accident Sequence Development	• Initiating event analysis • Success criteria evaluation • Accident sequence analysis • Systems analysis	• Human reliability analysis • Parameter estimation • Accident sequence quantification
Release Analysis	• Accident progression analysis	• Source term analysis
Consequence Assessment	• Consequence analysis	• Health and economic risk estimation

The third level, Consequence Assessment, analyses the transport of radioactive material through the environment and assesses the health and economic consequences resulting from accidents.

An analysis that includes all three levels described in Table F-2 allows for the assessment of risk since it provides both the frequency and consequence of potential accident sequences. For existing LWRs, a PRA of the reactor core that includes the Accident Sequence Development, Release Analysis, and Consequence Assessment technical elements is referred to as a Level 3 PRA.

It should not be inferred that the PRAs for all new reactors will involve the three separate levels of analysis shown in Table F-2. Depending on the risk metrics used in the licensing process, results typically provided from the “accident sequence development” level may not be utilized. It is possible that a PRA for some new reactor designs will develop accident sequences that start with an initiating event and end at radioactive release to the environment (i.e., the technical elements for the first two levels shown in Table F-2 would be performed together). A consequence assessment would then be performed for the resulting end states. It also should not be inferred that the technical elements will be performed in the order presented in Table F-2. For example, “accident progression analysis” may be performed before the “accident sequence analysis.” Finally, it is important to realize the various PRA technical elements may be worked in parallel and iteration between technical elements will be a necessary component of the PRAs for new reactors.

F.3 Accident Sequence Development Technical Elements

The PRA used in licensing new reactors will have to be full scope, include both internal and external events and address the reactor during all operating modes. The attributes for the accident sequence development portion of a full scope PRA are discussed in this section. Separate sets of attributes are presented to address the different methods used to analyze internal events, internal flooding, internal fire, seismic events, and other external events.

F.3.1 Internal Events Analysis

Internal events refers to accidents resulting from internal causes in the plant initiated by hardware failures, operator actions, and internal fires and floods. The technical elements for a PRA that addresses hardware and operator related internal initiating events are discussed in this section. Internal initiators that result in floods or fires require additional PRA attributes which are discussed separately in Sections F.3.2 and F.3.3, respectively.

The PRA models, system success criteria, and data developed for the analysis of internal events form the basis for the analysis of other accident initiators. Modification of these models, including human error probabilities, is often required to reflect the affect of internal flooding, fire, and external event initiators on accident progression including SSC and human response. In addition, additional models and data can also be required for the analysis of these other initiators. Thus, the attributes identified in this section are applicable for all accident initiators. Additional attributes for analyzing other accidents are presented in subsequent sections and include considerations for modifying the internal event models and human error probabilities, and obtaining additional data.

Initiating event analysis identifies and characterizes the initiating events that can upset plant stability and challenge critical safety functions during all plant operating states (i.e., full-power, shutdown, and transitional states). A systematic method for identifying potential initiators must be utilized. Events that have a frequency of occurrence greater than $1x10^{-7}$/yr are identified and characterized. An understanding of the nature of the events is performed such that events are grouped into certain classes, depending on their frequency of occurrence. Such a grouping allows the protective features to have reliability and performance that is commensurate with the frequency of the initiator group, so as to limit the frequency of accidents to acceptable levels. The high level attributes for the initiating event analysis are shown in Table F-3. These attributes are applicable for both internal and external events.

Table F-3 Initiating event analysis attributes.

Item	Attribute
IE-1	Use a systematic process to identify a complete set of plant-specific initiators covering all modes of operation and all sources of radioactive material on site
IE-2	Identify the required safety functions and associated systems required to mitigate each identified initiating event.
IE-3	Group initiators for each POS and source of radioactive material into classes such that the events in the same group have similar mitigation requirements.
IE-4	Screening of initiating events is performed in such a fashion that no significant risk contributor is eliminated from the PRA.

For the future reactor technologies, initiating event consideration may be substantially different from those for current US LWRs. Examples are events associated with on-line refueling, recriticality due to more highly enriched fuel and fuels with higher burnup, and chemical interactions with some reactor coolants or structures. In particular, initiators that cause a plant trip and result in operators taking actions that could defeat important safety features in new plants (e.g., passive cooling) or cause conditions outside the designer' expectations, could be important. Furthermore, the identification of initiators will be more important than for in past LWR PRAs since the PRA will be used to select LBEs. For these reasons, more emphasis will be required on the use of systematic methods to identify the initiating events modeled in the PRA. Searches for applicable events at similar plants (both those that have occurred and those that have been postulated) and use of existing deductive methods (e.g., top logic models, fault trees, and Failure Modes and Effects Analysis) could both be utilized in this effort.

Success criteria analysis is used to distinguish the path between success and failure for components, human actions, trains, systems, structures and sequences given an initiating event. In all cases, the success criteria are to be fully defensible and biased such that issues of manufacturer or construction variability, code limitations and other uncertainties are unlikely to result in a failure path being considered a success path. Ensuring that success paths are truly success paths will be supported by requiring regulatory margin for selected key variables and by encouraging the incorporation of operational margin. For any given criterion, when the margin between the selected criteria and the estimated failure point is small, it becomes more essential that the success criteria calculations account for uncertainty in the models and input parameters.

The codes used to evaluate success criteria need to be validated and verified in sufficient detail over the expected range of parameters. The sequence of events in future reactors could be much longer than currently seen in current US LWRs. Thus the parameters used in evaluating key parameters in the PRA models (e.g., timing information used to evaluate human error probabilities and the environments that components will have to operate) will need to be determined for the duration of the sequence. In addition, the success criteria for some systems may need to change as the sequence progresses.

The success criteria evaluation will have to include systems needed to mitigate accidents involving core and containment cooling, inventory makeup, and reactivity control. The high level attributes for the success criteria analysis are shown in Table F-4. They are applicable to success criteria evaluations required for the analysis of internal and external initiators.

New reactor designs are moving towards the simplification of plant systems with extensive use of passive features. A simplified system is one that is more easily operated and maintained or has reduced the number of components necessary to provide the safety and performance functions (thereby reducing the number of failure points and modes) and, therefore, should be more resistant to human errors. Passive systems that rely on pressure, gravity, or thermal gradients offer the opportunity to reduce the number or complexity of active systems and potentially the need to rely on active safety-grade support systems. The challenge is to demonstrate the capability and reliability of passive systems to meet the core cooling requirements and to deal with their longer response time in PRAs. In addition, there is the potential for events during an accident to adversely effect the structural integrity of the passive systems (e.g., jet impingement could result in a failure of an accumulator support causing the accumulator to fall and fail). The impact of accident phenomena on passive systems also needs to be considered in the PRAs for new reactors.

Table F-4 Success criteria analysis attributes.

Item	Attribute
SC-1	Perform thermal/hydraulic, structural, and other supporting engineering evaluations capable of providing success criteria for each safety function and system available to perform those functions, event timing information sufficient for determining sequence timing and required mission times, determining the relative impact of accident phenomena on SSC and human actions, and the impact of uncertainty on the determination of these parameters.
SC-2	Base the overall success criteria for the PRA and the system, structure, component, and human action success criteria used in the PRA on best-estimate engineering analyses that reflect the features, procedures, and operating philosophy of the plant.
SC-3	Codes used to evaluate success criteria are applicable for evaluating the phenomena of interest and have been validated and verified in sufficient detail over the expected range of parameters.

Accident sequence analysis determines, chronologically (to the extent practical), the different possible progression of events (i.e., accident sequences) that can occur from the start of the initiating event to either successful mitigation or a required end-state (e.g., different levels of radiation exposure at the site boundary consistent with the proposed frequency-consequence criteria in Chapter 6). Although the accident sequences for current LWRs generally delineate sequences for the core and containment response in separate levels of the PRA, it may be more reasonable for new reactors PRAs to include both aspects in a single accident sequence model (i.e., the accident progression analysis may be incorporated into the Accident Sequence Development portion of a PRA). In either case, the accident sequences account for all the systems that are used (and available) and operator actions performed to mitigate the initiator based on the defined success criteria and that will be delineated in plant operating procedures (e.g., plant emergency and abnormal operating procedures) and training (note that the accident sequence delineation will identify the steps needed in emergency procedures and help guide the training of operators). The availability of a mitigating system should include consideration of the functional, phenomenological, time-related, and operational dependencies and interfaces between the different systems and operator actions during the course of the accident progression. For multi-unit sites, cross-tying systems between units is included in the accident sequence models. The accident sequences must be delineated for all accident initiators involving the reactor core and other radioactive sources onsite. The high level attributes for the accident sequence analysis are shown in Table F-5 and are applicable for accident sequences resulting from either internal or external events.

If, as delineated in this Framework, accident sequences will be used to define the LBEs and determine the safety significance of systems, the accident sequences delineated will be more than those that result in either a mitigated state or severe core damage as is currently done in LWR PRAs. Sequences resulting in intermediate states of core damage and/or levels of radioactive release will also have to be delineated and quantified. The delineation of these sequences may require that different levels of system success criteria be defined and delineated as separate events in the PRA models. An important element of the accident sequence analysis element is to define the necessary end states that match the required licensing risk metrics whether they be the dose at the site boundary or a different risk metric (e.g., surrogates to the Quantitative Health Objectives).

Table F-5 Accident sequence analysis attributes.

Item	Attribute
AS-1	Define the end states to be considered in the accident sequence delineation.
AS-2	Identify the plant-specific scenarios that can lead to successful mitigation, radiation exposure at the site boundary, or other end states following each initiating event or initiating event category.
AS-3	Include all capable mitigating systems and operator actions (including recovery actions) that would be expected to be used for each safety function required to reach the defined end states.
AS-4	Include functional, phenomological, time-related, and operational dependencies and interfaces (including those resulting from modular designs, shared systems at multiple unit sites, and different POSs) that can impact the ability of the mitigating systems to operate and/or function.

Current PRAs are usually performed for a single unit or sometimes for two sister units. New reactors (e.g., PBMR) may operate multiple modular units together at a site with a centralized control room. The PRAs for modular reactor designs need to address potential interactions among the multiple units. This includes common accident initiators, common support system dependencies, interactions between units caused by accident phenomena (e.g., smoke generated by fire), and the potential effects of smaller operator staffs in a common control room responding to potential common cause initiators (such as seismic events).

Future reactor accident sequence could be simplified with the use of passive systems. A passive system might force the sequence to successful mitigation quickly and without the use of other systems or operator interaction. The presence of passive systems requires that a PRA accurately characterize accident sequences to a level of detail that identifies the thermal-hydraulic behavior of the reactor necessary to insure that the passive system is functioning in the regime it was designed for.

Systems analysis identifies the different combinations of failures that can prevent a required mitigating system from performing its function as defined by the success criteria evaluation. The developed system model represents the as-built and as-operated system and includes hardware and instrumentation (and their associated failure modes), and human failure events that would prevent the system from performing its defined function. During design phases of a new nuclear power plant, the systems analysis can be used to help design the system and establish the required operating procedures. The basic events representing equipment and human failures are developed in sufficient detail in the model to account for dependencies between the different systems and to distinguish the specific equipment or human events that have a major impact on the system's ability to perform its function. Different initial system alignments, including those utilized during different POSs and those required to support the development of the accident sequences necessary to define the LBEs, are also modeled. The high level attributes for the systems analysis are shown in Table F-6. The attributes are applicable for the analysis of systems required to mitigate either internal and external initiating events.

Table F-6 Systems analysis attributes.

Item	Attribute
SY-1	Develop models for systems identified in the accident sequence analysis that include both active and passive component failures, human errors, equipment unavailability due to test and maintenance, and external conditions for which the system will not successfully mitigate an accident.
SY-2	Develop the system models using success criteria that are supported with engineering analysis.
SY-3	Include common cause failures, inter-system and intra-system dependencies (e.g., support systems, harsh environments, and conditions that can cause a system to isolate or trip), alternative alignments, and dependencies on the POS in the system model development.
SY-4	Develop system models for those systems needed to support the systems contained in the accident sequence analyses.
SY-5	Develop system models, as required, to determine how initiating events can occur.

The systems analysis for PRAs of new reactors will have to address unique features including:

- Simplified and passive systems
- Digital I&C systems
- Smart equipment

PRA methods for modeling these types of systems may have to be developed.

Future reactor designs may use passive systems and inherent physical characteristics (confirmed by sensitive nonlinear dynamical calculations) to ensure safety, rather than relying on the active electrical and mechanical systems. For plants with passive systems, fault trees may be very simple when events proceed as expected and event sequences may appear to have very low frequencies. The real work of PRA for these designs may lie in searching for unexpected scenarios. Innovative ways to structure the search for unexpected conditions that can challenge design assumptions and passive system performance will need to be developed or identified and applied to these facilities. The risk may arise from unexpected ways the facility can reach operating conditions outside the design assumptions. A HAZOP-related search scheme for scenarios that deviate from designers' expectations and a structured search for construction errors and aging problems may be the appropriate tools. Some example scenarios include:

- The operator and maintenance personnel place the facility in unexpected conditions.

- Gradual degradation has led to unobserved corrosion or fatigue or some other physical condition not considered in the design.

- Passive system behavior (e.g., physical, chemical, and material properties) is incorrectly modelled.

Digital systems typically have not been used extensively in operating LWRs and, thus, have not been considered in many existing PRAs. In new reactors, instrumentation and control (I&C) systems will normally be digital. Digital I&C systems may have different operational and reliability characteristics than the analog systems used in current LWRs. Thus, digital systems may have failure modes that are different from those in analog systems. For example, digital systems may fail due to smaller voltage spikes or sooner under loss of cabinet ventilation, or may fail due to software errors. Inadequate consideration of potential digital system failure modes can lead to the failure of the system to function properly under postulated conditions. It is not readily apparent that these reliability aspects of digital systems can be addressed with existing PRA methods. The technical considerations and guidance for including digital systems in PRA needs to be developed.

Automated surveillance and diagnostic systems, as well as artificial intelligence systems are currently being developed and likely will be incorporated in new reactor designs within the next 10 years. Smart equipment incorporates sensors, data transmission devices, computer hardware and software, and human-machine interface devices that continuously monitor and predict the system performance and remaining useful life of equipment. The use of smart equipment could replace the current practice of scheduled inspection and maintenance with maintenance or replacement dictated by the measured condition of the equipment and predictions of its continued performance. Modeling considerations include the reliability of the smart equipment sensors, data transmission devices, and computer systems. In addition, the reliability of the software developed to predict the continued performance of equipment and the decision making process (i.e., artificial intelligence logic) will have to be addressed.

Human reliability analysis identifies the human failure events (HFEs) that can negatively impact normal or emergency plant operations and systematically estimates the probability of the HFEs using data (when available), models, or expert judgment. Human errors associated with normal plant operation (referred to as pre-accident errors) leave a component, train, or system in an unrevealed, unavailable state. Human failure events during emergency plant operations (referred to as post-accident errors) result in either the failure to perform a required action (error of omission) or the performance of a wrong action (error of commission). Errors of commission can be particularly important during shutdown and refueling POSs when a substantial amount of maintenance is being performed. Quantification of the probabilities of these HFEs is based on plant and accident specific conditions, where applicable, including any dependencies among actions and conditions. The high level attributes for the human reliability analysis are shown in Table F-7. They are applicable to HFEs that can occur following either an internal or external event.

During the design and startup phases of an new reactor, the PRA can provide valuable insights regarding the importance of human actions, which can then be emphasized in procedures (e.g., plant emergency and abnormal operating procedures) and training programs. Consideration should be given to conditions that could shape the action's failure probability (e.g., complexity, time available for action completion, procedure quality, training and experience, instrumentation and controls, human-machine interface and the environment). It is expected that procedural guidance will be developed for all actions credited within the PRA and that training will be risk-informed. In addition, the modeling of human actions in the PRA along with the use of simulators and/or mockups can be used to show that staffing is adequate for the evaluated level of safety.

Table F-7 Human reliability analysis attributes.

Item	Attribute
HR-1	Use a systematic process to review normal and emergency procedures and work practices to identify and define HFEs that would result in initiating events or pre- and post-accident human failure events that would contribute to or negatively impact the mitigation of initiating events.
HR-2	Account for dependencies between human actions when evaluating HFEs.
HR-3	Place HFEs in the PRA logic models such that the impact of the HFEs on components, trains, and systems are properly accounted for.
HR-4	Develop the probabilities of the identified HFEs taking into account scenario and plant-specific factors (e.g., procedures, simulator training, POS-specific performance shaping factors, man-machine interface, and equipment accessibility) and incorporating dependencies between different HFEs.
HR-5	Use plant-specific engineering evaluations to determine cues and the available time window for required operator actions and the environments present at the sites for performing required actions.
HR-6	Model recovery actions only when it had been demonstrated that the action is plausible and feasible.

The operators' role in new reactors will be different than that in current generation reactors. New reactors are proposed to be built on the premise that they will be less susceptible to human errors and that, if an event occurs, human intervention will not be necessary for an extended period of time. In addition, the operators' interactions with plant systems may be different in a digital I&C environment. Differences in the man machine interface related to new types of displays, touch screen controls, etc. may impact the potential operator errors. In the extreme, with "smart" control systems, the operators' role could become more of a "supervisory" task as opposed to the "hands-on" operation in current plants. Thus, the main "job" of the operators may be to monitor system behavior and ensure that shutdown occurs properly when necessary. In addition, operator performance may be affected by having multiple modules that share the same control room. Thus, the tasks to be performed by operating crews in new reactors will be different from that in existing control rooms. The likelihood of errors of commission or omission needs to be understood under these conditions.

Parameter estimation involves the quantification of the frequencies of the initiating events and the equipment failure probabilities (including common cause events) and equipment unavailabilities of the systems modeled in the PRA. The estimation process includes a mechanism for addressing uncertainties, has the ability to combine different sources of data in a coherent manner, including the actual operating history and experience (when available) of the plant, applicable generic experience, and expert elicitation. The plant-specific data used in this process reflects the configuration and operation of the plant. Initially, there will be no available date for new reactors. Therefore, parameter estimates will have to be generated using generic data sources. To the extent possible, the generic data values should reflect the design, environmental, and service condition of the components for which the parameter estimates are generated. Expert elicitation can be used when plant-specific and generic data is unavailable and/or of poor quality. The high level attributes for parameter estimation required in the analysis of all accident initiators are shown in Table F-8.

Table F-8 Parameter estimation attributes.

Item	Attribute
PE-1	Define each parameter (i.e., initiating event, component failure, component unavailability due to test or maintenance, and component common cause failures) in terms of the PRA logic models, basic event boundary, POS, and the appropriate model used to evaluate the event probability or frequency.
PE-2	Include consideration of the design, environmental, and services conditions of the components when grouping components into a homogeneous population for the purpose of component failure probability estimation.
PE-3	Chose generic parameter estimates (i.e., initiating event frequencies and component failure probabilities, including common cause) and collect plant-specific data consistent with the parameter definition of PE-1 and the grouping of PE-2 and accounting for POS-specific impacts where appropriate.
PE-4	Base parameter estimates on relevant generic industry plant-specific evidence and integrate generic and plant-specific data (when feasible) using accepted techniques and models such as those provided in NUREG/CR-6823 [NRC 2003].
PE-5	Provide both mean values and a statistical representation of the uncertainty for the parameters.

The use of appropriate data is crucial to the quality of the PRA. New reactors introduce different systems and components and, hence, the data may not be sufficient and in some areas appropriate. Furthermore, the susceptibility of these components to failure in the environments created during accidents, including external events, needs to be addressed. Understanding the uncertainties is a very important aspect for any PRA; this is especially true for new reactors, given the limited or lack of operating experience and the expected significant use of the PRA in the licensing process.

Accident sequence quantification involves integration and evaluation of the PRA models to provide estimates of the required risk metrics needed to support reactor licensing including an understanding and quantification of the contributors to uncertainty. The significant contributors to the risk metrics are also identified and include the importance of radioactive material sources, POSs, initiating events, accident sequences, component failures, human actions, important dependencies, and key assumptions and models. Importance measures are used in the licensing process to determine safety-significant SSCs which in turn determines the special treatment they will receive to ensure their reliability. In addition, the quantification process is used to trace the results to the inputs and verify that the results reflect the design, operation, and maintenance of the plant. The mechanics of the quantification process are also reviewed to verify that computer codes are providing the correct results. This can include validation of computer codes and verification that truncation limits used in the process are not significantly impacting the quantified results. The high level attributes for accident sequence quantification are shown in Table F-9.

If, as delineated in this Framework, accident sequences will be used to define the LBEs and determine the safety significance of systems, the accident sequences delineated will be more than those that result in either a mitigated state or severe core damage as is currently done in LWR PRAs. Sequences resulting in intermediate states of core damage and/or levels of radioactive release will also have to be delineated and quantified. The evaluation of these sequences will require that the success of components, trains, and systems be properly accounted for in the sequence quantification process.

Table F-9 Accident sequence quantification attributes.

Item	Attribute
QU-1	Quantify the required end-state for each accident sequence and provide the required risk metrics.
QU-2	Use appropriate models and codes that have been verified and validated for the quantification.
QU-3	Ensure that method-specific limitations and features (e.g., truncation) do not significantly change the results of the quantification process.
QU-4	Ensure that all dependencies are appropriately included in the quantification process (e.g., shared systems, initiating event impacts, and common human actions). Also ensure that system successes are properly accounted as well as failures.
QU-5	Identify significant contributors (including assumptions, initiating events, POSs, accident sequences, component failures, and human errors) to the required end-states and verify the results reflect the as-built and as-operated plant.

Table F-9 Accident sequence quantification attributes.

Item	Attribute
QU-6	Characterize and quantify the uncertainties in the PRA results including parameter and model uncertainty and the contribution from assumptions. Understand their potential impact on the results.

Identification and quantification of uncertainties in an new reactor PRA will help decision makers determine whether reducing the uncertainties by performing more research or strengthening the regulatory requirements and oversight (e.g., defense-in-depth and safety margins) should be pursued. A PRA provides a structured approach for identifying the uncertainties associated with modeling and estimating risk.

There are three types of uncertainty: parameter, modeling, and completeness:

- Parameter uncertainty associated with the basic data; while there are random effects from the data, the most significant uncertainty is epistemic (i.e., is this the appropriate parameter data for the situation being modeled?)

- Model uncertainty associated with analytical physical models and success criteria n the PRA can appear because of modeling choices, but will be driven by the state-of-knowledge about the new designs and the interactions of human operators and maintenance personnel with these systems

- Completeness uncertainty associated with factors not accounted for in the PRA by choice or limitations in knowledge, such as unknown or unanticipated failure mechanisms, unanticipated physical and chemical interaction among system materials, and, for PRAs performed during the design and construction stages, and all those factors affecting operations (e.g., safety culture, safety and operations management, training and procedures, use of new I&C systems)

The quantification of parameter uncertainty is well understood, and additional guidance is not needed beyond establishing those uncertainties. Sensitivity studies are an important means for examining the impacts of modeling uncertainties. Sensitivity studies can be useful early in the licensing process to highlight important areas of uncertainty where more research may be required to reduce the uncertainty, or, if the uncertainty cannot be reduced, where more defense- in-depth may be needed. The PRA can be used to examine the tradeoff between reducing the uncertainty through research and adding defense-in-depth or additional safety margin to cope with the uncertainty. With regard to completeness uncertainty, PRAs will always be susceptible missing unknown factors that can influence the results.

F.3.2 Internal Flood PRA

An internal flood PRA generally utilizes the models generated for random internal initiators modified to include consideration of the type of flood initiator, the potential for flood propagation, and the impact of flooding environments on both the equipment located in the flooded areas and on the operator actions. For certain new reactor designs, the flooding mediums of concern may include other fluids (e.g. liquid metal or helium) in addition to water and steam. The attributes for an internal flood PRA must address all of these mediums and include internal floods initiated during all modes of plant operation. Internal flooding initiators that can adversely affect sources of radioactivity other than the core are also analyzed.

An important aspect of flooding and other spatial-related accidents (e.g., fire, seismic, and other external event analysis) is the determination of whether failure of equipment in one or more locations can result in core damage. The evaluation of these types of initiators provides critical information on the adequacy of the spatial separation and redundancy of equipment necessary to prevent and mitigate these initiators.

Flood source identification identifies the plant areas where flooding or a release of other coolant material (e.g., helium) could result in significant accident sequences. Flooding areas are defined on the basis of physical barriers, mitigation features, and propagation pathways. For each flooding area, flood sources that are due to equipment (e.g., piping, valves, pumps) and other sources internal to the plant (e.g., tanks) are identified. Specific flooding mechanisms are examined that include failure modes of components, human-induced (including maintenance-induced) mechanisms, and other release mechanisms. Flooding types (e.g., leak, rupture, spray), flood sizes, and temperature and pressure are determined. Flood areas that do not have flood sources can be screened from further analysis if they contain no flood initiators or no propagation paths from other areas. Plant walkdowns are performed to verify the accuracy of the information. Temporary alignments during different POSs are included in this process. The high level attributes for flood source identification are shown in Table F-10.

Table F-10 Flood source identification attributes.

Item	Attribute
FSI-1	Define flood areas by dividing the plant into physically separate areas where flood areas are independent in terms of flooding effects and flood propagation. Temporary alignments during different POSs are included in this process.
FSI-2	Identify potential flood sources including propagation from other areas, their associated flooding mechanisms, and the harsh environments that are introduced. Unique sources and alignments during different POSs are identified.
FSI-3	Characterize the types of potential fluid releases, their capacities, and other important parameters such as temperature and pressure.
FSI-4	Perform plant walkdowns to verify the definition of flood areas, the sources of flooding, and the location of SSCs.

Flood scenario evaluation identifies the potential flooding scenarios for each flood source by identifying flood propagation paths from the flood source to its accumulation point (e.g., pipe and cable penetrations, doors, stairwells, failure of doors, or walls). Scenarios are developed for all POSs. Plant design features (e.g., flood alarms, flood dikes, curbs, drains, barriers, or sump pumps) or operator actions that have the ability to terminate the flood are identified in this effort. The susceptibility of each SSC in a flood area to flood-induced mechanisms is examined (e.g., submergence, spray, high or low temperature, pipe whip, and jet impingement). Flood scenarios are developed by examining the potential for propagation and giving credit for flood mitigation. Flood scenarios can be eliminated on the basis of accepted screening criteria (e.g., a flood within the area does not cause an initiating event or an area with no significant flood sources and the nature of the flood does not cause equipment failure). The high level attributes for flood scenario evaluation are shown in Table F-11.

Flood sequence quantification provides estimates of the risk metrics due to internal floods. The flood-induced initiating events are identified and quantified, and the internal event PRA models are modified to include flooding effects. Specifically, accident sequence and system models are modified to address flooding phenomena and flood-induced SSC failures, human error probabilities are adjusted to account for performance shaping factors (PSFs) that are due to flooding, and flood-specific human errors (e.g., recovery actions) are added where appropriate. Additional analyses are performed as required (e.g., calculations to determine success criteria for flooding mitigation and parameter estimates for flooding failure modes). The internal flood accident sequences are quantified to provide the required end-state frequencies. The sources of uncertainty are identified and their impact on the results analyzed. The sensitivity of the model results to model boundary conditions and other key assumptions is evaluated using sensitivity analyses to look at key assumptions both individually or in logical combinations. The combinations analyzed are chosen to fully account for interactions among the variables. The high level attributes for flood scenario evaluation are shown in Table F-12.

Table F-11 Flood scenario evaluation attributes.

Item	Attribute
FSE-1	For each flood source in each flood area, identify propagation paths to other flood areas.
FSE-2	Identify plant design features (e.g., drains, sumps, alarms, dikes) or operator actions that have the ability to terminate the flood propagation.
FSE-3	Identify the SSCs located in each flood area and associated flood propagation paths and identify their susceptibility to the failure mechanisms introduced by the flood source.
FSE-4	Develop potential flooding scenarios for each POS (i.e., the set of knowledge regarding the flood area, source, flood rate and capacity, operator actions, and SSC damage) that accounts for flood propagation, flood mitigation systems, and operator actions, and identifies susceptible SSCs.
FSE-5	Temporary configurations of barriers during different POSs that affect flood propagation and mitigation are included in the development of flood scenarios for each POS.
FSE-6	Screen out potential flood areas using acceptable criteria (e.g., none of the flood scenarios can cause a reactor trip or affects accident mitigating systems).

F.3.3 Internal Fire PRA

An internal fire PRA generally utilizes the models generated for random internal initiators modified to include consideration of the fire initiator, the potential for fire and smoke propagation, and the impact of fire on both the equipment located in the areas and on the operator actions. Of specific concern is the impact of the fire on cables leading to the potential for spurious component operation, loss of motive power, or loss of the ability to initiate a component. As is the case for other internal initiators, an internal fire PRA includes fires during all modes of plant operation and can address all sources of radioactivity including the reactor core, waste, and the spent fuel pool.

An important aspect of internal fire and other spatial-related accidents (e.g., flooding, seismic, and other external event analysis) is the determination of whether failure of equipment in one or more locations can result in core damage. The evaluation of these types of initiators provides critical information on the adequacy of the spatial separation and redundancy of equipment necessary to prevent and mitigate these initiators. For fire, the performance of a fire PRA for an new reactor can be used in place of the 10 CFR 50 Appendix R safe-shutdown analysis that was required for older LWRs.

Table F-12 Flood sequence quantification attributes.

Item	Attribute
FSQ-1	Identify the initiating event (from the internal event PRA) that would occur in each flood scenario using a structured and systematic process. Grouping of initiators for different flood areas and sources into classes can be performed when the events in the same group have similar mitigation requirements.
FSQ-2	Estimate flood initiated event frequencies per the attributes in the Parameter Estimation section.
FSQ-3	Review the accident sequence models from the internal event PRA for the appropriate initiating event and modify sequences as necessary to account for any flood-induced phenomena.
FSQ-4	Modify the system models to account for flooding-induced component failures.
FSQ-5	Modify human recovery failure events to account for flood-related impacts and quantify any flood-specific recovery action.
FSQ-6	Quantify the flood scenarios to obtain the desired risk metrics in accordance with the attributes identified for the internal event PRA accident sequence quantification but accounting for the combined effects of failures caused by flooding and by random equipment failures or unavailability due to test or maintenance.
FSQ-7	Identify significant contributors (including assumptions, initiating events, POSs, accident sequences, component failures, and human errors) to the required end-states and ensure that all flood significant sequences are traceable and reproducible.
FSQ-8	Characterize and quantify the uncertainties in the results including parameter and model uncertainty and the contribution from assumptions. Understand their potential impact on the results.

Fire area screening can be performed to reduce the amount of work involved in performing a fire PRA. The plant is first partitioned into fire areas based on selected criteria which includes consideration of both permanent (e.g., fire-rated walls) and active fire barriers (e.g., fire dampers and water curtains). Temporary alignments during different POSs are also considered. Each identified fire area is subjected to a screening analysis with the goal of eliminating fire areas which are not risk significant from detailed analysis. Both qualitative and quantitative screening analyses can be used. Qualitative screening identifies fire area where an unsuppressed fire in the area does not result in damage to equipment that can result in a plant transient, is required to mitigate the transient, and does not spuriously activate equipment that would adversely affect operation of mitigation equipment. For areas that can not be qualitatively screened, quantitative screening can be performed. Quantitative screening generally involves bounding quantitative methods that combines estimates of the frequency of fires and the resulting conditional plant damage. The limited quantitative assessment generally assumes all equipment in the fire area is lost and therefore does not credit fire detection and suppression activities and other features that might limit the extent of fire growth and damage (e.g., fire wraps and separation). Plant walkdowns are performed where possible to verify the accuracy of the information used in defining the fire areas and in performing the screening analysis. During the early design phase, verification of the assumptions and screening criteria will come from evaluating the plant designs and operational philosophies. The high level attributes for fire area screening are shown in Table F-13.

Table F-13 Fire area screening attributes.

Item	Attribute
FS-1	Identify the elements or features for use in partitioning the plant into separate fire areas. Partition the plant according to this criteria. Temporary alignments during different POSs are included in this process.
FS-2	For each fire area, identify all equipment in the area that can result in a plant transient and that can be used to mitigate transients including support systems. The location of cables required for operation of the identified equipment are also identified.
FS-3	Define and justify the criteria used in both the qualitative and quantitative screening process.
FS-4	Perform and document the screening assessment. Plant configurations during different POSs are included in the screening process.
FS-5	Perform walkdowns (when possible) or design verification to confirm the screening decisions.

Fire initiation analysis determines the physical characteristics of the detailed fire scenarios analyzed for the unscreened fire areas and their frequencies. The analysis needs to identify a range of scenarios in each area (including the maximum expected fire) that result in a plant transient and significantly affect the plant response. The possibility of seismically induced fires should be considered as well as fire scenarios unique to different POSs. The physical characterization of the identified scenarios should provide the initial conditions for the models used to predict the behavior of the fire following initiation and be of sufficient detail to support the fire damage analysis (discussed subsequently). The characterization should recognize that different fire initiation mechanisms (e.g., cable overheating, high-energy switchgear faults, or transient fires) can lead to different fire scenarios. The scenario frequencies estimates reflect plant-specific experience, to the extent available, and generic industry fire information. Fire severity factors can be used to address different sizes of fires. The high level attributes for a fire initiation analysis are shown in Table F-14.

Table F-14 Fire initiation analysis attributes.

Item	Attribute
FI-1	Identify all potential fire sources and resulting scenarios in each unscreened area. Consider fire sources present during different POSs.
FI-2	Provide a physical characterization for each fire scenario that includes the fire source physical and thermal characteristics.
FI-3	Calculate fire scenario frequencies accounting for plant-specific features and using both plant-specific and generic industry experience where appropriate.
FI-4	Provide a rational bases for apportioning fire frequencies.

Some new reactor designs may present unique fire concerns. Specific examples include the fire potential related to the liquid metal and graphite used in the reactor designs and the affect that the potential fires can have on the passive systems. Identification of potential side-affects or failures of the passive systems as a result of fires will be necessary.

Fire damage analysis determines the conditional probability that sets of potentially risk-significant contributors (i.e., components including cables) will be damaged during a fire scenario. The probability that a given component is damaged by the fire is equal to the probability that the component's damage threshold is exceeded before the fire is successfully controlled or suppressed. All damage mechanisms including exposure to heat, smoke, and suppressants are considered. The analysis addresses components whose direct or indirect damage from a fire will cause an initiating event, affect the systems required to mitigate an initiating event, or cause other adverse conditions (e.g., spurious opening of a valve, spurious indications, or structural failure). Circuit analysis is required to identify how different power, control, and instrumentation cable failures result in component failure or adverse system operation. Components for which functionality under fire conditions cannot be determined are assumed to fail in the most challenging mode for the scenario being considered.

Fire models are used to predict the behavior of fires in compartments including the time to individual component damage and the potential for fire or fire effects (e.g., smoke) spreading to other areas. The fire models should reflect compartment-specific features (e.g., ventilation, geometry) and target-specific features (e.g., cable location relative to the fire). Fire growth to other compartments is accounted for in the model and addresses the availability and potential failure of both passive and active fire barriers. Configurations during different POSs must be accounted for when predicting the associated fire behavior.

The potential for fire damage should also address the potential for fire suppression prior to reaching a realistic damage threshold. The fire suppression analysis accounts for the scenario-specific time to detect, respond to, and suppress the fire. Both automatic and manual suppression efforts and the potential for self-extinguishment should be credited. The availability of suppression systems, dependencies between systems, and potential adverse affects on manual suppression efforts (e.g., smoke) are considered. Temporary alignments during different POSs are included in this evaluation.

The models used to analyze fire growth, fire suppression, and fire-induced component and barrier damage must be consistent with actual nuclear power plant fire experience, tests, and experiments. Data used in the analyses should reflect plant-specific experience to the extent practical. The high level attributes for a fire damage analysis are shown in Table F-15.

Table F-15 Fire damage analysis attributes.

Item	Attribute
FD-1	Identify all potentially significant component and barrier damage mechanisms (including impacts from exposure to heat, smoke and suppressants) and specify damage criteria.
FD-2	Identify components and barriers susceptible to fire-related damage mechanisms in each unscreened fire area. Component susceptibility should consider all potential component failure modes.
FD-3	Analyze specific fire scenarios using fire models that address plant-specific factors affecting fire growth and component and barrier damage (e.g., ventilation).
FD-4	Circuit analysis is performed to identify the impacts of fire-induced electrical cable failures.
FD-5	Evaluate the potential for propagation of fire and fire effects (e.g., smoke) between fire compartments.
FD-6	Meet the Systems Analysis attributes and include plant-specific experience and reflect scenario-specific conditions in the modeling of fire suppression systems. Address the dependency between various forms of automatic and manual suppression and account for fire-effects on manual suppression.
FD-7	Fire models and data used in the fire damage analysis are consistent with actual fire experience (when available) and experiments.
FD-8	Temporary configurations of barriers and suppression systems during different POSs are included in the fire damage analysis for scenarios specific to the POS.

Plant response analysis and quantification involves the modification of appropriate internal event PRA models in order to quantify the probability of a desired end-state, given damage to the sets of components defined in the fire damage analysis. All potential fire-induced initiating events that can result in significant accident sequences, including events such as loss of plant support systems, loss-of-offsite power, and loss of decay heat removal during shutdown are considered. For multi-unit sites, interactions between multiple nuclear units during a fire event are addressed including cross-tying systems between units. The analysis addresses the availability of non-fire affected equipment and any required manual actions. Specific fire-related response actions (e.g., de-energizing circuits or manual actions in the plant) are included in the response model. For fire scenarios involving control room abandonment, the analysis addresses circuit interactions, including the possibility of fire-induced damage prior to transfer to the alternate shutdown methods (if applicable). The human reliability analysis of operator actions addresses fire effects on operators (e.g., heat, smoke, loss of lighting, effect on instrumentation) and fire-specific operational issues (e.g., fire response operating procedures, training on these procedures, potential complications in coordinating activities).

The fire PRA quantification identifies sources of uncertainty and analyses their impact on the results. The sensitivity of the model results to model boundary conditions and other key assumptions are evaluated using sensitivity analyses to look at key assumptions both individually or in logical combinations. The combinations analyzed are chosen to fully account for interactions among the variables. Fire significant sequences need to be traceable and reproducible so the fire propagation can be followed and the consequences identified. The high level attributes for a fire plant response analysis are shown in Table F-16.

Table F-16 Fire response analysis attributes.

Item	Attribute
PR-1	Identify the fire-induced accident initiating events resulting from each fire scenario.
PR-2	Include fire scenario impacts in the models for systems required to mitigate the resulting accident initiator. Add unique fire-induced failures such as spurious operation of components as required.
PR-3	Include plant-specific fire response strategy and actions in the response analysis.
PR-4	Identify potential circuit interactions which can interfere with safe shutdown.
PR-5	Modify human recovery failure events to account for fire-related impacts and quantify any fire-specific operator action.
PR-6	Estimate the required end-state frequency for each fire-induced scenario. Quantify the fire scenarios to obtain the desired risk metrics in accordance with the attributes identified for the internal event PRA accident sequence quantification but accounting for the combined effects of failures caused by fires and by random equipment failures or unavailability due to test or maintenance.
PR-7	Identify significant contributors (including assumptions, initiating events, POSs, accident sequences, component failures, and human errors) to the required end-states and ensure that all fire significant sequences are traceable and reproducible.
PR-8	Characterize and quantify the uncertainties in the results including parameter and model uncertainty and the contribution from assumptions. Understand their potential impact on the results.

Control rooms in future reactors could look dramatically different than those in current LWRs. The ability of the operators to perform alternate shutdown upon abandonment of the control room will need to be investigated. For future reactors, operators might be able to perform alternate shutdown remotely, possibly from hand-held devices that require no interaction with the control room. The designs and capability of the systems of the future reactors should describe these possibilities.

F.3.4 Seismic PRA

A seismic analysis is required for all plants. A seismic PRA includes consideration of the impact of the seismic event on both the equipment and on the operator actions. Of specific concern is the impact of the earthquake on relays which can lead to the potential for spurious component operation or loss of the ability to initiate a component. In addition, an earthquake can cause correlated failures of similar components located at different locations and other dependent failures due to mechanisms such as structural failure. As is the case for internal initiators, a seismic PRA includes analysis of seismic events that occur during all modes of plant operation and that can affect different sources of radioactive material at the plant site.

Seismic hazard analysis estimates the frequency of different intensities of earthquakes based on a site-specific evaluation reflecting recent data and site-specific information. The analysis can be based on either historical data or a phenomenological model, or a mixture of the two. If existing studies are used to establish the seismic hazard, it is necessary to confirm that the basic data and interpretations that were used are still valid in light of current information. What ever the source of data, the hazard analysis should reflect the composite distribution of the informed technical community. Necessary inputs to the analysis include geological, seismological, and geophysical data, local site topography, surficial geologic and geotechnical properties. All sources of potentially damaging earthquakes and all credible mechanisms influencing vibratory ground motion should be accounted for in the hazard analysis. In addition, the effects of the local site response (e.g., topography and site geotechnical properties) should be included. Other seismic hazards such as fault displacement, landslide, soil liquefaction, or soil settlement should be reviewed to determine if they need to be included in the seismic PRA. Uncertainties in each step of the hazard analysis are propagated and included in the final hazard estimates for the site. The high level attributes for a seismic hazard analysis are shown in Table F-17.

Table F-17 Seismic hazard analysis attributes.

Item	Attribute
SH-1	Base the frequency of earthquakes at the site on a site-specific probabilistic seismic hazard analysis that reflects the composite distribution of the informed technical community. If an existing hazard analysis is used, confirm that the data and information is still valid.
SH-2	The hazard analysis uses pertinent site information (e.g., geological, seismological, and geophysical data; site topography) and historical information.
SH-3	The hazard analysis considers all sources of potentially damaging earthquakes that can affect the seismic hazard at the site.
SH-4	The hazard analysis accounts for all credible mechanisms influencing vibratory ground motion that can occur at the site.
SH-5	Perform screening to address other seismic hazards, such as; fault displacement, landslide, soil liquefaction, or soil settlement, that need to be included in the seismic PRA.

Seismic fragility analysis evaluates the fragility or vulnerability of SSCs using plant-specific, SSC-specific information and an accepted engineering method for evaluating the postulated failure. The seismic fragility of an SSC is defined as the conditional probability of its failure at a given value of a seismic motion parameter (e.g., peak ground acceleration). Fragilities should be realistic and plant specific based on actual conditions of the SSCs in the plant and confirmed through a detailed walkdown when possible. Fragilities are determined for SSCs identified in the plant system model but SSCs with high seismic capacities can be excluded from detailed analysis. The seismic-fragility calculations are based on plant-specific data that is supplemented as needed by earthquake experience data, fragility test data, and generic qualification test data.

Generic data can be used in the estimation of SSCs fragilities in the early stages of the PRA. As the reactor design and operational conditions develop, the fragilities should be updated to represent the plant-specific design and conditions. The high level attributes for a seismic fragility analysis are shown in Table F-18.

Table F-18 Seismic fragility analysis attributes.

Item	Attribute
SF-1	Develop realistic fragility estimates for all SSCs identified in the seismic systems analysis.
SF-2	Define and justify the criteria for screening of high seismic capacity SSCs, if screening is performed.
SF-3	Seismic fragilities are generated for relevant failure modes of structures, equipment, and soil (e.g., structural failure, equipment anchorage failure, soil liquefaction).
SF-4	The seismic fragility analysis incorporates the findings of a detailed walkdown focusing on anchorage, lateral seismic support, and potential interactions is performed.
SF-5	Base calculations of seismic-fragility parameters on plant-specific data, supplemented as needed by earthquake experience data, fragility test data, and generic qualification test data.

Seismic systems analysis and quantification involves the integration of seismic hazard frequencies, seismic fragilities, and random equipment failures to quantify the seismic-related risk during all POSs. The internal-events PRA models are used as the framework to perform the quantification and are modified to incorporate seismic-induced failures. The systems analysis includes identification of the types of plant transients induced by the earthquake, inclusion of seismically-induced component failures (including relay chatter) and structure failures, seismic-related dependent failures, the potential for seismic-induced fires or internal floods, and the impact of the earthquake on human errors. Random component failures are retained in the models such that all combinations of random and seismically-induced failures are identified in the model quantification. POS-specific system alignments are also accounted for in the seismic system model. All SSCs identified in the systems and accident sequence used in the seismic-PRA model require a fragility analysis.

The seismic PRA quantification identifies sources of uncertainty and analyzes their impact on the results. The sensitivity of the model results to model boundary conditions and other key assumptions are evaluated using sensitivity analyses to look at key assumptions both individually or in logical combinations. The combinations analyzed are chosen to fully account for interactions among the variables. The high level attributes for a seismic systems analysis are shown in Table F-19.

Table F-19 Seismic systems analysis and quantification attributes.

Item	Attribute
SS-1	Identify the seismic-induced initiating events and other important failures caused by the effects of an earthquake during each POS that can contribute to an undesired end state.
SS-2	Adapt the internal-events PRA model to include seismic-induced failures along with random failures. Account for scenarios during each POS.
SS-3	Include other seismic-related failures such as relay chatter, seismic-induced fires or floods, and structural failure that can contribute significantly to an undesired end-state.
SS-4	Integrate the seismic hazard frequencies and the seismic fragilities into the plant system model.
SS-5	Quantify the seismic scenarios to obtain the desired risk metrics in accordance with the attributes identified for the internal event PRA accident sequence quantification but accounting for the combined effects of failures caused by the earthquake and by random equipment failures or unavailability due to test or maintenance.
SS-6	Modify human recovery failure events to account for seismic-related impacts and include any seismic-specific recovery action.
SS-7	Identify significant contributors (including assumptions, initiating events, POSs, accident sequences, component failures, and human errors) to the required end-states and ensure that all significant sequences are traceable and reproducible.
SS-8	Characterize and quantify the uncertainties in the results including parameter and model uncertainty (using sensitivity analysis) and the contribution from assumptions. Understand their potential impact on the results.

F.3.5 Risk Assessment of Other External Events

The potential for external events other than earthquakes (e.g., high winds, hurricanes, aircraft impacts, and external flooding) occurring at a plant is reviewed and those that are important included in the plant PRA. The external event PRA includes consideration of random failures and the impact of the external events on SSCs and on operator actions. As is the case for internal initiators, external events are evaluated for all modes of plant operation.

An important aspect of external event analysis is the determination of whether failure of equipment in one or more locations caused by the external event can result in radioactive material release. The evaluation of these types of initiators provides critical information on the adequacy of the spatial separation and redundancy of equipment necessary to prevent and mitigate these initiators.

Screening and bounding analysis identifies external events other than earthquakes that may challenge plant operations and require successful mitigation by plant equipment and personnel. A screening process can be used to identify external events that can be excluded from further consideration in the PRA analysis. The screening process considers all sizes or intensities of specific external events (e.g., impacts from both large and small aircrafts). Two examples of screening criteria are: (1) the plant meets the design criteria for the external event, or (2) it can be shown using an analysis that the mean value of the design-basis hazard used in the plant design is less than 10^{-7}/year. If an external event that cannot be qualitatively screened out using acceptable criteria, then a demonstrably conservative or bounding analysis, when used with quantitative screening criteria, can provide a defensible basis for screening the external event from the requirement for a detailed analysis. External events that can not be screened out are subjected to detailed analysis. The bounding and detailed analysis must consider the occurrence of external events during all modes of operation.

Several current US LWRs sites may be submitted for possible future reactor sites. Existing sites will have very similar external events to consider but the results of the external events on the future reactors must be evaluated independently from the LWR on the site. The consequences the external event has on the future reactor may be different from the LWR and the systems in the future reactor will have different capabilities. Specifically, the impact of the external event on passive systems used in future reactors will have to be considered when performing the screening and bounding analysis. External events that threaten the integrity of the passive system or reduce the passive systems' mitigation capabilities need to be identified. The high level attributes for performing an external event screening and bounding analysis are shown in Table F-20.

Table F-20 External event screening and bounding analysis attributes.

Item	Attribute
SB-1	Identify credible external events (including natural hazards and man-made events) that may affect the plant. Consider a credible range of intensities or sizes of events where applicable.
SB-2	Define and justify the screening criteria used to eliminate external events from the scope of the PRA. Apply the screening criteria based on the plant's design and licensing basis relevant to the external event.
SB-3	Perform bounding evaluations of external events during all POSs, if required for comparison to quantitative screening criteria.
SB-4	Perform walkdowns of the plant and surrounding site to confirm the basis for screening of any external event.

Hazard analysis estimates the frequency of occurrence of different sizes or intensities of external events (e.g., hurricanes with various maximum wind speeds) at the site. The hazard analysis can be based on site-specific probabilistic evaluations reflecting recent site-specific data. It may be performed by developing a phenomenolgical model of the event with parameter values estimated from available data or expert opinion, by extrapolating historical data, or a mixture of the two. Since there may be large uncertainties in the parameters and mathematical model of the hazard, it is important the hazard characterization addresses both aleatory and epistemic uncertainties. This is generally accomplished by representing the output of the hazard analysis as a family of hazard curves that reflect the exceedence frequency for different hazard intensities. The hazard analysis can be used in the screening and bounding analysis described previously. The high level attributes for an external event hazard analysis are shown in Table F-21.

Table F-21 External event hazard analysis attributes.

Item	Attribute
HA-1	Characterize the range of intensities for each unscreened external event.
HA-2	Base the frequencies of external events at the site on a site-specific and plant-specific hazard analysis.
HA-3	Use up-to-date databases, site information, and historical information.
HA-4	Address both aleatory and epistemic uncertainties in the analysis to obtain a family of hazard curves.

Fragility analysis determines the conditional probability of failure of SSCs given a specific intensity of an external event. For significant contributors (i.e., SSCs whose failure may lead to unacceptable damage to the plant given occurrence of an external event), a realistic and plant-specific fragility analysis is performed using accepted engineering methods and data for evaluating postulated failures. In the absence of plant-specific data, the use of experience data, fragility test data, generic qualification test data, and expert opinion can be used with thorough and defensible justification. The fragility analysis is based on extensive plant walkdowns reflecting as-built, as-operated conditions. Since there may be large uncertainties in the material properties, understanding of SSC failure modes, use of approximations in modeling, it is important the fragility analysis reflect both aleatory and epistemic uncertainties. This is generally accomplished by representing the output of the fragility analysis as a family of fragility curves with each curve reflecting the conditional probability of failure for different hazard intensities. The high level attributes for an external event fragility analysis are shown in Table F-22.

Table F-22 External event fragility analysis attributes.

Item	Attribute
FA-1	Base the conditional probability of SSC failures from a specific external event on a site-specific and plant-specific hazard analysis.
FA-2	Base calculations of fragility parameters on plant-specific data, supplemented as needed by experience data, fragility test data, and generic qualification test data.
FA-3	Conduct walkdowns when possible to identify plant-unique conditions, failure modes, and as-built conditions.
FA-4	Address both aleatory and epistemic uncertainties in the analysis to obtain a family of fragility curves.

External event systems analysis and quantification assesses the accident sequences initiated by the external event that can lead to an undesired end-state during all modes of operation. The system model is generally adapted from the internal events PRA models and includes external-event-induced SSC failures, non-external-event-induced failures (random failures), and human errors. When necessary, human error data is modified to reflect unique circumstances related to the external event under consideration. The system analysis is well coordinated with the fragility analysis and is based on plant walkdowns and the plant design. The results of the external event hazard analysis, fragility analysis, and system models are assembled to estimate frequencies of the required end-state.

An important aspect in understanding the PRA results is understanding the associated uncertainties. Uncertainties in each step are propagated through the process and displayed in the final results. The quantification process is capable of conducting necessary sensitivity analyses and identifying significant sequences and contributors. The high level attributes for an external event systems analysis are shown in Table F-23.

Table F-23 External events systems analysis and quantification attributes.

Item	Attribute
SQ-1	Identify the initiating events and other important failures caused by the effects of the external event that can contribute to an undesired end state during all POSs.
SQ-2	Adapt the internal-events PRA model to include failures that can be caused by the external event along with random failures. Include any unique common cause failures including correlated and dependent failures and any unique alignments during different POSs.
SQ-3	Include other external event-related failures and failure modes such as loss-of-offsite power, induced fires or floods, and structural failure that can contribute significantly to an undesired end-state.
SQ-4	Integrate the external event hazard frequencies and the SSC fragilities into the plant system model.
SQ-5	Quantify the external event scenarios to obtain the desired risk metrics in accordance with the attributes identified for the internal event PRA accident sequence quantification but accounting for the combined effects of failures caused by the external event and by random equipment failures or unavailability due to test or maintenance.
SQ-6	Modify human recovery failure events to account for external event-related impacts and include any recovery actions specific to the external event.
SQ-7	Identify significant contributors (including assumptions, initiating events, POSs, accident sequences, component failures, and human errors) to the required end-states and ensure that all significant sequences are traceable and reproducible.
SQ-8	Characterize and quantify the uncertainties in the results including parameter and model uncertainty (using sensitivity studies) and the contribution from assumptions. Understand their potential impact on the results.

F.4 Release Analysis Technical Elements

The high level attributes for the Release Analysis portion of the PRA are discussed in this section. The Release Analysis evaluates the physical processes of an accident and the corresponding response of the confinement barriers (including the controlled leakage barrier that provides a fission product containment functional capability), and the subsequent transport of the material to the environment. The end point of Release Analysis is an estimation of the inventory of radioactive material released to the environment, the timing of the release, and the associated probabilities. As a result, accident sequences identified in the Accident Sequence Development portion of the

PRA can be categorized with regard to their frequency, severity, and time of release. A Release Analysis is performed for accident sequences involving any source of radioactive material initiated by internal and external events during all modes of operation.

Accident progression analysis evaluates the type and severity of challenges to the integrity of available barriers (e.g., the vessel and the controlled leakage barrier that provides a fission product containment functional capability) that may arise during postulated accident sequences. The capacity of the available confinement barriers to withstand these challenges is also characterized. A probabilistic framework is used to integrate the two assessments and integrated to generate an estimate of the conditional probability of barrier failure or bypass for accident sequences that result in radioactive material release. In addition, a characterization of the size, timing, and location of the release is determined for input into evaluation of the resulting source term.

The accident progression analysis includes the dependence of the barrier responses on the accident sequence. The barrier response may be included as an integral part of the accident sequence development portion of the PRA. Alternatively, important characteristics for each accident sequence such as the availability of SSCs can be carried forth from the accident sequence development portion of the PRA to a separate accident progression analysis. Any characteristic of the plant response to a given initiating event that would influence either the subsequent barrier response or the resulting radionuclide source term to the environment are identified. Some characteristics of interest related to the reactor core would be; the status of coolant injection systems, the status of heat removal systems, the recoverability of failed systems after an undesirable end-state, and the interdependence of various systems. Grouping of accident sequences with similar behavior can be performed to reduce the amount of analysis required in the accident progression phase of the PRA. The accident progression analysis also models the effects accident phenomena (e.g., high temperatures or pressure) has on the available plant systems and human actions necessary to prevent containment failure or bypass. In addition, the effects of the internal and external accident initiators on these systems and human actions and the potential for additional random system failures are also included in the analysis.

The physical processes involved in accident progression must be identified and understood. For accidents involving the reactor core, this involves both in-vessel and confinement/containment processes that can result in failure of those physical barriers. New accident phenomena different from those identified for LWRs are likely for new reactor designs. Typically, the accident phenomena have been modeled in integral accident analysis codes which are then used to evaluate the progression of the accident. The code calculations can provide a basis for estimating the timing of major accident phenomena and for characterizing a range of potential barrier loads. Since some of the accident phenomena may not be included in an integral code, additional sources of information including engineering analyses of particular issues, experimental data, and expert judgement are often utilized to support the code calculations. Furthermore, since integral accident analysis codes are not always validated in some areas, the codes cannot be used without a clear understanding of the limitations of the models and a thorough understanding of the physical processes involved in the accident progression. Sensitivity studies are required to determine the importance of assumptions made in the accident progression analysis.

The manner and location of confinement/containment failure can be very important in determining the potential consequences from an accident involving the reactor core. Challenges to a confinement/containment can take many forms including increases in internal pressure, high temperatures, erosion of concrete structures, shock waves, and internally generated missiles. New containment failure modes may be possible in new reactor designs. A structured process is utilized to identify the potential confinement/containment (and other barrier) failure modes for the accident sequences of concern. Containment analysis computer codes are often used to determine containment capacities for specific challenges based on established failure criteria.

The timing of major accident phenomena and the subsequent loadings produced on the barriers are evaluated against the capacity of the barriers to withstand the identified challenges. A probabilistic framework is used to combine the two pieces to determine the probability of barrier failure. The potential for subsequent system failures in addition to failures occurring in the earlier phase of the accident are included in the probabilistic assessment. The framework (generally an event tree) allows for modeling dependencies between different accident phenomena, the timing of the phenomena, and most importantly, provides a means to propagate uncertainty distributions for the accident phenomena and barrier response. The high level attributes for an accident progression analysis are shown in Table F-24.

Table F-24 Accident progression analysis attributes.

Item	Attribute
AP-1	For each accident sequence, identify important attributes that can influence the accident progression, barrier (e.g.,confinement/containment) response, and subsequent radionuclide release. Include the impact of accident initiators and unique alignments during different POSs on confinement/containment and other barrier systems that are not modeled in the Accident Sequence Development portion of the PRA.
AP-2	For each accident sequence, identify accident phenomena that can adversely affect accident mitigating systems and operator actions, and challenge barrier integrity.
AP-3	Use verified and validated accident analysis codes to evaluate the progression of the accident. Supplement the code calculations with engineering analyses of particular issues, experimental data, and expert judgement as required.
AP-4	Use verified and validated codes to evaluate the vessel, confinement/containment, and other barrier capacity to withstand the challenges introduced by accident phenomena. This requires identification of the barrier failure criteria.
AP-5	Use a probabilistic framework to assess vessel, confinement/containment, and other barrier system performance. Include the potential for subsequent system failures in addition to failures occurring in the earlier phase of the accident.
AP-6	Estimate the probability of barrier failure. Provide a characterization of the size, timing, and location of the release for input into evaluation of the resulting source term.
AP-7	Characterize and quantify the uncertainties in the results including parameter and model uncertainty (using sensitivity studies) and the contribution from assumptions. Understand their potential impact on the results.

For existing LWRs, the accident progression analysis was for accidents resulting in severe core damage. For new reactors PRAs that are used in the licensing process, the accident progression analysis will have address not only severe accidents, but also LBEs. The release mechanisms for many LBEs will be due to confinement/containment bypass caused by random system failures or failures resulting directly from the accident initiator (e.g., a seismic-induced failure). The evaluation of many LBEs will thus not require as detailed accident progression evaluation as is performed for severe accidents.

Source term analysis provides a quantitative characterization of the radiological release to the environment resulting from each accident sequence leading to barrier failure or bypass. The characterization includes the time, elevation, and energy of the release and the amount, form, and size of the radioactive material released to the environment. The source term characterization must be sufficient for determining offsite consequences. The high level attributes for a source term analysis are shown in Table F-25.

Table F-25 Source term analysis attributes.

Item	Attribute
ST-1	Use verified and validated computer codes to calculate the source terms from specific accidents of concern. The codes must be capable of modeling important radionuclide release, transportation, and deposition phenomena.
ST-2	Reflect plant-specific features of the system design and operation in the calculations. Include impacts resulting from system alignments during different POSs.
ST-3	Include accident sequence specific characteristics in the calculations that affect the timing, form and magnitude of radioactive material released from the fuel, coolant, and confinement.
ST-4	Characterize the source term with respect to the time, elevation, and energy of the release and the amount, form, and size of the radioactive material released to the environment.
ST-5	Characterize and quantify the uncertainties in the results including parameter and model uncertainty (using sensitivity studies) and the contribution from assumptions. Understand their potential impact on the results.

Deterministic computer code calculations that reflect plant-specific features of system design and operation are used to model the radionuclide release, transportation, and deposition phenomena in the reactor (or other locations of radioactive material) and confinement/containment. The computer codes should be verified to cover the range of conditions included in the calculations. For accident sequences involving the reactor core specific characteristics affecting the timing, form and magnitude of radioactive material released from the fuel and coolant are also accounted for in the computer evaluations. Examples of these characteristics include the reactor vessel pressure at the time of the release and the availability of containment spray systems to reduce the source term. Uncertainties related to radionuclide behavior under accident conditions exists and must be considered in order to characterize uncertainties in the radionuclide source term associated with individual accident sequences.

The source term analysis must provide sufficient information on the radionuclide release to completely define the input to the consequence assessment codes used for calculating health and economic consequences. The number of consequence assessments can be reduced by combining accident sequences resulting in similar source terms into release categories. Characteristics of accident progression and containment performance that have a controlling influence on the magnitude and timing of radionuclide release to the environment can be used to group sequences with similar source terms into appropriate release categories.

F.5 Consequence Assessment Technical Elements

The high level attributes for Consequence Assessment portion of the PRA are described in this section. The Consequence Assessment evaluates the consequences of an accidental release of radioactivity to the public and the environment. A PRA that includes a Consequence Assessment is needed to compare the determined numerical values for the frequency and consequence of accidents with the QHOs and the Frequency-Consequence curve provided in Chapter 6. To accomplish this, the Consequence Assessment is performed for accident sequences involving any source of radioactivity, initiated by internal and external events during all modes of operation.

Consequence analysis evaluates the offsite consequences of an accidental release of radioactive material from a nuclear power plant expressed in terms of human health, environmental, and economic measures. The consequence measures of most interest focus on impacts on human health. Specific measures of accident consequences developed in a PRA can include: the number of early fatalities, the number of early injuries, the number of latent cancer fatalities, population dose at various distances from the plant, individual dose at various distances from the plant, individual early fatality risk defined in the early fatality QHO, individual latent cancer risk defined in the latent cancer QHO, and land contamination. The last three are of primary interest in the proposed Framework for licensing new reactors.

A probabilistic consequence assessment code is used for estimating the consequences of postulated radiological material releases. The code calculations typically require information on the local meteorology including wind speed, atmospheric stability, and precipitation. Information is also required on demographics, land use, property values, and other information concerning the area surrounding the site. The consequence code typically require the analyst to make assumptions on the value of parameters related to the implementation of protective actions following an accident. Examples of these assumptions include:

- the (site-specific) time needed to warn the public and initiate the emergency response action (e.g., evacuation or sheltering),
- the effective evacuation speed,
- the fraction of the offsite population which effectively participates in the emergency response action,
- the degree of radiation shielding afforded by the building stock in the area,
- the projected dose limits assumed to trigger normal and hot spot relocation during the early phase of the accident,
- the projected dose limits for long-term relocation from contaminated land, and
- the projected ingestion doses used to interdict contaminated farmland.

The values or assumed values for the above parameters have a significant impact on the consequence calculations and need to be justified and documented. In particular, the influence of the accident initiator (particularly external events such as earthquakes) needs to be addressed. In addition, for PRAs performed as part of the design certification process for new reactor designs, the lack of a specific site for the plant requires that some assumptions be made in order to perform the consequence assessment. These assumptions need to be realistic and well documented.

The high level attributes for a consequence analysis are shown in Table F-26.

Table F-26 Consequence analysis attributes.

Item	Attribute
OC-1	Identify the offsite human health, economic, and environmental consequence measures required following a release of radioactive material.
OC-2	Use a probabilistic consequence assessment code to estimate the required consequences using site-specific meteorology information, evacuation and sheltering plans, population data, and other required data and assumptions.
OC-3	Justify and document all parameter values and assumed parameter values.
OC-4	Ensure that the consequence code has been validated and verified.
OC-5	Characterize and quantify the uncertainties in the results including parameter and model uncertainty (using sensitivity studies) and the contribution from assumptions. Understand their potential impact on the results.

Health and economic risk estimation is the final step in a PRA that proceeds all the way to a Consequence Assessment. It integrates both the frequency and consequence results for accident sequences to compute the selected measures of risk. The high level attributes for an external event systems analysis are shown in Table F-27.

Table F-27 Health and economic risk estimation attributes.

Item	Attribute
HE-1	Identify the risk measures required from the output of the PRA.
HE-2	Merge the results from the different elements of the PRA in a self-consistent and statistically rigorous manner to obtain the required risk measures.

The severe accident progression and the fission product source term analyses conducted in the Release Analysis portion of the PRA and the consequence analysis conducted in the Consequence Assessment part of the PRA are performed on a conditional basis. That is, the evaluations of alternative severe accident progressions, resulting source terms, and consequences are performed without regard to the absolute or relative frequency of the postulated accidents. The final computation of risk is the process by which each of these portions of the PRA are linked together in a self-consistent and statistically rigorous manner. The important attribute by which the rigor of the process is judged is the ability to demonstrate traceability from a specific accident sequence through the relative likelihood of alternative accident progressions and measures of barrier performance and ultimately to the distribution of fission product source terms and accident consequences.

An important aspect in understanding the PRA results is understanding the associated uncertainties. Uncertainties in each step of the PRA are propagated through the process and displayed in the final results. The quantification process is capable of conducting necessary sensitivity analyses and identifying significant sequences and contributors.

F.6 References

[NRC 2003] Atwood, C.L., et al. 2003. “Handbook of Parameter Estimation for Probabilistic Risk Assessment.” NUREG/CR-6823, U.S. Nuclear Regulatory Commission, September 2003.

APPENDIX G
SELECTION OF TOPICS FOR POTENTIAL REQUIREMENTS

G. SELECTION OF TOPICS FOR POTENTIAL REQUIREMENTS

G.1 Introduction

In Chapter 8, the general process for the identification of topics for which risk-informed and performance-based technical and administrative requirements should be provided was discussed. The purpose of this appendix is to apply this process to each of the five protective strategies described in Chapter 5 and to the administrative area. Section G.2 below describes the application of the process to the five protective strategies and Section G.3 describes its application to the administrative area.

G.2 Identification of Technical Topics for the Protective Strategies

Chapter 5 discussed a structure involving protective strategies whereby each protective strategy represents an important element of safety that, if accomplished, will ensure the design, construction and operation of the nuclear power plant (NPP) results in achieving the overall safety objectives. The protective strategies discussed in Chapter 5 are:

- physical protection,
- stable operation,
- protective systems,
- barrier integrity, and
- protective actions.

The protective strategies represent a high level defense-in-depth structure for developing requirements, in that each one represents a line of defense against the uncontrolled release of radioactive material and/or the adverse impact on the health and safety of workers and the public. The identification of the topics for which technical requirements should be provided to ensure the success of each protective strategy is described in Sections G.2.1 through G.2.5 below.

G.2.1 Physical Protection

The physical protection protective strategy ensures that adequate measures (e.g., design, operating practice, and intervention capability) are in place to protect workers and the public against intentional acts (e.g., attack, sabotage) that could compromise the safety of the plant or lead to radiological release. Physical protection is applied to all elements of plant design, including the other protective strategies, and involves both extrinsic protective measures ("guns, guards, and gates") to block access to attackers and intrinsic design features to minimize their possible success should they gain access, as well as provide protection from external attack. Diversion of nuclear material is also included in the scope of this protective strategy. The logic tree in Figure G-1 lays out the possible paths that can lead to failure of the physical protection protective strategy. At the top level, failure of the physical protection protective strategy can occur due to (1) failure of protective measures to perform consistent with assumptions in the security analysis, (2) failure due to improper analysis or implementation of requirements, and (3) failure due to challenges beyond what were considered in the design. Accordingly, the requirements need to address all three of the above pathways integral with safety and preparedness to ensure robust physical protection.

G. Selection of Topics

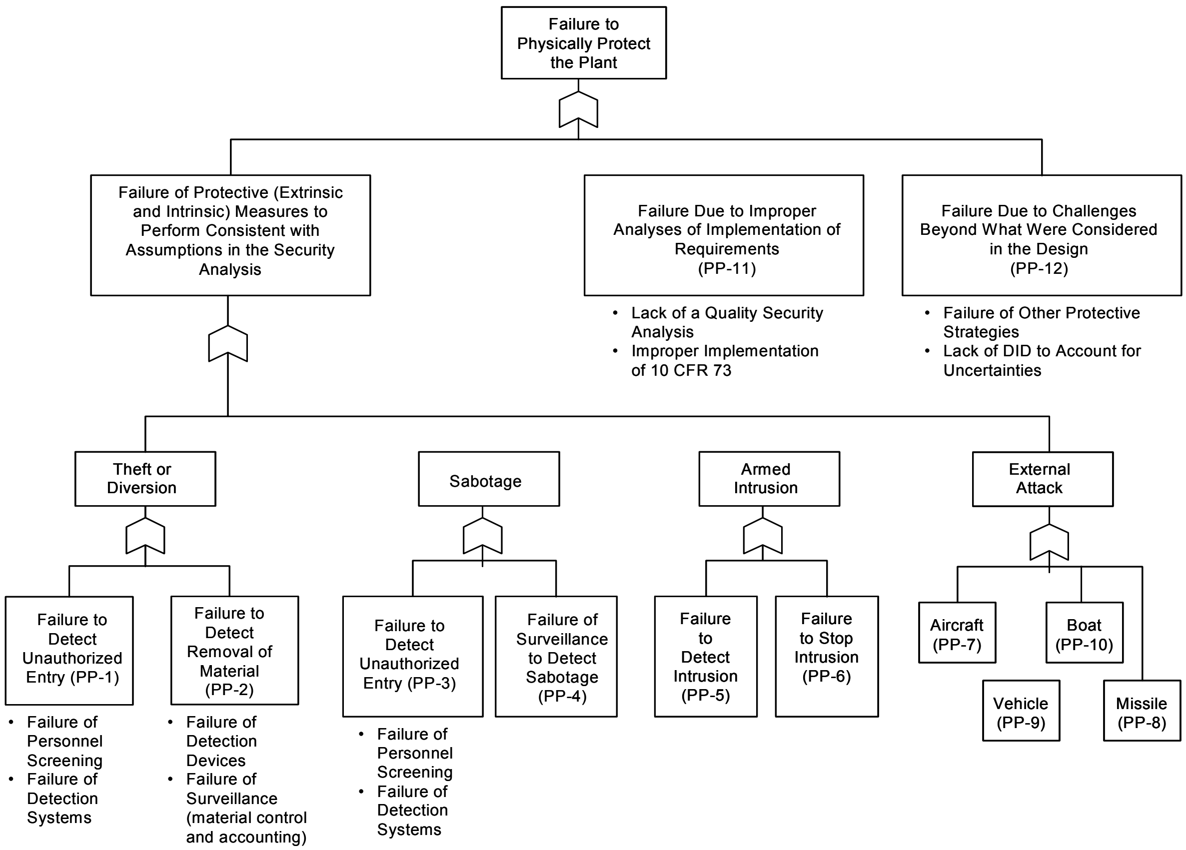

Figure G-1 Logic tree for the physical protection strategy.

There are three major pathways, shown in Figure G-1, that result in topics which the requirements need to address to protect against their failure, that are discussed below.

For the first major pathway (failure of protective measures), the following subjects need to be addressed:

- theft and diversion,
- sabotage,
- armed intrusion, and
- external attack.

For theft/diversion or sabotage to be successful, there would need to be a failure to prevent or a failure to detect an unauthorized entry. Failure to prevent could be caused by failure of the personnel screening process (i.e., a person who works at the plant is the thief or saboteur) and a failure of physical barriers (e.g., doors, locks) to prohibit entry into vital areas. Failure of detection devices, material control and accounting or surveillance to detect sabotage could also lead to failure of this protective strategy. It is recognized that 10 CFR 73, “Physical Protection of Plants and Materials” contains requirements to protect against theft/diversion and sabotage, including checking for personnel trustworthiness and controlling access to plant protected and vital areas. Accordingly, 10 CFR 73 requirements should be applied.

Likewise, 10 CFR 73 contains requirements to address armed intrusion, up to and including the design basis threat (DBT). The 10 CFR 73 requirements address items such as the nature of the guard force, physical barriers and intrusion detection capability. Over time, if the DBT changes, the ability of the plant's physical protection capability to cope with the revised DBT would also need to be assessed.

10 CFR 73 also includes provisions to address certain types of external attacks. These include requirements for vehicle barriers, physical separation and multiple barriers to prevent access to vital equipment. However, not all types of external attacks are addressed in 10 CFR 73, particularly those by aircraft or missile.

For the second major pathway, failure prevention is dependent upon the proper implementation of 10 CFR 73 requirements and correct security analyses. Accordingly, ensuring proper implementation of 10 CFR 73 requirements and quality analyses is essential to the success of this protective strategy. Thus, requirements related to security quality analysis, and the use of validated safety analysis tools are essential.

For the third major pathway (challenges beyond what were considered in the design) protection is provided by the other protective strategies (i.e., they represent additional lines of defense) and by application of the defense-in-depth principles to account for completeness uncertainty, as discussed below.

Applying the defense-in-depth principles to this protective strategy suggests the following topics need to be addressed by the requirements for physical protection:

- Physical protection should address prevention as well as mitigation. Traditional security measures, in conjunction with the other protective strategies, address both. However, to help provide high assurance of protection, all security related events considered in the design should be assessed to ensure that both prevention and mitigation measures are provided for each event considered.

- Physical protection should not be dependent upon a single element of design, construction or operation. The combination of protective measures (personnel screening, access control, barriers, etc.) defined in 10 CFR 73 should provide multiple layers of defense, along with the other protective strategies. However, each security related event considered in the design should be assessed to ensure that protection of public heath and safety is not dependent upon a single piece of plant equipment, system, structure or operator action.

- Physical protection should account for uncertainties and provide appropriate safety margins. Requiring security be considered integral with design, including a safety and security assessment and security performance standards as discussed in Section 6.7, will help address uncertainties and provide safety margin, thus providing high assurance of protection of public health and safety.

- Physical protection should be directed toward preventing an unacceptable release of radioactive material to the environment. In this regard, the security assessment should include an analysis of the release of radioactive material and use the security performance standards discussed in Section 6.7 as the basis for decisions.

- Plant siting should consider the ability to implement protective measures to protect the public.

Table G-1 summarizes the logic tree of Figure G-1 in the form of questions that need to be addressed by the requirements to ensure that the pathways that could lead to failure of the physical protection protective strategy are adequately covered. The answers to the questions then represent the topics which the requirements would need to address. The table is organized by the three top level pathways of the logic tree and the answers to the questions (i.e., topics) are arranged by whether they apply to design, construction, or operation. It should be noted that each question is identified by an alpha-numeric label which is tied back to the logic tree.

Table G-1 Physical protection.

Protective Strategy Questions	Answers to Questions		
	Design	**Construction**	**Operation**
Failure of Protective Measures for Theft/Diversion			
• How should theft and diversion be detected? (PP-1)	• Conduct security assessment integral with design and preparedness, including security performance standards.	• N/A	• Implement results of security assessment, plus 10 CFR 73 requirements. - Provide detection systems and surveillance
• How should unauthorized removal of material be detected? (PP-2)	• Use 10 CFR 73 requirements for detection capability.		• Detection and surveillance (i.e., 10 CFR 73) check for loss (material control and accounting). - personnel screening - detection systems
Failure of Protective Measures for Sabotage			
• How should unauthorized entry be prevented? (PP-3)	• Use 10 CFR 73 requirements plus conduct security assessment integral with design and preparedness, including security performance standards.	• Access Control	• Implement results of security assessment, plus 10 CFR 73 requirements. - verify trustworthiness of personnel (i.e., personnel screening) - detection systems
• How can sabotage be detected? (PP-4)	• N/A	• QA, QC and surveillance to check for sabotage.	• Surveillance to check for sabotage.

Table G-1 Physical protection.

Protective Strategy Questions	Answers to Questions		
	Design	**Construction**	**Operation**
Failure of Protective Measures for Armed Intrusion			
• How can armed intrusion be detected? (PP-5)	• Conduct security assessment integral with design and preparedness, including security performance standards.	• N/A	• Implement results of security assessment, plus 10 CFR 73 requirements.
• How can armed intrusion be stopped? (PP-6)	• 10 CFR 73 requirements.	• N/A	• Use 10 CFR 73 requirements.
Failure of Protective Measures for External Attack			
• How can vital areas be protected from external attacks from: - aircraft (PP-7) - missile (PP-8) - vehicle (PP-9) - boat (PP-10)	• Conduct security assessment integral with design and preparedness (including performance standards) plus use 10 CFR 73 requirements.	• N/A	• Implement results of security assessment plus 10 CFR 73 requirements. • Include in training program.
Failure Due to Improper Analyses or Implementation of Requirements			
• How can failure be prevented due to incorrect implementation of 10 CFR 73 requirements or poor analyses? (PP-11)	• Conduct independent review of security provisions and assessments. • Ensure correct DBT and security analyses using validated analytical tools (e.g., PRA).	• QA/QC	• Conduct independent review of security provisions and assessments. • Update analyses, as necessary, to be current with threat situation.
Challenges Beyond What was Considered in the Design			
• How can challenges beyond what were considered in the design (i.e., uncertainties) be accounted for? (PP-12)	• Other protective strategies and DID principles provide additional protection. • Require a security assessment integral with design and preparedness (including assessment of beyond DBTs and use of security performance standards).	• N/A	• Implement results of security assessment. • Update assessment to be current with threat situation.

N/A = Not applicable

As can be seen in Table G-1, many of the requirements that would be needed to address this protective strategy already exist in 10 CFR 73. The Framework and technology-neutral requirements would not change these requirements (i.e., any future design using the technology neutral requirements would also have to meet 10 CFR 73 requirements). However, for defense-in-depth reasons, Table G-1 does propose that, in addition to 10 CFR 73, future designs also consider physical protection in an integrated fashion as part of the design. Security considerations can affect the design of plant systems, structures and components with respect to their:

- location, separation, orientation or independence
- power supply
- accessability
- vulnerability to external attack
- events to be considered in the safety analysis

It is proposed that designers perform a safety and security assessment on their designs against a range of threats, including beyond the DBT, using a set of security performance standards (as proposed in SECY-05-0120), and discussed in Section 6.7 of the Framework. Security considerations would then be factored into the design.

Accordingly, the requirements would need to include a requirement for applicants to conduct such a safety and security assessment, including assessment against the proposed security performance standards. Chapter 6 (Section 6.7) discusses the proposed security performance standards. These standards are considered a policy issue (as discussed in Appendix C, Section C.2.9) since they represent a fundamental change from traditional security evaluations. Guidance on conducting a safety and security assessment would be provided in a separate document.

It should be noted that the scope of the proposed security assessment and performance standards goes beyond what is required today by requiring a limited set of beyond DBTs and threats from enemies of the U.S. (see 10 CFR 50.13) be considered in the assessment.

G.2.2 Stable Operation

The stable operation protective strategy ensures that design, construction, maintenance and operating practice minimize the inadvertent challenges that could adversely impact plant performance and safety. Events will occur from time to time that cause the plant to deviate from normal conditions. Some of these events are outside the control of the designers of the plant or operating personnel such as weather, loss of offsite power and seismic events. Most, however, are within the control of the designers and the plant operating personnel such as human error, equipment failure and poor design. In any case, the plant needs to be designed for a range of events (i.e., those that are expected to occur one or more times during the life of the plant as well as those that are not expected to occur but, nevertheless, are within the frequency range of events to be considered in the design). However, the risk from plant operation is directly proportional to the number and nature of events that affect stable operation. Therefore, limiting the number and nature of these events as a protective strategy can directly improve safety.

Figure G-2 is a logic tree that shows the various pathways that can affect stable operation. At the top level, stable operation can be affected by (1) failure to design, construct, maintain and operate the plant consistent with the assumptions in the licensing analysis, (2) failure due to improper analyses or implementation of requirements, and (3) failure due to challenges beyond what were considered in the design. Accordingly, the requirements need to address all three of the above major pathways to ensure stable operation.

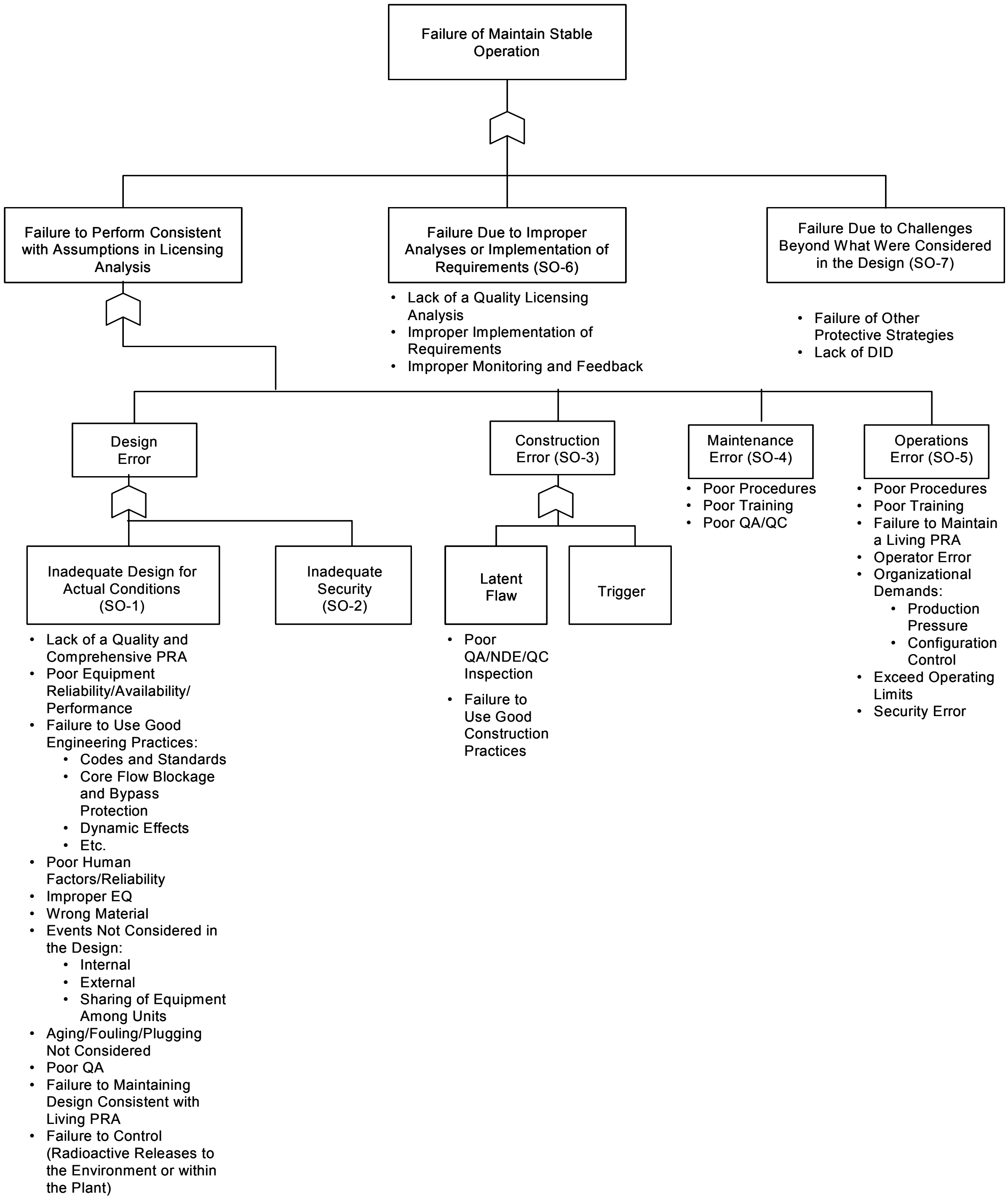

Figure G-2 Logic tree for the stable operation strategy.

There are three major pathways, shown in Figure G-2, that result in topics which the requirements need to address to protect against their failure, that are discussed below.

The first major pathway involves failure to maintain the assumptions in the licensing analysis. One item that can cause assumptions in the licensing analysis to not be maintained is a poor design. Such design errors could result in a design that has failed to include certain events (and, therefore, the design does not address them), wrong assumptions on equipment availability, reliability or performance (e.g., inadequate environmental qualification), design attributes that do not promote minimizing errors (e.g., poor human factors design) or other items the design failed to consider (e.g., plant aging, wrong materials, control of releases of radioactive material, etc.). Thus the use of good engineering practices (e.g., use of accepted consensus design codes and standards, equipment qualification (EQ), adequate fire protection provisions, including for liquid metal and graphite fires, etc.) and quality assurance (QA) in design is important to stable operation. To ensure safety significant SSCs are identified, a safety classification process should be used (see Chapter 6 for discussion). Safety significant SSCs should then receive special treatment to demonstrate their functionality. Another item that can affect stable operation is inadequate security. If protection against security related events is not sufficient, then unanticipated events affecting operation could be the result. The discussion on physical protection (Section G.2.1) provides guidance on protection in this area.

Construction and/or fabrication errors can also cause a failure to maintain assumptions in the licensing analysis. Such errors can leave undetected flaws in structures or equipment that, when triggered by a demand or by additional degradation over time, can lead to a failure that was not assumed in the analyses. Thus, good construction and manufacturing practices are important to stable operation, as well as good QA, quality control (QC), non-destructive evaluation (NDE), inspection, etc.

Maintenance errors can also cause assumptions in the licensing analysis to not be met. Such errors can lead to equipment failures, plant transients or common cause failures. Good procedures, training, QA and QC can help prevent such errors. Much of the current guidance contained in 10 CFR 50, Appendix B can be used for the Framework QA/QC guidance applicable to design, construction, maintenance and operation.

During plant operation, a number of items could lead to events affecting stable operation that are not consistent with what was considered in the licensing analysis. Events can be caused by poor work control, misalignments or poor communication. Events can also be caused by organizations and/or personnel not performing as assumed in the licensing analysis. This could be due to poor training, procedures, personnel errors or organizational influences (e.g., lack of staff or resources). To help protect against these kinds of failures, good operating practices, such as training programs and procedure development that incorporates the use of plant specific simulators to test procedures and train personnel should be used

Finally, operating limits can be exceeded that affect stable operation. Exceeding operating limits can result from a number of factors, including operator error, organizational pressures (e.g., production pressure) or equipment failure. To help protect against these kinds of failures, training programs and procedures should incorporate the use of plant specific simulators to test personnel and procedures.

Failure of the protective strategy can also be caused by improper analysis or implementation of requirements as represented by the second major pathway. The licensing analysis and the predicted plant response to postulated accidents depends upon assumptions related to equipment performance, reliability and availability and proper implementation of requirements. Thus, proper implementation and modeling of requirements (such as the event selection criteria in Chapter 6) and the use of validated analytical tools and QA are essential. Also, the use of monitoring and feed back and technical specifications can help ensure key requirements/limits are implemented and emphasized.

In a risk-informed and performance-based regulatory process, performance monitoring and feedback play an important role. Accordingly, it is important that the equipment and parameters selected for monitoring align closely with the key equipment and assumptions in the licensing analysis. With respect to the probabilistic risk assessment (PRA), the purpose of the monitoring and feedback will be to obtain actual data on equipment reliability, availability and performance to be used in updating the PRA. Such feedback will help confirm PRA data, adjust it to conform with reality and reduce uncertainties. With respect to performance-based requirements, monitoring will be mandatory to comply with the requirements. The frequency of monitoring and feedback will need to be determined so as to achieve its intended purpose.

For challenges beyond what were considered in the design, protection is provided by the other protective strategies (i.e., they are additional lines of defense) and by application of the defense-in-depth principles to account for completeness uncertainty, as discussed below.

Applying the defense-in-depth principles to this protective strategy suggests the following topics be included in the requirements for stable operation:

- Intentional acts to disrupt operation should be considered. Section G.2.1, "Physical Protection," provides guidance on how to prevent and protect against such disruptions.

- Designing the plant to prevent accidents is the main emphasis of the stable operation protective strategy. To ensure that the assumptions in the PRA on the number and nature of initiating events (IEs) are preserved, each applicant should be required to propose cumulative limits on frequency for each of the frequent, infrequent and rare event frequency categories. These would then be used to ensure PRA assumptions regarding initiating event frequencies are maintained over the life of the plant. In addition, considering accident mitigation in the design can also contribute to maintaining stable operation by limiting the effects of disruption so that plant personnel and unaffected equipment can respond to the disruption and limit its affect. Accordingly, plant systems and features directed toward accident mitigation also need to be included in the design. Sections G.2.3, G.2.4, and G.2.5 address such systems and features.

- Event sequences considered in the design that could disrupt stable plant operation should not be of such a nature as to defeat multiple protective strategies simultaneously. Accordingly, events with the potential to defeat multiple protective strategies such that the dose limits specified in the frequency-consequence (F-C) curve (Figure 6-1) would be exceeded, need to be kept to a frequency of less than 10^{-7}/plant year. Such events might include reactor pressure vessel rupture, combustible gas explosion, or energetic recriticality. Reducing the frequency of such events to less than 10^{-7}/plant year will help ensure that no single event can defeat multiple protective strategies sufficient to exceed dose limits.

- Uncertainties should be considered in assessing the frequency of events that could disrupt stable plant operation and appropriate safety margins provided. Accordingly, the licensing analysis needs to quantify uncertainties and the PRA and Licensing Basis Event (LBE) selection process use criteria that provide margin for uncertainties. Such margin is described in Chapters 6 and 7. In addition, the values selected for performance-based limits should be set with sufficient margin from failure such that, if exceeded, there is no immediate safety concern and time is available for corrective action.

- The effect plant siting could have on contributing to the disruption of stable plant operation should be considered in the design consistent with 10 CFR 100. This would include consideration of natural as well as man-made events.

Table G-2 presents a set of questions, based upon the logic tree in Figure G-2, that address the pathways that can affect stable operation. A unique alpha-numeric identifier is assigned to each question to tie it back to the logic tree. The questions focus on what can be done at the design, construction and operating stage to maintain stable operation. The answers to these questions are the topics which the requirements need to address to help ensure stable operation. The topics are arranged according to whether they apply to design, construction or operation. Discussed below are additional considerations related to implementation of the items discussed above.

Table G-2 Stable operation.

Protective Strategy Questions	Answers to Questions		
	Design	Construction	Operation
Failure to Maintain Assumptions - Design Error			
• What needs to be done to ensure the design is adequate for the expected actual conditions? (SO-1)	• use event and LBE selection criteria in Chapter 6	• N/A	• monitoring and feedback into the design
	• follow siting requirements (10 CFR 100) and consider effect of site specific events	• N/A	
	• ensure proper scope and quality of licensing analysis and consideration of uncertainties	• N/A	
	• use of good engineering practices: - use of consensus design codes and standards - good human factors design (e.g., automatic vs. operator action) - I&C qualification-software V&V - QA	• QA/QC • Testing • Inspection	• maintenance • training • procedures • ISI • IST • staffing

Table G-2 Stable operation.

Protective Strategy Questions	Answers to Questions		
	Design	Construction	Operation
	- proper EQ - core flow blockage and bypass prevention - reactor inherent protection (e.g., no positive power coefficient) - assess dynamic effects - consider effects of sharing of equipment among units - qualified materials - qualified safety analysis tools		
	• Properly designed electric power systems if needed for safety	• N/A	
	• Use of prototype testing	• N/A	
	• Research and Development	• N/A	
	• safety classification (see Chapter 6)	• N/A	
	• fire protection	• N/A	• use of appropriate fire fighting materials
	• prevention of brittle fracture	• N/A	
	• leak before break	• N/A	• aging management program
	• consider plant aging, corrosion, etc. in the design	• N/A	
	• failure to control radioactive releases to the environment or within the plant	• N/A	• maintain intermediate heat transfer loop at a higher pressure than RCS
	• specify reliability goals consistent with PRA - reliability assurance program - specify goals on initiating event frequency	• N/A	• monitoring and feedback
	• maintain design consistent with PRA	• N/A	• monitoring and feedback

Table G-2 Stable operation.

Protective Strategy Questions	Answers to Questions		
	Design	**Construction**	**Operation**
• What needs to be done to provide adequate security? (SO-2)	• see physical protection protective strategy	• see physical protection protective strategy	• see physical protection protective strategy
Failure to Maintain Assumptions - Construction Error			
• What needs to be done to prevent construction or manufacturing flaws? (SO-3)	• Specify construction/ manufacturing methods to be used.	• Use of good construction/ manufacturing practices, including attention to factory fabrication and fabrication outside the U.S. • QA/QC • NDE • Inspection	• Surveillance • ISI • Testing
Failure to Maintain Assumptions - Maintenance Errors			
• What needs to be done to prevent maintenance errors? (SO-4)	• N/A	• N/A	• procedures • maintenance training • maintenance QA/QC
Failure to Maintain Assumptions - Operation Error			
• What needs to be done to limit operational errors? (SO-5)	• Consider human factors and man-machine interface as part of the design.	• N/A	• Utilize good operating practices: - training - procedures - maintenance - configuration and work control - use of simulators • technical specifications • security • personnel qualification • maintain PRA
Failure Due to Improper Analyses or Implementation of Requirements			
• How can failures due to improper analyses or implementation of requirements be prevented? (SO-6)	• Ensure quality analysis and that plant is designed consistent with licensing analysis, including event selection criteria in Chapter 6.	• Ensure plant is constructed consistent with design.	• Ensure plant is maintained and operated consistent with licensing analysis.

Table G-2 Stable operation.

Protective Strategy Questions	Answers to Questions		
	Design	**Construction**	**Operation**
	• QA	• QA/QC	• Ensure fuel and replacement part quality is maintained over the life of the plant
			• Monitoring and feedback • Technical specifications
Challenges Beyond What were Considered in the Design			
• How can challenges beyond what were considered in the design (i.e., uncertainties) be accounted for? (SO-7)	• Apply other protective strategies and DID principles.	• N/A	• Surveillance • Monitoring and feedback
	• Frequency of events that could simultaneously defeat the protective systems, barrier integrity and protective actions strategies should be kept below 10^{-7} per plant year. • Consideration of uncertainties in PRA, LBE section and setting performance limits.	• N/A • N/A	• N/A • N/A

N/A = Not Applicable

G.2.2.1 Design Stage

At the design stage the key topics that should be covered in the requirements are related to (1) ensuring that the analysis that supports the plant design and safety is as complete as possible, is based upon accepted methods and data applicable to the design and quantifies uncertainties and (2) using good engineering practices in the design to help ensure high reliability/availability of equipment and promote good man-machine interface. Good engineering practices can generally be considered to include items such as the use of accepted codes, standards and practices; QA and QC; EQ; qualified materials and analytical tools and other items that promote good design.

Other important considerations for new plants are ensuring that the reliability and availability of equipment is consistent with assumptions in the licensing analysis (i.e., reliability assurance and special treatment), siting, the need for research and development and how to use the results of prototype testing to support licensing. Each of these is discussed below.

Use of Validated Analysis Tools

The licensing analysis, including the PRA, is only as good as the analysis tools and the qualifications of those that use them. These analysis tools would be used to address items such as:

- thermal-hydraulic analysis
- reactor physics analysis
- seismic response analysis
- structural (e.g., containment) analysis
- systems analysis
- accident (e.g., core damage, source term release and transport) analysis

Accordingly, the analytical tools used for such analysis need to be qualified for the conditions (e.g., temperature), materials and phenomena expected during normal operation and in the plant response to accidents.

Where consensus codes and standards are used for design, there is confidence that the methods and material properties specified for use are qualified for and adequately address the phenomena and range of conditions expected. Where consensus codes and standards are not available, it will be necessary for the applicant to demonstrate that the analytical tools proposed for use have been validated against actual operational or experimental data, including the range of conditions and phenomena expected. Operational and experimental data at different scales and different conditions is generally used to validate thermal-hydraulic and accident analysis tools. For other analytical tools, the applicant would need to demonstrate the validation process. In any case, the validation process and results would likely need to be submitted to NRC for review.

Reliability Assurance Program

For all safety significant equipment (as determined by the safety classification process described in Chapter 6) which is first of a kind equipment, or equipment with little operating experience under the planned conditions, the applicant should have a reliability assurance program to demonstrate the reliability, availability and performance assumed in the licensing analyses. Such a reliability assurance program should include sufficient research and development, EQ, testing and analysis to demonstrate that the equipment will perform as assumed. At the operating stage, the program should also call for the monitoring of equipment performance, reliability and availability for consistency with the licensing analysis over the life of the plant, including feedback into the licensing analysis. To help mitigate the effects of aging on SSC performance, reliability or availability, an aging management program should also be developed in conjunction with the design and implemented over the life of the plant.

Special Treatment

SSCs that are identified as safety significant (using the safety classification process described in Chapter 6) should receive special treatment to demonstrate they perform under the conditions in which they are expected to operate. Special treatment can be different, depending upon the SSC and the conditions under which it needs to perform its safety functions. Special treatment generally consists of one or more of the following items: QA/QC, EQ (for temperature, humidity, radiation, etc.), and Seismic qualification.

For safety significant first of a kind equipment or equipment being used under new service conditions, the reliability assurance program described above can provide the special treatment. For other safety significant SSCs, the special treatment needed can be technology and design specific. The PRA can be a useful tool for identifying under what conditions the SSCs are to function and thus identifying what special treatment is needed.

Siting

The relationship between the risk-informed and performance-based approach described in the Framework and 10 CFR 100, “Siting Requirements of Nuclear Power Plants” is intended to be one where the requirements of 10 CFR 100, Subpart B, would apply and the risk-informed and performance-based requirements would contain the dose criteria. These dose criteria would be the same as are currently contained in 10 CFR 50.34(a)(1)(ii)(D), (i.e., the “worst” 2-hour dose and the dose at the outer edge of the Low Population Zone for the duration of the accident are less than 25 rem TEDE). The dose calculation would be based upon the deterministic accident (discussed in Section G.2.4) selected to meet defense-in-depth principle #5, which requires a controlled leakage barrier, independent from the fuel and RCS, with a capability to limit releases of radioactive material to the environment to acceptable levels. As discussed in Section G.2.4, the deterministic accident would be selected to address potentially larger source terms than considered in the PRA and would be analyzed mechanistically. However, it needs to be recognized that the Framework also calls for a range of low probability accidents (rare event category) to be analyzed and meet the doses represented by the F-C curve which are to be analyzed for the same duration and distance as is used in the siting dose calculation. Accordingly, design acceptability (and indirectly site acceptability) includes consideration of accidents beyond what has traditionally been considered in the licensing basis.

Research and Development

Applicants are responsible for performing sufficient research and development to validate analytical assumptions and tools. Such research and development may consist of separate effects and/or integral system tests and may be conducted in full scale or partial scale facilities. In general, the requirements should specify that research and development would be expected on key plant safety features when these features are new (i.e., not previously licensed) or are to be used under conditions which go beyond previous use or experience. The scope of research and development should be sufficient to verify performance of the features over the range of conditions for which they are expected to function, including the effects of fuel burnup and plant aging. Examples of the types of research and development which might be expected are:

- fuel performance testing (steady state and transient)
- passive decay heat removal system testing
- NDE methods testing
- reactor shutdown system testing
- materials testing.

Applicants should propose the research and development necessary to support the licensing of their designs.

Use of Prototype Testing

New plants may also propose the use of a demonstration plant, in lieu of conducting extensive research and development. In this case, the demonstration plant would be used to demonstrate the safety of the design in lieu of a series of separate research and development efforts. If such an approach is to be accepted, the applicant would need to address:

- What would be the objective of the test program:
  - Which aspects of plant safety can be addressed by demonstration plant testing?
  - Which types of analytical tools could be validated?
  - What phenomena could be addressed?

- What would be the scope of the test program:
  - How would the test program be selected?
  - Would it be conducted during initial startup only?
  - How would plant aging, irradiation, burnup effects be tested?
  - Would tests cover the full range of the accidents or only partial ranges, with the remainder done by analysis?
  - What instrumentation would be required?

- Are any special provisions needed in case the tests do not go as planned (e.g., robust containment, EP, has to be on a remote site, DOE site, etc.)?

- How would equipment reliability assumptions be verified?

- What acceptance criteria would be necessary (e.g., scope, treatment of uncertainties)?

- Would there be any limitations on future design changes?

- If the initial demonstration plant is to be licensed, how would this be accomplished?

Also, documentation describing the test program and the test program results needs to be specified.

G.2.2.2 Construction Stage

At the construction stage, good construction practices will help ensure the plant is built as intended. Accordingly, each of the topics identified for construction is directed toward ensuring the application of good construction practices so that the plant is built as intended. Many regulatory requirements related to the construction of new plants are expected to be similar in many ways to those employed in the past (e.g., QA, QC, inspection). Where existing requirements are applicable, they will be incorporated into the new licensing structure. It is expected that NRC's role in construction will be similar to that employed previously involving QA/QC and on-site inspections. A framework regarding such inspections is contained in NUREG-1789[9], “10 CFR Part 52 Construction Inspection Program Framework Document” and should be used as guidance in preparing construction/inspection requirements. In addition, the PRA will provide insights regarding the

[9] NUREG-1789: “10 CFR Part 52 Construction Inspection Program Framework Document”

importance of various plant features and can be used to help identify items for inspection. The construction of new plants, however, is expected to rely more on the following:

- factory fabrication to produce modules that can be installed in the field, thus reducing the amount of field fabrication;

- utilize components fabricated outside the U.S. and possibly to non-U.S. codes and standards; and

- in the case of HTGRs, have safety highly dependent upon the quality of the fuel fabrication and inspection process.

NRC has had experience with each of these; however, requirements will need to be developed addressing these topics, as discussed below.

Factory Fabrication

NRC's role in the scope of vendor inspection and transportation needs to be addressed, focusing on those aspects of fabrication and transportation that can affect safety. In particular, insights from the PRA can be used to identify key features that are important to safety and should be inspected.

Fabrication Outside the United States

The role of NRC in inspecting and regulating components fabricated outside the U.S. needs to be addressed, building upon current experience in this area. The preferred approach would be to establish requirements on the applicant to provide controls and inspections on non-U.S. vendors that ensure quality, thus putting the burden on the applicant, not NRC. NRC would then specify what documentation is to be submitted by the applicant to confirm the appropriate quality has been achieved. In addition, the use of non-U.S. codes and standards for design and fabrication will require staff review and acceptance. As directed by the Commission in its SRM of June 26, 2003, staff review of international codes and standards is to be done on a case-by-case basis, in the review of applications or pre-application submittals.

Fuel Quality

How to ensure fuel quality over the life of the plant is an issue of concern (this concern is particularly applicable to HTGRs, whose fuel quality is key to plant safety and needs to be controlled at the fuel fabrication facility). To address fuel quality over the life of the plant, the requirements need to cover what documentation, controls and testing an applicant/licensee needs to provide to ensure the fuel that is put into the reactor is satisfactory (this approach would put the burden on the licensee versus NRC to ensure fuel quality).

G.2.2.3 Operating Stage

At the operating stage, good operating practices (such as the use of procedures, training, etc.) will help minimize human errors and maintain the plant in a condition consistent with the PRA and safety analysis.

Since the quality of the operation of a NPP can have a large impact on safety and risk, it is important that the requirements for future NPPs address the key aspects of operation that are important to safety. Many areas associated with operation are expected to be similar to those for currently operating plants. For these areas, requirements for new plants can build upon and utilize much of the existing regulatory requirements, since they are largely technology-neutral in nature. These areas would include:

- training
- use of procedures
- maintenance
- work control
- configuration control.

However, due to the technology-neutral nature of the proposed licensing approach, the use of PRA, the protective strategy structure and the defense-in-depth principles, certain aspects of the requirements will need to be different. Specifically, the development of requirements in the following areas will require a technology-neutral and risk-informed approach:

- radiation protection
- surveillance and inspection
- worker protection during accidents
- staffing
- technical specifications
- human factors
- corrective actions
- safety-security and preparedness interface

Requirements will need to be developed addressing these topics, as discussed below.

Radiation Protection

The design needs to include provisions limiting radiation doses to workers and the public from routine operation consistent with 10 CFR 20. This includes implementing the concept of As Low As Reasonably Achievable (ALARA) for workers and for releases to the environment. In this regard, 10 CFR 50, Appendix I provides guidance on permissible releases to the environment for light water reactors (LWRs). The risk-informed and performance-based licensing approach will need to develop criteria or generic guidance to apply the ALARA concept to other technologies. Additionally, for technologies using intermediate heat transfer systems between the reactor coolant system and the power generation system, the pressure in the intermediate system should be higher than in the RCS to ensure leakage into the RCS, thus confining radioactive material to the RCS.

Surveillance and Inspection

Risk information can be useful in identifying and prioritizing SSCs for inspection and surveillance. In addition to focusing resources on the most important SSCs, risk information can also help identify the failure modes which contribute most to risk, and therefore, should be looked for in conducting surveillance and inspection (e.g., unavailability).

Importance measures (e.g., risk achievement worth) can be particularly useful in the identification process by assessing the importance of individual SSCs to overall plant risk. Thus, the requirements and their implementing guidance should call for the use of risk information and importance measures in developing, implementing and maintaining inspection and surveillance programs.

Worker Protection During Accidents

10 CFR 20 and 10 CFR 50, Appendix I, provide guidelines for worker protection from radiation during normal operation. However, 10 CFR 50 does not address worker protection during accident conditions. Guidelines for such protection should be provided and, for radiation exposure, 10 CFR 20.1201 and 10 CFR 20.1206 requirements could be used. For non-radiological hazards (e.g., temperature, chemicals, etc.) limits should also be established consistent with existing standards for occupation exposure. Section G.2.5 provides additional discussion on this item.

Staffing

The size, composition and role of the operating staff may be different for new plants. Factors that could affect staffing are:

- the modular nature of some designs
- the use of passive safety features
- longer plant response times
- the use of non-LWR technologies

The PRA will be an important source of information to help establish the number, role and responsibilities of the operating staff. In developing requirements for staffing, the burden should be on the applicant to demonstrate through modeling of human actions, the use of simulators and/or mockups, the PRA and safety analysis what human actions are needed and what size and qualifications of the operating staff are necessary to carry out these actions, consistent with the guidelines for worker protection described above.

Technical Specifications (TS)

Technical specification limits will need to be established at the technology-specific and design specific level. A scheme that utilizes insights from the PRA should be developed. This scheme would involve using the licensing basis events from the frequent, infrequent and rare categories for the specific design. The SSC operability and performance limits associated with these LBEs would then be included in the technical specifications. In addition, the success criteria from the PRA should be reviewed to establish the TS limits. Lessons learned from efforts to risk-inform the technical specifications for currently operating LWRs should be considered in developing the requirements and any implementing guidance. It is likely that some experience will be needed in order to gain confidence in the limits that would be established by such a scheme.

Human Factors

A design that employs good human factors and man-machine interface practices will contribute to stable and safe plant operations. In this regard, guidelines have been developed for good human factors designs practices and good control room design practices for LWRs. These are found in

NUREG-0711[10], "Human Factors Engineering Program," and could be used as guidance to supplement the requirements. However, in general the requirements should, in a technology-neutral manner, address good human factors engineering practices that promote carrying out operations in a timely and accurate fashion, such as:

- lighting
- accessability
- labeling
- color coding
- environmental conditions (e.g., temperature, humidity, radiation)
- procedures
- training

Likewise, good man-machine interface practices (especially when interfacing with computer controlled equipment) should be addressed in a technology-neutral manner in the requirements. This would include:

- navigation through computerized procedures or diagnostic systems, and
- information displays.

Guidance on good man-machine interface practices is found in NUREG-0700[11], "Human-System Interface Design Review Guidelines." Finally, the PRA can provide valuable insights regarding the importance of human actions, which can then be emphasized in procedures and training programs.

Corrective Actions

Establishing and maintaining a corrective action program is fundamental to ensuring good operations. However, in a risk-informed approach, the PRA can provide valuable insights when problems arise regarding risk, which can factor into allowable outage times and priorities for corrective actions. Accordingly, the requirements should call for a corrective action program to be established and maintained with the following characteristics:

- the scope of the corrective action program should be defined by the scope of the PRA,

- the priority of corrective actions should be consistent with their risk importance, as identified using the PRA,

- the extent of performance monitoring should be commensurate with the safety importance of the SSCs,

- performance monitoring information should be fed back into the PRA in a timely fashion, and

- corrective actions should be directed toward ensuring the assumptions in the PRA remain valid or appropriate changes should be made to the design/operations to reflect the as monitored performance.

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[10] NUREG-0711: "Human Factors Engineering Program"

[11] NUREG-0700: "Human System Interface Design Review Guidelines"

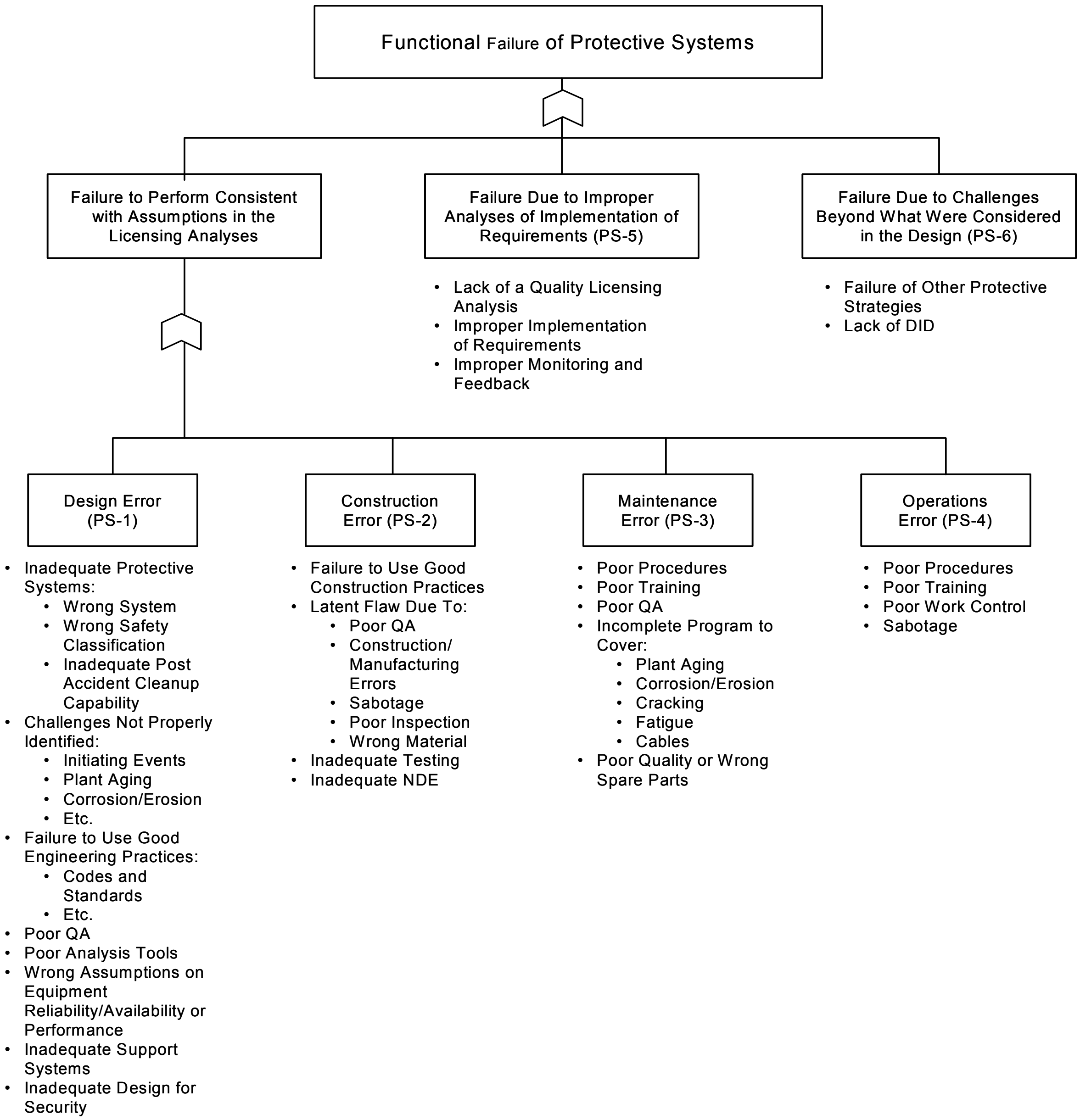

Figure G-3 Logic tree for the protective systems strategy.

Safety-Security and Preparedness Interface

When plant configurations or procedures are changed (due to maintenance, plant modifications, technical specification changes, etc.) the impact on security and preparedness needs to be considered with respect to factors such as changes in target sets, vulnerabilities, etc. Such impacts need to be factored into decision-making and the need for any compensatory measures.

Likewise, changes in security or preparedness measures also need to be assessed with respect to their impact on plant safety.

G.2.3 Protective Systems

The protective systems protective strategy ensures that, should a challenge occur, systems are in place that will mitigate the resulting event sequences, i.e., arrest the sequences with no damage or minimize damage to the suite of barriers considered in the barrier integrity protective strategy.

The pathways leading to functional failure of a set of protective systems are shown in the logic tree of Figure G-3. The scope of the protective systems covered by this strategy include the front line protective systems and their support systems: those systems that provide needed services to the front line protective systems (e.g., I&C, electric power, and cooling). Note that the actual definition of protective system sets that need to fail to lead to the actual loss of a protective function will depend on the details of final system design. At the top level, the major pathways leading to functional failure of protective systems are (1) failure of the protective systems to perform consistent with assumptions in the licensing analyses, (2) failures due to improper analyses or implementation of requirements, and (3) failures due to challenges beyond what were considered in the design. Each of these top level pathways is discussed further below.

Items that contribute to failures in the first top level pathway are design errors, construction (which includes manufacturing) errors, maintenance and operational errors. Design errors can lead to system failure by not properly including the events or conditions under which protective systems need to function, the system performance needed to respond to these events, or the support systems needed into the design. Such design errors can result from poor QA, wrong assumptions on equipment performance or reliability/availability or not using good engineering practices in the design. Failures can also result from inadequate support systems or poor design for human actions or security. Accordingly, good QA is needed along with the use of good engineering practices and validated analytical tools. Also, protective systems should receive a safety classification and special treatment (e.g., QA, EQ) consistent with their safety importance to ensure they are available and operable when needed during the operating stage.

Construction and manufacturing errors can also lead to protective systems failure by introducing latent flaws or by not thoroughly inspecting or testing the systems for conditions under which they are to operate. The latent flaws can be the result of poor inspection, poor QA or QC, use of the wrong material or fabrication techniques or sabotage. Accordingly, the use of good construction, testing, inspection and QA/QC practices are important to preventing failures.

Maintenance errors can also contribute to failure of protective systems. Maintenance programs that are incomplete may miss important contributors to failure such as plant aging, corrosion, etc. Poor training, procedures, spare parts, or QA/QC can cause maintenance errors and allow them to go undetected. Accordingly, maintenance programs should be comprehensive, including items such as aging management, and use of trained personnel and verified procedures.

Operations errors can also cause failure of protective systems. Such errors can result from poorly trained operators, poor procedures, poor work or configuration control or sabotage. Accordingly, the requirements need to address these factors.

The second major pathway to failure of protective systems is that associated with failures due to improper analyses or implementation of requirements. Accordingly, ensuring quality analyses, the use of validated analytical tools and QA, along with items such as monitoring/feedback, technical specifications and safety classification should be used to ensure proper analyses and implementation of requirements during design and operation.

For the third major pathway (failures due to challenges beyond what were considered in the design), protection is provided by the other protective strategies (i.e., they are additional lines of defense) and by application of the defense-in-depth principles to account for completeness uncertainty.

Applying the defense-in-depth principles to this protective strategy leads to the following:

- Provide protective systems that can respond to intentional acts as well as inadvertent events. As described in Section G.2.1, “Physical Protection,” security related issues and events need to be considered as an integral part of the design process. As discussed in Section G.2.1, a safety and security assessment should be done integral with design to assess whether or not protective systems design should be modified to make them less vulnerable to intentional acts or better able to mitigate intentional acts.

- Provide protective systems that prevent events from leading to major plant damage as well as preventing the uncontrolled release of radioactive material to the environment should major plant damage occur. Applicants need to propose availability and reliability goals for the protective systems in consideration of the expected frequency of the events they are intended to respond to. Protective systems responding to events expected to occur one or more times during the life of the plant (frequent events in Chapter 6) should have high availability and reliability, whereas protective systems that are in the design to respond to events not expected to occur (infrequent and rare events in Chapter 6) may have a lower availability and reliability. To ensure this concept is implemented, the requirements need to require the designer to propose availability and reliability goals for the protective systems commensurate with the above, with overall plant risk goals and with assumptions used in the PRA.

- Ensure key plant safety functions (i.e., reactor shutdown and decay heat removal) are not dependent upon a single element of design, construction, maintenance or operation. As stated in Chapter 6, the use of risk information replaces the single failure criterion in many areas. However, to account for uncertainties in the performance of key safety function, it is considered reasonable to retain a single failure criterion for key safety functions. Accordingly, each of the key safety functions should be accomplished by redundant, independent and diverse means (means is intended to be a complete system, not one train in a redundant system), with each means having reliability and availability goals commensurate with overall plant risk goals. This represents a structuralist approach to defense-in-depth for these important functions to account for unquantified uncertainties, including common cause failure. It is intended that the requirement for redundant, diverse and independent means for reactor shutdown and decay heat removal be applied in the following manner:

  — The design should ensure that for frequent and infrequent event sequences, redundant, diverse and independent means for reactor shutdown and decay heat removal are available. For frequent events, the reliability and availability of the redundant, independent and diverse shutdown and decay heat removal systems should be sufficient such that no frequent event will make them inoperable. For infrequent events, which may involve loss of one decay heat removal system or means of reactor shutdown, the other system or means should have sufficient reliability and availability to be considered functional and ensure that the acceptance criteria for infrequent event sequences are met.

- — This functional requirement would not apply to event sequences in the rare category.

- In assessing the performance of protective systems, uncertainties in reliability, and performance should be accounted for and appropriate safety margins provided. For new types of equipment or equipment with little or no operating experience at the conditions it will experience, a reliability assurance program (see Section G.2.2) needs to be provided to demonstrate and monitor equipment to ensure the assumptions on reliability, availability and performance used in the PRA and safety analyses are met. As discussed in Chapter 6, regulatory limits that are related to the failure of a piece of safety significant equipment, barrier or function should be set at the lower end of the expected uncertainty band so as to have an insignificant probability of failure as long as the limit is not exceeded, thus providing margin to the actual expected failure point. Also, the source term to be used in the safety analysis is to be that associated with the 95% confidence level (i.e., 95% of the ST is expected to be below the value used in the safety analysis). Use of the 95% value is intended to provide margin for the uncertainty in modeling and in calculating the various phenomena associated with fission product release and transport. Finally, as discussed in Chapter 6, the dose calculated for LBEs is to be compared to the F-C curve using the 95% confidence value of the calculation. The use of the 95% value of the calculation is, among other things, intended to demonstrate the conservatism of the PRA calculations (i.e., margin between the PRA analysis results and the F-C curve).

  In addition to the items discussed above, two other areas that will inherently result in safety margins are worth noting. These areas are: (1) the use of consensus codes and standards in the design of components and structures provides additional safety margins due to the conservatism built into their design rules and (2) the use of the NRC Safety Goal QHOs as the level of safety to be achieved provides margin to the "adequate protection" standard for licensing.

- Unacceptable releases of radioactive material should be prevented. Accordingly, a means to prevent the uncontrolled release of radioactive material should be included in the design, consistent with the barrier integrity protective strategy (See Section G.2.4).

- Plant siting can affect the types and performance of safety systems since site specific hazards may be different. Site specific hazards and conditions should be considered in the design consistent with 10 CFR 100 and the licensing analysis.

The above defense-in-depth considerations are reflected in the topics which the requirements should address, as shown in Table G-3. Table G-3 identifies the questions that need to be answered to address each of the potential causes of protective system failure. The answers to these questions are organized by whether they apply to design, construction or operation and identify the topics which the technology-neutral requirements need to address to ensure the success of this protective strategy. These topics are directed toward ensuring that quality analyses are used in the design process, that good engineering practices are used in the design and construction, that the equipment is tested, maintained and inspected over the life of the plant and that plant operations are conducted in a fashion that assures high reliability and availability of the protective systems (e.g., use of procedures and training need to be employed to minimize human errors). These considerations also apply to safety-significant support systems as well as the front line protective systems.

Finally, in assessing the performance of the protective systems (and the performance resulting from the other protective strategies), the design should meet the F-C curve and the QHOs, as described in Chapter 6. The F-C curve is to be met by each accident sequence in the PRA and in the LBE analysis. The QHOs represent an overall assessment of plant risk (considering all plant operating states and SSCs, including spent fuel storage). It is intended that the QHOs be assessed in an integrated fashion such that all new reactors on a site need to meet the QHOs considering their risk in a cumulative fashion.

Table G-3 Protective systems.

Protective Strategy Questions	Answers to Questions		
	Design	**Construction**	**Operation**
Failure to Perform Consistent with Assumptions - Design Errors			
• How should systems be designed to ensure adequate performance and safety? (PS-1)	• Use licensing analysis to determine protective and support system needs (i.e., need quality licensing analysis)	• N/A	• Use PRA to feedback operational experience into design.
	• Meet F-C curve	• N/A	• N/A
	• Meet QHOs, including integrated risk	• N/A	• N/A
	• Meet LBE acceptance criteria (Chap 6)	• N/A	• N/A
	• Use good engineering practices: - consensus design codes and standards - I&C qualification - software V&V - QA - qualified materials - EQ - combustible gas control - coolant/water/ fuel reaction control - qualified analytical tools - quality licensing analysis to determine performance and reliability needed	• N/A	• N/A
	• Specify monitoring and cleanup capability to recover from off normal events (e.g., reactor coolant contamination, containment atmosphere)	• N/A	• Tech specs
	• Safety classification (see Chapter 6)	• N/A	• Tech specs
	• Consider plant aging/ corrosion, etc.	• N/A	• Surveillance

Table G-3 Protective systems.

Protective Strategy Questions	Answers to Questions		
	Design	**Construction**	**Operation**
	• Designer to specify reliability/availability goals consistent with PRA	• N/A	• Monitoring and feedback
Failure to Perform Consistent with Assumptions - Construction Error			
• What needs to be done to prevent construction errors? (PS-2)	• Specify good construction/ fabrication practices as part of the design.	• Use good construction/ fabrication practices: • consensus codes and standards • QA/QC • access control	• N/A
Failure to Perform Consistent with Assumptions - Maintenance Errors			
• What needs to be done to prevent maintenance errors? (PS-3)	• N/A	• N/A	• procedures • training • QA/QC • comprehensive maintenance program, including: - plant aging - cables - corrosion - etc. • quality spare parts
Failure to Perform Consistent with Assumptions - Operation Errors			
• What needs to be done to prevent operations errors? (PS-4)	• Consider human factors and man-machine interface as part of design (e.g., automatic vs. operator actions)	• N/A	• procedures • training • use of simulator • technical specifications • surveillance • ISI • testing • good work control
Failures Due to Improper Analyses or Implementation			
• How can failures due to improper analyses or implementation of requirements be prevented? (PS-5)	• Ensure quality analysis and that plant is designed consistent with PRA and safety analysis. • QA	• Ensure plant is constructed consistent with design. • QA/QC	• technical specifications • monitoring and feedback

Table G-3 Protective systems.

Protective Strategy Questions	Answers to Questions		
	Design	Construction	Operation
Failures Due to Challenges Beyond What Were Considered in the Design			
• How can challenges beyond what were considered in the design (i.e., uncertainties) be accounted for? (PS-6)	• Provide 2 independent redundant and diverse means to shutdown the reactor and remove decay heat	• N/A	• N/A
	• reliability assurance program	• N/A	• N/A

N/A = Not applicable

G.2.4 Barrier Integrity

The barrier integrity protective strategy is intended to ensure that the design provides sufficient physical (or chemical) barriers to prevent the uncontrolled release of radioactive material. The number and nature of the barriers will be technology and design dependent. However, if at least one barrier remains, the public is protected and workers are given a measure of protection. Barrier integrity depends on adequate design, construction, maintenance and operation and, in some cases, on the success of protective systems. The logic tree of Figure G-4 lays out the events that can lead to functional failure of the barriers. Barrier integrity applies to barriers associated with the reactor. Figure G-4 begins by identifying three major top level pathways that can lead to failure. These are:

- Failure to perform consistent with assumptions in the licensing analyses;
- Failures due to improper analyses or implementation of requirements; and
- Failures due to challenges beyond what were considered in the design.

Each of these is discussed in more detail in the following paragraphs.

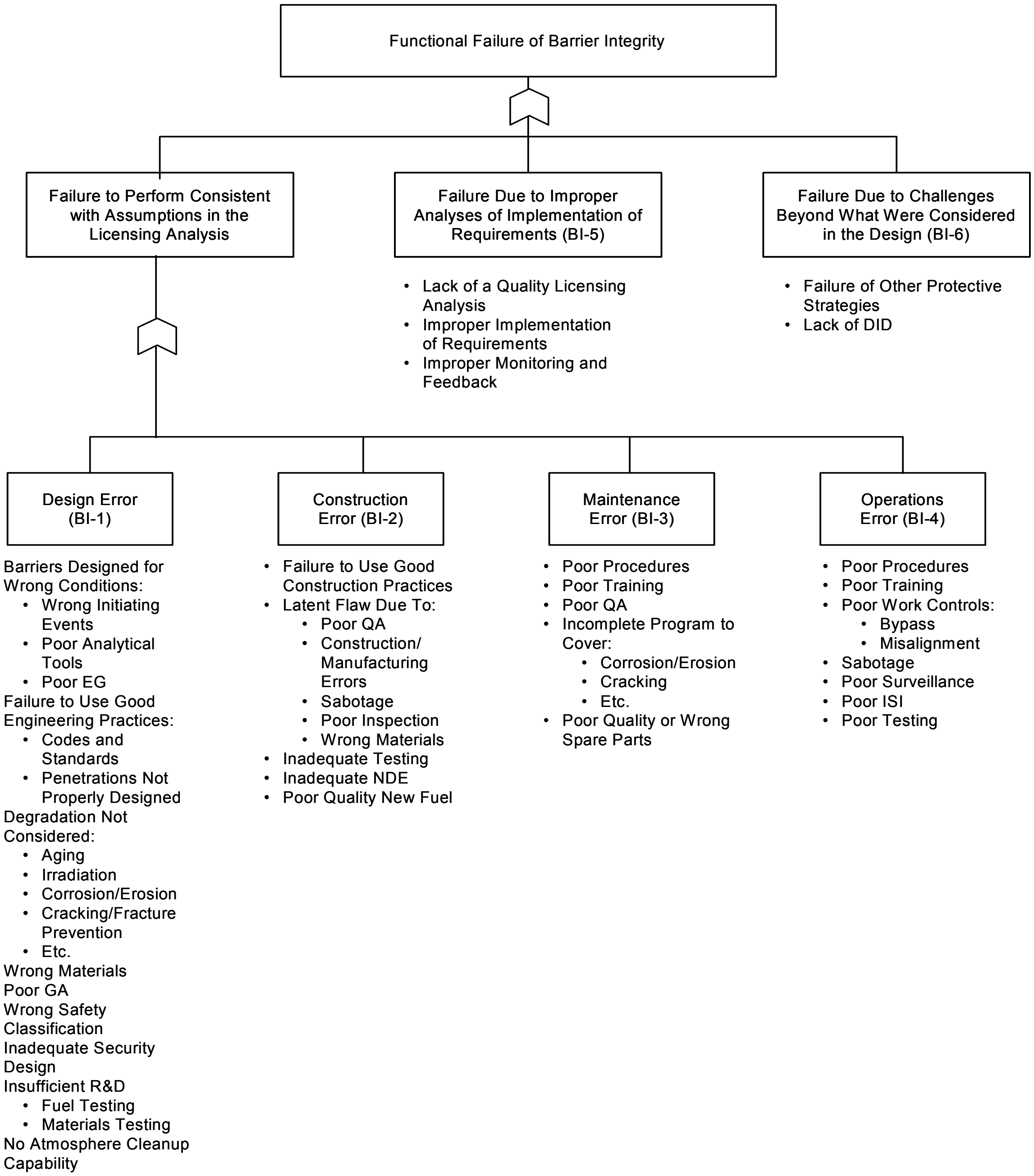

Figure G-4 Logic tree for the barrier integrity strategy.

The first major pathway (Failure to Perform Consistent with Assumptions in the Licensing Analyses) can be affected by design, construction, maintenance or operation errors, as discussed below.

Design errors leading to barrier failure can occur because the design is inadequate for the actual conditions that occur or conditions in excess of the design conditions occur. Failure can also occur by a failure of security, i.e., a loss of physical protection. Other design factors affecting barrier integrity are failure to consider barrier degradation mechanisms, poor QA/QC, poor penetration design and poor selection of materials (e.g., fracture prevention). Use of good engineering practices (e.g., codes and standards) can help reduce the potential for design errors.

Construction and manufacturing errors are another source of barrier failure. Using good construction practices and having a good QA and QC program during the construction phase is essential to ensuring the plant is built as intended. Inspection, NDE and testing of barriers as construction proceeds are means to ensure the plant has been built as intended. Manufacturing processes for the fuel need to be controlled and qualified to ensure that fuel performance and source terms are consistent with design assumptions.

Maintenance errors are another potential source of barrier failure. These can occur due to leaving equipment in the wrong position, making a work error (e.g., forgetting to install a seal), not being trained or not following procedures. Accordingly, good work control, training and procedures are needed as well as a post maintenance test program to verify that barrier integrity is established. Finally, the maintenance program needs to cover all important degradation mechanisms that can affect barrier integrity.

Preventing operational errors is also important to maintaining barrier integrity. Poor procedures, training or work control could lead to barrier bypass or loss of integrity. To help prevent these errors, good training programs, verified procedures, surveillance, ISI and testing are needed. Also, sabotage is a potential source of barrier failure.

The second major pathway to barrier failure is associated with failures due to improper analyses or implementation of requirements. The licensing analysis will determine what barriers need to be in the design and how they should be designed. For normal operation, reliable barriers to retain the fission products in the reactor and reactor coolant in the coolant system are necessary to meet the low levels of radioactive material release specified for normal operation. To ensure reliable barriers, the barriers should be designed and built to accepted consensus design codes using materials qualified for the intended service and accepted quality assurance measures.

For off-normal conditions, the event selection criteria discussed in Chapter 6 can be used to define the event scenarios and conditions which need to be considered in designing the barriers. These criteria categorize event scenarios into those that are expected to occur one or more times during the life of the plant (frequent events), those that may occur once in a population of plants (infrequent events) and those considered in assessing overall plant risk and emergency preparedness (rare events).

Deterministic acceptance criteria for frequent and infrequent events have been developed in Chapter 6. Criteria on plant risk have also been developed in Chapter 6. To ensure the barriers perform as intended, they need to be qualified for the service conditions expected and designed to prevent brittle fractures. This may involve research and development to verify fuel performance/source term and equipment qualification (EQ) and mechanical testing to verify the performance of the barrier material and any associated mechanical items. Also, the analysis of barrier performance under off-normal conditions will require safety analysis tools that need to be validated against experimental data. Depending upon the importance of the barriers to meeting the acceptance criteria, they may be assigned a safety classification (as described in Chapter 6) that will help ensure their performance, availability and operability are maintained over the life of the plant.

It is also important that the assumptions associated with the analysis be properly implemented and controlled. Accordingly, items such as monitoring/feedback, technical specifications and safety classification need to capture the key assumptions and provide control over the plant configuration and operation.

For the third major pathway (unanticipated challenges and failures), protection is provided by the other protective strategies (i.e., they are additional lines of defense) and by application of the defense-in-depth principles to account for completeness uncertainty.

Applying the defense-in-depth principles to the barrier integrity protective strategy leads to the following:

- The number of barriers and their design should be based upon both intentional as well as inadvertent events. By requiring the design be done in an integral fashion considering security (see Section G.2.1) and preparedness (see Section G.2.5), the barriers need to consider both.

- The barriers should be designed with both accident prevention and mitigation in mind. Accident prevention will be achieved by ensuring that the barriers are designed to be highly reliable and can withstand a range of off-normal conditions. High reliability needs to be achieved by the use of good engineering practices (such as the use of consensus design codes and standards, qualification of materials, QA, etc.) in the design and performing surveillance, inspection and testing during the plant lifetime. Barriers also need to be designed to maintain their integrity for events expected to occur during the plant lifetime such that their failure does not become an initiating event.

  Accident mitigation will be achieved by ensuring the barriers perform their function of containing radioactive material. The events for which they need to perform their function, their design and their degree of leak tightness will be design dependent, as will the total number of barriers needed. Minimum requirements for barriers are discussed below.

- Defense-in-depth requires that key safety functions not be dependent upon a single element of design, construction, operation or maintenance. Application of this principle to barrier integrity implies multiple barriers are needed, since containment of radioactive material is considered a key safety function. Accordingly, at least two independent, redundant and diverse barriers to the release of radioactive material should be provided, since the failure of one of these barriers (e.g., the reactor coolant system barrier) could be an initiating event. In general, the barriers, in conjunction with other plant features, need to be capable of limiting dose to the public consistent with the frequency consequence curve in Chapter 6.

- In the design and safety analysis, uncertainties in reliability and performance need to be accounted for and appropriate safety margins provided. As discussed in Chapter 6, regulatory limits that are related to the failure of a piece of safety equipment, barrier or function should be set at the lower end of the expected uncertainty band so as to have an insignificant probability of failure as long as the limit is not exceeded, thus providing margin to the actual expected failure point. However, not all uncertainties can be quantified. Therefore, it is considered reasonable to require each design to have additional radiological containment functional capability to mitigate against accident scenarios that result in a release of larger amounts of radioactive material from the reactor core and coolant than anticipated. This would then provide margin to account for unquantified uncertainties that result in a larger source term available for release to the environment (e.g., from security related events). Accordingly, as a deterministic defense-in-depth provision, each design should have a radiological containment functional capability (i.e., the capability to establish a controlled low leakage barrier), separate from the fuel and RCS, in the event plant conditions result in the release of radioactive material from the core and reactor coolant system in excess of anticipated conditions. This functional capability needs to be maintained for all frequent and infrequent events and for any rare or security related events where credit is taken for its performance. The specific conditions regarding the leak tightness, temperature, pressure and time available to establish the containment functional capability will be design specific. The containment functional capability should be based upon a process that defines those events representing a serious challenge to fission product retention in the core and coolant system. The events need to be agreed upon between the applicant and the NRC consistent with the technology and safety characteristics of the design. The events could represent situations where fission product retention in the core and coolant system suddenly changes due to small changes elsewhere, low probability (i.e., rare) events from the PRA, a security related event or an assumed fuel damage event. The provision discussed in this paragraph is a policy issue requiring Commission review and direction and is discussed in Appendix C.

For LWRs, core melt accidents will likely continue to be used to establish the design conditions for the containment functional capability. For non-LWRs, examples of the types of events that could be considered for establishing the design conditions for the controlled leakage barrier are:

- <u>HTGRs</u>
  - graphite fire in the core
  - water ingress to the core
  - loss of coolant accident in conjunction with poor quality fuel
- <u>LMRs</u>
  - flow blockage in the core
  - large liquid metal fire
  - loss of normal heat removal in conjunction with poor quality fuel

The selection of the events to be used to establish the design conditions for the radiological containment functional capability is not intended to impose a traditional LWR type containment on all technologies, but rather to allow each technology to have designs that reflect their unique safety characteristics while providing margin for uncertainties in the source term available for release to the environment (e.g., venting to the atmosphere early in an accident scenario may be acceptable for some technologies). These events should also be used for siting purposes as described in Sections 6.4.3 and G.2.2.1.

The selected events should be analyzed mechanistically to determine the timing, magnitude and form of radionuclide released into the reactor building, and the resulting temperature, pressure and other environmental factors (e.g., combustible gas) in the building over the course of the event. The timing of closure and the allowable leak rate should then be established such that the worst two-hour exposure at the EAB and the exposure at the outer edge of the LPZ for the duration of the event do not exceed 25 rem TEDE. Chapter 6 contains additional guidance regarding analysis of this event.

- Barriers need to prevent the unacceptable release of radioactive material. Accordingly, to account for uncertainties (see paragraph above), the reactor should have a radiological containment functional capability independent from the fuel and RCS, as discussed above.

- Barrier integrity interfaces with siting in that some aspects of barrier performance may be determined by site characteristics (e.g., meteorology, population distribution). Likewise, barrier integrity can also affect the type and extent of off-site protective measures needed. These should be accounted for in the design.

The above defense-in-depth considerations have been factored into the requirement topics shown in Table G-4. Table G-4 shows a set of questions and answers associated with the Barrier Integrity protective strategy. The questions are organized by the top level branches of the logic diagram and the answers (i.e., the topics which need to be covered by the requirements) are arranged by whether they apply to design, construction or operation.

Table G-4 Barrier integrity.

Protective Strategy Questions	Answers to Questions		
	Design	Construction	Operation
Failure to Perform Consistent with Assumptions - Design Errors			
• How should adequate barrier design (integrity and reliability) be assured? (BI-1)	• Design barriers consistent with: - Chapter 6 event selection criteria - Chapter 6 LBE acceptance criteria (probabilistic, e.g., F-C curve, and deterministic) - Safety classification - EQ - Consider degradation mechanisms • Provide barriers for: - fission product retention (in the fuel) - coolant retention (in the reactor cooling system) - Other capability, as necessary to meet safety objectives • Use good engineering practices: - quality assurance - materials qualification - brittle fracture prevention - use of accepted design codes and standards	• N/A	• N/A

Table G-4 Barrier integrity.

Protective Strategy Questions	Answers to Questions		
	Design	Construction	Operation
	- use of validated safety analysis tools - consider aging and other degradation phenomena - proper penetration design - atmosphere cleanup capability		
Failure to Perform Consistent with Assumptions - Construction Errors			
• What needs to be done to prevent construction errors? (BI-2)	• specify construction/ manufacturing techniques at the design stage.	• use good construction/ manufacturing practices: - consensus construction codes and standards - QA/QC - inspection - testing - NDE - assure fuel quality over the life of the plant • access control • surveillance	• N/A
Failure to Perform Consistent with Assumptions - Maintenance Error			
• What needs to be done to prevent maintenance errors? (BI-3)	• N/A	• N/A	• verified procedures • training • QA/QC • have a comprehensive maintenance program • use quality spare parts
Failure to Perform Consistent with Assumptions - Operations Error			
• What needs to be done to prevent operational errors? (BI-4)	• Use good HF and HMI engineering • Use fault tolerant designs	• N/A	• verified procedures • training • use of simulator • work control • surveillance • ISI • IST

Table G-4 Barrier integrity.

Protective Strategy Questions	Answers to Questions		
	Design	**Construction**	**Operation**
Failures Due to Improper Analyses or Implementation of Requirements			
• How can failures due to improper analyses or implementation of requirements be prevented? (BI-5)	• Use verified analytical tools • Quality PRA and safety analyses • QA (i.e., ensure plant is designed consistent with PRA and licensing analysis).	• QA/QC (i.e., ensure plant is constructed consistent with design).	• technical specifications • monitoring and feedback
Failures Due to Challenges Beyond What Were Considered in the Design			
• How can challenges beyond what were considered in the design (i.e., uncertainties) be accounted for? (BI-6)	• at least 2 barriers for the prevention of releases of FP from the reactor	• N/A	• EP
	• provisions to establish a radiological containment functional capability independent of fuel and RCS for the reactor.	• N/A	

N/A = Not applicable

G.2.5 Protective Actions

The protective actions strategy ensures that adequate systems, equipment, and practices are in place to control and terminate the accident progression, to minimize damage to the barriers, to limit the release of radionuclides, to protect workers, and to limit public health effects. Protective actions generally include EOPs, accident management and on-site and off-site emergency preparedness.

Figure G-5 is a logic tree showing the pathways that can lead to failure of protective actions. At the top level, three major pathways to failure are: (1) failure to take protective actions consistent with assumptions in the licensing analysis, (2) failure due to improper analyses or implementation of requirements, or (3) failures due to challenges beyond what were considered in the design. Each of these top level pathways is discussed further below.

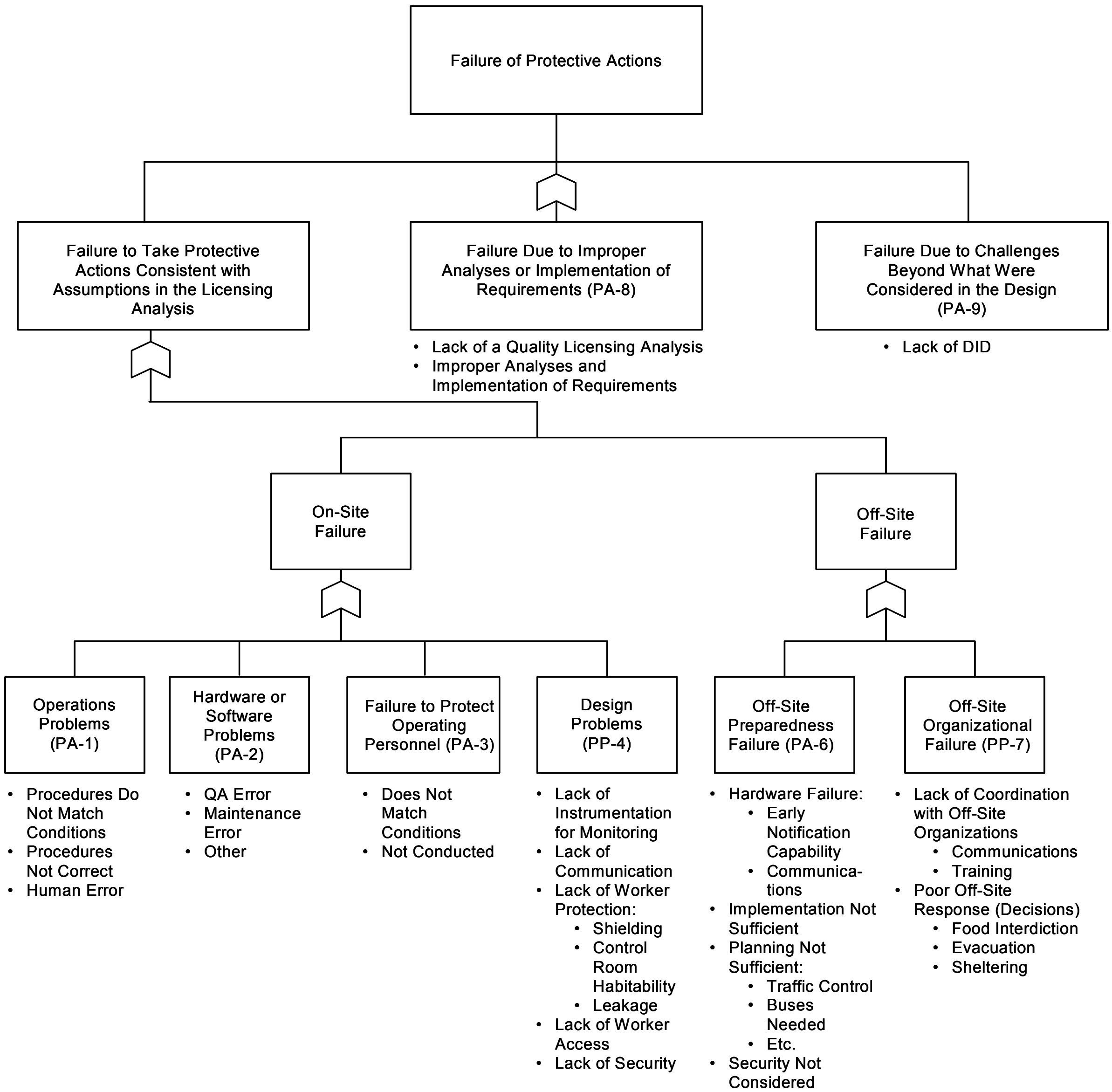

Figure G-5 Logic tree for the protective actions strategy.

In the first top level pathway (Failure to Take Protective Action Consistent with Assumptions in the PRA and Safety Analysis), failure can be associated with either on-site or off-site actions, as shown in Figure G-5. Failure of on-site protective actions can be associated with operations, hardware or software, training or design (e.g., inability to monitor radioactivity releases). Off-site failures can occur in areas regulated by the NRC or in areas controlled by other agencies. For example, state and local officials are responsible for many aspects of the off-site response (e.g., evacuation).

On-site failures due to operational problems can result in failure to terminate the accident (thus making conditions on-site, and possibly off-site, worse) or failure to adequately protect operating personnel. Operating personnel are vital to plant safety and are called on to perform safety related actions during design basis and beyond-design-basis events (e.g., accident management actions). Accordingly, protection of the operating staff during accidents should also be considered in the design and operation of future reactors.

General Design Criteria (GDC) 19 of 10 CFR Part 50 Appendix A currently requires main control rooms to be designed to ensure habitability under a variety of conditions, including design basis accident conditions. The conditions which need to be considered include a postulated source term representative of a LWR core melt accident (or an alternate source term) and chemical releases. As a result, LWR main control rooms are provided with shielding and habitability systems that ensure the safety of the operators during the postulated conditions. Accordingly, the risk-informed and performance-based requirements should include a similar provision for protection of control room staff during accidents, recognizing the use of the PRA to select the accident scenarios which need to be considered and the use of scenario specific source terms.

However, no corresponding requirements exist in 10 CFR 50 for protection of the operating staff outside the main control room, who may be called upon to perform accident management actions and communicate with other staff during accident situations. In the development of accident management programs for existing LWRs (which were developed on the basis of a voluntary industry initiative), it was recognized that access by the operating staff to certain portions of the plant was essential to carry out the planned actions. Accordingly, NEI, in its "Severe Accident Issue Closure Guidelines" document[12] (NEI-91-04, Rev. 1, dated December 1994) on the development of accident management programs, identified operational and phenomenological conditions as factors which need to be assessed in planning and implementing operator accident management actions.

For new plants, the risk-informed and performance-based requirements should require that the procedures and accident management programs consider the environment (e.g., temperature, radiation) in which local operator actions take place and ensure that the design (e.g., shielding, access) and procedures sufficiently protect all the operators so that the actions can be safely accomplished without serious injury. For radiation exposure (during such activities), the limits in 10 CFR Part 20.1201, "Occupational Dose Limits" should be used for frequent event scenarios and 10 CFR Part 20.1206, "Planned Special Exposures" should be used as the measure to prevent serious injury for personnel outside the control room during frequent and rare event scenarios. Regulatory Guide 8.38 provides additional guidance in this area regarding access to high radiation areas. For personnel inside the control room, limits similar to those in GDC-19 could be used. Scenario specific source terms may be used in the assessment, consistent with those used in other accident analyses. Other accepted limits should be applied for other hazards (temperature, chemicals, etc.).

On-site hardware or software problems can lead to unintended actions and/or poor decisions. Accordingly, measures to ensure reliable equipment and software are needed. Poor training can also lead to the same consequences as poor operations or poor hardware/software. Training programs need to be complete and conducted periodically to keep operating personnel skills current. Design problems can result in needed equipment not being present, instrumentation and/or communication not sufficient to understand the accident, personnel access and habitability restricted more than anticipated or personnel injury or death. Therefore, during the design stage,

[12] NEI-91-04, Rev. 1: "Severe Accident Issue Closure Guidelines"

accident scenarios (including those related to security) need to be considered integral with the design and measures to ensure good EOPs and accident management need to be provided.

Off-site preparedness failures can lead to failure to take measures needed to protect the public. Such failures could be due to hardware problems (e.g., failure to notify), poor planning (e.g., traffic jams delay evacuation) or insufficient implementation to monitor the accident consequences. Off-site organizational failures can also lead to failures to adequately protect the public. Such failures can be due to poor coordination among off-site authorities, poor communication, poor training or poor decisions (i.e., not implementing the appropriate protective measures at the appropriate time).

The second top level pathway is associated with failures due to improper analyses or implementation of requirements. Quality analyses and the use of verified analytical tools are essential. In addition, the EOPs and AM procedures should be developed in an integrated fashion with the design so that the design can provide reasonable measures for AM and ensure the procedures are consistent with the PRA and safety analysis.

For the third top level pathway (failures due to challenges beyond what were considered in the design), protection is provided by the application of the defense-in-depth principles to account for completeness uncertainty. Applying the defense-in-depth principles to this protective actions strategy leads to the following:

- The development of protective actions should consider intentional acts as well as inadvertent events. The physical protection protective strategy (Section G.2.1) provides further guidance on evaluating security integral with design.

- Protective actions should include measures to terminate the accident progression (referred to as EOPs, and accident management) and pre-planned measures to mitigate the accident consequences (referred to as emergency preparedness). The EOPs, AM procedures and EP need to be developed in an integrated fashion with the design.

- The accomplishment of protective actions should not rely on a single element of design, construction, maintenance or operation. As such, normal operating, EOPs, accident management and EP procedures should be developed so as not to have key safety functions dependent upon a single human action or piece of equipment.

- Protective actions should be developed in consideration of uncertainties and appropriate safety margins provided. As a structural defense-in-depth measure, emergency preparedness should be included in the design and operation to account for unquantified uncertainties. However, it may be reasonable to adjust the extent and timing of EP measures depending on the safety characteristics of the design (e.g., source term, timing of release). Therefore, modification to current EP requirements to allow for consideration of plant specific characteristics should be considered. It should be noted that this is a policy issue requiring Commission review and direction and is discussed in Appendix C.

- Prevention of unacceptable releases of radioactive material should be part of the AM program.

- Plant siting will affect EP and should be considered in developing EP plans.

The above defense-in-depth considerations are reflected in Table G-5. Table G-5 below summarizes each of these pathways in the form of questions, the answers to which identify the topics that the risk-informed and performance-based requirements need to address to prevent pathway failure. The answers (i.e., topics) are arranged according to whether they apply to design, construction or operation.

Table G-5 Protective actions.

Protective Strategy Questions	Answers to Questions		
	Design	Construction	Operation
Failure to take Protective Actions Consistent with Assumptions: On-Site Failure			
• How can operations problems be prevented? (PA-1)	• N/A	• N/A	• Develop comprehensive training programs and require periodic training. • Use of simulator • Use of verified procedures
• How can hardware and software be assured to be operable? (PA-2)	• Reliability assurance program for hardware • Software V and V	• Testing • QA/QC • N/A	• Maintenance program • Testing
• How can it be assured operating personnel are properly protected? (PA-3)	• Provide appropriate shielding and habitability for the control room and other areas needing access.	• N/A	• Establish comprehensive worker protection programs, training and monitoring. • Ensure 10 CFR 20 requirements are complied with.
• How can design deficiencies/ problems be prevented? (PA-4)	• Develop EOP and AM design features and procedures integral with design, including identifying equipment, monitoring instrumentation, and communication needs. • Provide alternate shutdown location	• N/A • N/A	• N/A • N/A
• How can adequate on-site preparedness be assured? (PA-5)	• Develop on-site EP plans and procedures integral with design	• N/A	• N/A • Training • Procedures • Conduct drills & training to demonstrate effectiveness of on-site EP

Table G-5 Protective actions.

Protective Strategy Questions	Answers to Questions		
	Design	Construction	Operation
Failure to Take Protective Actions Consistent with Assumptions - Off-Site Failure			
• How can adequate off-site preparedness be assured? (PA-6)	• Provide adequate emergency planning • Consider security related events	• N/A	• Conduct drills and training to demonstrate effectiveness of off-site EP • Integrate security and preparedness
• How can adequate off-site organizational performance be assured? (PA-7)	• Provide reliable communication equipment	• N/A	• Conduct drills and training to demonstrate effectiveness of EP
Failures Due to Improper Analyses or Implementation of Requirements			
• How can failures due to improper analyses or implementation of requirements be prevented? (PA-8)	• Quality licensing analyses • Use verified analytical tools • Develop EOPs and AM procedures integral with design. • QA	• N/A • N/A • N/A • N/A	• Ensure training program is comprehensive and conducted periodically. • Use of simulator • N/A
Failures Due to Challenges Beyond What Were Considered in the Design			
• How can challenges beyond what were considered in the design (i.e., uncertainties) in protective actions be accounted for? (PA-9)	• Consider security related events beyond the DBT. • Develop EOPs and AM integral with design. • Do not have key safety functions dependent upon a single human action or piece of hardware.	• N/A • N/A • N/A	• Consider security related events beyond the DBT. • Training • Drills • EP • Procedures

N/A = Not applicable

As can be seen from Table G-5, there are a number of topics that should be addressed in the requirements to assure an adequate protective actions strategy. Some of these (e.g., drills, training) can utilize the existing requirements contained in 10 CFR 50, while others will need to be developed in a technology-neutral fashion consistent with a risk-informed approach. A major item in this regard would be a requirement for the development of the design (and its associated systems and instrumentation) in an integrated fashion with security, including the development of

EOP and AM procedures. Such an integrated process would help ensure that the procedures address all of the relevant accident scenarios in the PRA (and scenarios from security considerations) and that the design includes features that facilitate AM.

G.2.6 Summary of Topics for the Protective Strategies

Sections G.2.1 through G.2.5 identify the technical topics that the risk-informed and performance-based requirements need to address to ensure the success of the protective strategies. Some of the topics identified are applicable to more than one protective strategy (e.g., QA, training, etc.). Accordingly, a summary table (Table G-6) has been prepared that consolidates the technical topics from Tables G-1 through G-5, eliminating any duplication. Table G-6 also organizes the topics in a more logical fashion (i.e., by subject) and identifies the appropriate question numbers from Table G-1 through G-5 that identified that topic.

Table G-6 Summary of technical topics for potential requirements.

Topic		Framework Description
(A)	**General Topics Common to Design, Construction and Operation**	
1)	QA/QC (Questions PP-4, SO-1, SO-3, SO-4, SO-6, PS-1, PS-2, PS-3, PS-5, BI-1, BI-2, BI-3, BI-5, PA-2, PA-8)	Appendix G - Section G.2.2
2)	PRA scope and quality (PP-11, SO-1, SO-6, PS-1, PS-5, BI-1, BI-5, PA-8)	Chapters 6, 7 and Appendix F
3)	Use of risk information	All Framework chapters
4)	Integration of Safety, Security and Preparedness (PP-1 through 12)	Chapter 3
(B)	**Physical Protection**	
1)	General (10 CFR 73) (PP-1 through 11)	Appendix G - Section G.2.1
2)	Security performance standards (PP-1 through 12)	Section 6.4
(C)	**Good Design Practices**	
1)	Plant Risk (PS-1, BI-1): - Frequency-Consequence curve - QHOs (including integrated risk)	Chapter 6
2)	Criteria for selection of LBEs (SO-1)	Chapter 6
3)	LBE acceptance criteria (PS-1): • frequent events (dose, plant damage) • infrequent events (dose, plant damage) • rare events (dose) • link to siting	Chapter 6
4)	Initiating event severity (SO-1)	Appendix G - Section G.2.2

Table G-6 Summary of technical topics for potential requirements.

Topic		Framework Description
5)	Safety classification and special treatment (SO-1, PS-1, BI-1, BI-5)	Chapter 6
6)	Equipment Qualification (SO-1, PS-1)	Section G.2.2
7)	Licensing analysis (SO-1) • realistic analysis, including failure assumptions • source term	Chapter 6
8)	Siting and site specific considerations (SO-1)	Appendix G - Section G.2.2
9)	Use consensus design codes and standards (SO-1, PS-1, BI-1)	Appendix G - Section G.2.2
10)	Materials qualification (SO-1, PS-1, BI-1)	Appendix G - Section G.2.2
11)	Protection against natural phenomena (SO-1)	Appendix G - Section G.2.2
12)	Dynamic effects (SO-1)	Appendix G - Section G.2.2
13)	Sharing of structures, systems and components (SO-1)	Appendix G - Section G.2.2
14)	Reactor shutdown and decay heat removal (PS-6)	Appendix G - Section G.2.3
15)	Barriers to release of radioactive material (BI-1, BI-6)	Appendix G - Section G.2.3
16)	Radiological containment functional capability (BI-6)	Appendix G - Section G.2.4
17)	Radiological containment atmosphere cleanup (BI-1)	Appendix G - Section G.2.4
18)	Fracture prevention of radiological containment pressure boundary (BI-1)	Appendix G - Section G.2.4
19)	Electric Power Systems (SO-1)	Appendix G - Section G.2.2
20)	Piping systems penetrating radiological containment boundary (BI-1)	Appendix G - Section G.2.4
21)	Closed System Isolation Valves (BI-1)	Appendix G - Section G.2.4
22)	Vulnerability to a single human action or hardware failure (PA-9)	Appendix G - Section G.2.5
23)	Plant aging and degradation (SO-1, PS-1, BI-1)	Appendix G - Section G.2.2
24)	Reactor inherent protection (i.e., no positive power coefficient, limit control rod worth, stability, etc.) (SO-1)	Appendix G - Section G.2.2
25)	Human factors/man-machine interface (SO-1, SO-5, PS-4, BI-4)	Appendix G - Section G.2.2
26)	Fire protection (SO-1)	Appendix G - Section G.2.2
27)	Control room design (PA-3)	Appendix G - Section G.2.5

Table G-6 Summary of technical topics for potential requirements.

Topic		Framework Description
28)	Alternate shutdown location (PA-4)	Appendix G - Section G.2.5
29)	Reactor core flow blockage and bypass prevention (SO-1)	Appendix G - Section G.2.2
30)	Reliability and availability (SO-1, PS-1, PS-6, PA-2) • establish Reliability Assurance Program • specify goals on initiating even frequency	Appendix G - Section G.2.2
31)	Research and Development (SO-1)	Appendix G - Section G.2.2
32)	Use of prototype testing (SO-1)	Appendix G - Section G.2.2
33)	Combustible gas control (PS-1)	Appendix G - Section G.2.3
34)	Energetic reaction control (PS-1)	Appendix G - Section G.2.3
35)	Prevention of reactor coolant boundary brittle fracture (SO-1)	Appendix G - Section G.2.2
36)	Reactor coolant pressure boundary (SO-1)	Appendix G - Section G.2.2
37)	Reactor coolant activity monitoring and cleanup (PS-1)	Appendix G - Section G.2.3
38)	I and C System (SO-1, PS-1, PA-2) • analog • digital • HMI	Appendix G - Section G.2.2
39)	Protection of operating staff (PA-3)	Appendix G - Section G.2.5
40)	Control of releases of radioactive materials to the environment (SO-1)	Appendix G - Section G.2.2
41)	Monitoring radioactivity releases (PP-4)	Appendix G - Section G.2.5
42)	Qualified analysis tools (SO-1, SO-6, PS-1, PS-5, BI-1, BI-5, PA-8)	Chapter 6
(D)	**Good Construction Practices**	
1)	Use accepted codes, standards, practices (SO-3, PS-2, BI-2)	Appendix G - Section G.2.2
2)	Security during construction/fabrication (See (B) above)	Appendix G - Section G.2.1
3)	NDE during construction/fabrication (SO-3, BI-2)	Appendix G - Section G.2.2
4)	Inspection during construction/fabrication (SO-1, SO-3, BI-2)	Appendix G - Section G.2.2
5)	Testing during construction/fabrication (SO-1, BI-2)	Appendix G - Section G.2.2

Table G-6 Summary of technical topics for potential requirements.

Topic		Framework Description
(E)	**Good Operating Practices**	
1)	Radiation protection (PA-3)	Appendix G - Section G.2.2
2)	Maintenance program (SO-1, SO-5, PS-3, BI-3, PA-2)	Appendix G - Section G.2.2
3)	Personnel qualification (SO-5)	Appendix G - Section G.2.2
4)	Training (SO-1, SO-4, SO-5, PS-3, PS-4, BI-3, PA-1, PA-5, PA-6, PA-7, PA-8, PA-9)	Appendix G - Section G.2.2
5)	Use of Procedures (SO-1, SO-4, SO-5, PS-3, PS-4, BI-3, BI-4, PA-1, PA-5)	Appendix G - Section G.2.2
6)	Use of simulators (SO-5, PS-4, BI-4, PA-1, PA-8)	Appendix G - Section G.2.2
7)	Staffing (SO-1)	Appendix G - Section G.2.2
8)	Aging management program (SO-1)	Appendix G - Section G.2.2
9)	Surveillance program (SO-3, SO-7, PS-1, PS-4, BI-4)	Appendix G - Section G.2.2
10)	ISI (SO-1, SO-3, PS-4, BI-4)	Appendix G - Section G.2.2
11)	Testing (SO-1, SO-3, PS-4, BI-4, PA-2)	Appendix G - Section G.2.2
12)	Technical specifications (SO-5, SO-6, PS-1, PS-4, PS-5, BI-5, BI-6)	Appendix G - Section G.2.2
13)	Emergency Preparedness (PA-4, PA-5, PA-6, PA-9)	Appendix G - Section G.2.5
14)	Monitoring and feedback (SO-1, SO-6, SO-7, PS-1, PS-5, BI-5)	Appendix G - Section G.2.2
15)	Work and configuration control (SO-5, BI-4, PS-4)	Appendix G - Section G.2.2
16)	Maintenance of the PRA (SO-1, PS-1)	Chapter 7
17)	Fuel and replacement part quality (SO-6, PS-3, BI-3)	Appendix G - Section G.2.2
18)	Security (See B above)	Appendix G - Section G.2.1

It needs to be recognized that Table G-6 presents a broad, high level overview of the topics which the risk-informed and performance-based technical requirements need to address. Many details need to be developed in the course of writing the requirements. Accordingly, reference to the appropriate section in the Framework is also shown in Table G-6 for additional guidance.

As described in Sections G.2.1 through G.2.5, the defense-in-depth principles from Chapter 4 were applied to each protective strategy to ensure adequate treatment of uncertainties. Application of the defense-in-depth principles to each of the protective strategies (as described in Sections G.2.1

through G.2.5) has led to the identification of a number of topics, deterministically developed to address uncertainties. Although included in Table G-6, these are also summarized in Table G-7 to illustrate the defense-in-depth provisions identified by the application of the DID principles in Chapter 4.

Table G-7 Defense-in-depth (DID) provisions.

DID Principle	Physical Protection	Stable Operation	Protective Systems	Barrier Integrity	Protective Actions
1) Consider intentional and inadvertent events	Integral Design Process	Integral Design Process	Integral Design Process	Integral Design Process	Integral Design Process
2) Consider prevention and mitigation in design	Security Assessment and Security Performance Standards	Cumulative limit on IE frequencies.	Accident prevention and mitigation - fuel damage criteria - coolable geometry criteria	Accident prevention and mitigation - barrier integrity criteria	- Develop EOPs and accident management integral with design - EP
3) Not dependent upon a single element of design, construction, maintenance, operation	Security Assessment and Security Performance Standards	Ensure events that can fail multiple PS are <10^{-7}/plant year.	Provide 2 independent, redundant diverse means for: reactor shutdown and DHR.	Provide at least 2 barriers	No key safety function dependent upon a single human action or piece of hardware
4) Account for uncertainties in performance and provide safety margins	Security Assessment and Security Performance Standards	Reliability Assurance Program (RAP). Provide safety margins in performance limits.	Reliability and availability goals consistent with assumptions in PRA. RAP. Use of a conservative source term Provide safety margin in regulatory limits.	Provide radiological containment functional capability independent from fuel and RCS. Use of a conservative source term Provide safety margin in regulatory limits.	- EP - For safety margin, use conservative ST in calculations.

Table G-7 Defense-in-depth (DID) provisions.

DID Principle	Physical Protection	Stable Operation	Protective Systems	Barrier Integrity	Protective Actions
5) Prevent unacceptable release of radioactive material	Security Assessment and Security Performance Standards	Ensure events that can fail multiple PS are $<10^{-7}$/plant year.	N/A	Provide radiological containment functional capability independent from fuel and RCS	Accident Management
6) Siting	Security Assessment and Security Performance Standards	Limits on ext. event cumulative frequencies.	N/A	N/A	EP

N/A = Not applicable

G.3 Identification of Administrative Requirement Topics

As discussed earlier in this document, the Framework is to define the scope and content, and provide the overall technical basis for a risk-informed and performance-based approach for new plant licensing. Accordingly, in addition to topics for technical requirements, topics for administrative requirements also need to be considered. As discussed in Chapter 8, existing administrative requirements should be used provided they are risk-informed and performance-based. However, the administrative requirements for this new approach would have some differences from those in 10 CFR 50 because of the risk-informed and performance-based nature of the approach. In either case, the administrative requirements need to be complete, so as to make the set of requirements a stand alone licensing process.

Administrative requirements have an impact on safety in that they define processes, documentation and practices that are necessary to ensure accurate and adequate information is developed, maintained and reviewed such that there is assurance that the plant is designed, constructed, operated and maintained in accordance with the safety analysis. The administrative requirements also ensure sufficient information is provided to the regulator to allow independent verification of plant safety. In effect, this serves as an administrative defense-in-depth measure by providing an independent check on plant safety.

Figure G-6 is a logic tree that illustrates schematically the various elements of administration whose failure could impact safety. Each of the branches on the tree is discussed below with respect to identifying what needs to be done to ensure success of the branch. This then leads to identifying what topics the administrative requirements should address to be complete.

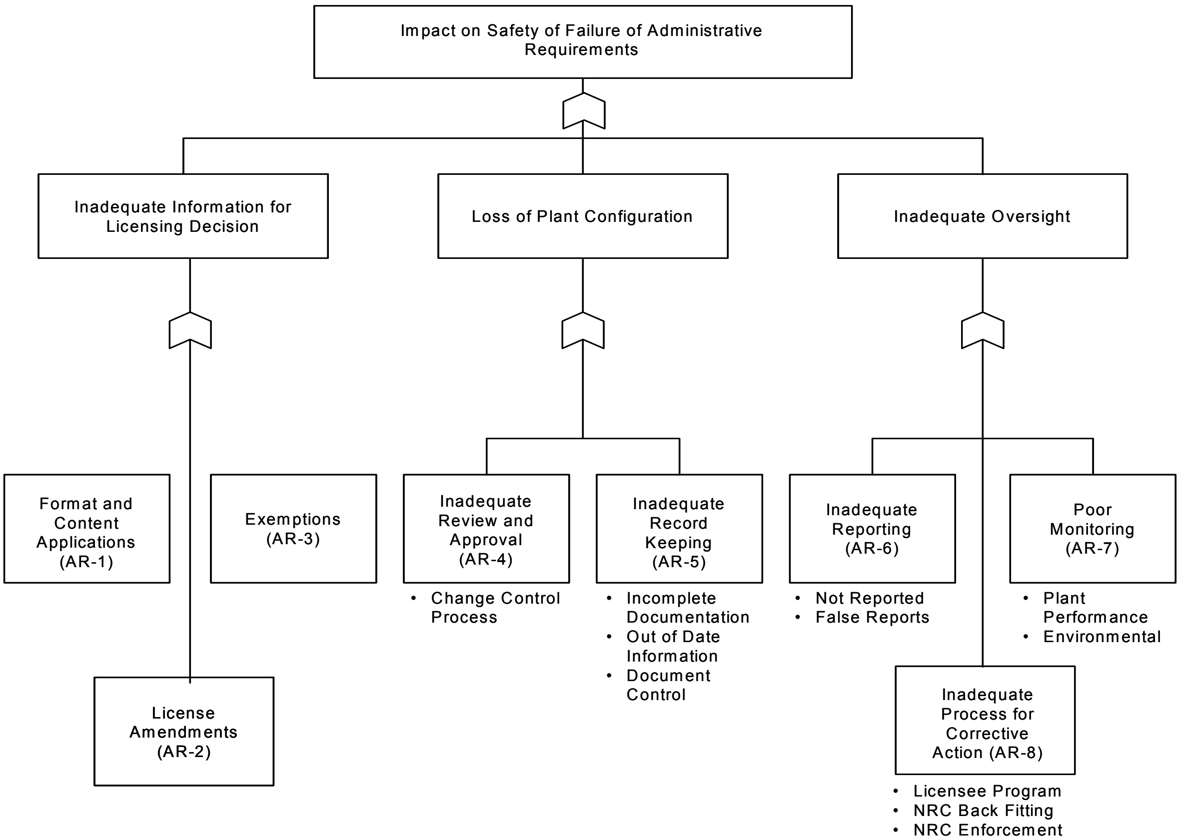

Figure G-6 Logic tree for the administrative area.

The first branch on the tree is associated with ensuring that the information necessary for licensing decisions is adequate. The licensing decisions that require information are:

- the initial application to build and operate a nuclear power plant;
- any amendments to the license after the initial OL is granted; and
- any exemptions to the regulations for initial licensing or subsequent amendments.

Each of these licensing actions requires certain types of information which the administrative requirements should address. However, due to the risk-informed and performance-based nature of the requirements, where PRA information will play a central role in establishing the safety case, the types of information required for each of these decisions will be different from that what is required under 10 CFR 50. In developing the requirements, such information needs will need to be defined.

Issues that will need to be addressed include:

- What information from the PRA should be part of the initial application, license amendment requests and exemption requests?

- What level of design, construction and operational detail needs to be submitted?

- What supporting research and development information needs to be submitted?

The second branch on the tree relates to maintaining the plant configuration up to date. This would include having a change control process that requires adequate review and approval of proposed changes and clearly identifies what changes require NRC approval and which do not. Since the regulatory structure for new plant licensing makes use of a PRA, the selection of licensing basis events (and the selection of SSCs for special treatment) may not be a one-time licensing step, carried out at the time of initial plant licensing and remaining fixed. (See Chapter 6 for a description of the selection process.) Instead, it can be expected that both the selection of LBEs and the safety classification of SSCs may change over the lifetime of the plant as operational experience and other new information add to, and reshape, the risk insights from maintaining the PRA. This potential for change in the LBEs and safety classification over time, due to maintaining the PRA up to date should be addressed. The frequency and manner of updating the PRA will have to be determined in a way that allows for regulatory stability and predictability, including compatibility with the design certification process in 10 CFR 52. Accordingly, the requirements should address a process for changes to the licensing basis. It needs to be noted that the licensing basis is also dependent on defense-in-depth, therefore, while the risk insights may change, the licensing basis may not necessarily change. Also, if the design has received design certification, the interface between the change control process and the design certification rule-making needs to be defined. To develop a change control process that accommodates the above, the following guidelines should be considered.

- The results of the PRA update should be compared to the plant licensing basis. Where changes in the licensing basis are needed to be consistent with the PRA update, they should be submitted to NRC for approval in a timely fashion.

- For plants built according to a certified design, if any of the proposed changes modify the certified (Tier 1 or Tier 2) portion of the design, a rule change to amend the certification should be processed and backfit considerations used to determine whether other plants of that same design need to make conforming changes.

- All other changes should be allowed to be made by the licensee, with appropriate documentation available for NRC audit.

Plant configuration can also be affected by inadequate record keeping. This could be due to incomplete or out of date documentation. Requirements for record keeping also need to be established.

The third branch of the tree relates to information and processes necessary for license and NRC oversight. Monitoring overall plant and SSC performance is an essential element in ensuring its PRA is maintained up to date and in the NRC oversight program. The use of risk information and importance measures can identify what SSCs should be monitored and what performance (e.g., reliability, availability) need to be achieved to remain consistent with the plant safety analysis. Limits on degraded performance can be established, based upon risk implications, that will provide margin to conditions where safety can be compromised. Thus, the requirements and their implementing guidance should call for the use of risk information and importance measures in developing, implementing and maintaining monitoring programs. This will require (1) the licensee to report certain information to NRC (e.g., events, inspection results, performance indicators, etc.) in an accurate and timely fashion, (2) the licensee to monitor certain aspects of plant performance and take corrective action (via design or operation) when necessary and (3) the NRC to initiate enforcement or backfit action if licensee performance or action is judged inadequate.

Requirements addressing what is expected from the licensee and what will trigger NRC actions should be included.

Table G-8 provides the questions resulting from Figure G-6, the answers to which identify the topics that need to be addressed by the administrative requirements.

Table G-8 Administrative areas.

Questions	Answers to Questions		
	Design	**Construction**	**Operation**
Inadequate Information for Licensing Decisions			
• What information needs to be submitted to support initial licensing? (AR-1)	• Standard format and content of applications	• Standard format and content of applications	• Standard format and content of applications
• What information needs to be submitted to support license amendments? (AR-2)	• N/A	• N/A	• Standard format and content of applications
• What information needs to be submitted to support exemptions? (AR-3)	• Standard format and content of applications	• Standard format and content of applications	• Standard format and content of applications
Loss of Plant Configuration			
• What is needed to ensure appropriate review and approval of plant changes? (AR-4)	• Change control process	• Change control process	• Change control process
• What information needs to be maintained? (AR-5)	• Identify documentation to be maintained (i.e., recordkeeping) • Documentation control process	• Identify documentation to be maintained (i.e., recordkeeping) • Documentation control process	• Identify documentation to be maintained (i.e., recordkeeping) • Documentation control process
Inadequate Oversight			
• What information is needed to support NRC oversight? (AR-6)	• N/A	• Reporting requirements	• Reporting Requirements
• What information is the licensee expected to monitor? (AR-7)	• N/A	• Inspection • Testing	• Plant performance • Environmental Releases

Table G-8 Administrative areas.

Questions	Answers to Questions		
	Design	Construction	Operation
• What corrective action processes are needed? (AR-8)	• N/A	• Licensee program • NRC enforcement	• Licensee program • NRC enforcement • NRC backfitting

N/A = Not Applicable

Table G-9 summarizes the topics which the administrative requirements should address based on the above. Other administrative requirements not related to safety will also be needed and these can be identified by a careful review of 10 CFR 50 and by including the appropriate requirement from 10 CFR 50 in the requirements.

Table G-9 Summary of administrative topics for risk-informed and performance-based requirements.

TOPIC	FRAMEWORK DESCRIPTION
• Standard format and content of application (AR-1)	• Appendix G - Section G.3
• Change control process (AR-4)	• Appendix G - Section G.3
• Record keeping (AR-5)	• Appendix G - Section G.3
• Documentation control (AR-5)	• Appendix G - Section G.3
• Reporting (AR-6)	• Appendix G - Section G.3
• Monitoring and feedback (AR-7): - plant performance - environmental releases	• Addressed under operational requirements (Section G.2.4)
• Corrective action program (AR-8)	• Appendix G - Section G.3
• Backfitting (AR-8)	• Appendix G - Section G.3
• License Amendments (AR-2)	• Appendix G - Section G.3
• Exemptions (AR-3)	• Appendix G - Section G.3
• Other legal and process items (e.g.,) - anti-trust - termination of license	• Appendix G - Section G.3 and Appendix H

APPENDIX H
APPLICABILITY OF 10 CFR 50 REQUIREMENTS

H. APPLICABILITY OF 10 CFR 50 REQUIREMENTS

As discussed in Chapter 8, the development of risk-informed and performance-based requirements should build upon previous work as much as possible. Accordingly, 10 CFR 50 needs to be reviewed to see where it would be appropriate to directly (or with modification) use its requirements in licensing using a risk-informed, performance-based licensing approach. Two main areas where this would appear to be appropriate are:

- those legal, financial and process requirements that were not identified by the technical considerations discussed in Chapter 8 and Appendix G, and

- those technical and administrative requirements that are not light water reactor (LWR) specific and are compatible with a risk-informed and performance-based approach.

An initial assessment of 10 CFR 50 has been made to identify where 10 CFR 50 requirements can be used directly, or with modification, in a risk-informed and performance-based licensing approach. The results of this assessment are shown in Tables H-1 and H-2. As can be seen from these tables, there are many 10 CFR 50 requirements and General Design Criteria that can be used directly. Also shown for the technical and administrative requirements contained in 10 CFR 50 is reference to the appropriate draft example requirements contained in Appendix J of this NUREG.

Table H-1 Initial assessment of applicability of 10 CFR 50 requirements.

10 CFR Part 50		Applicability
General Provisions		
50.1	Basis, Purpose, and Procedures Applicable	• Use 10 CFR 50 requirement.
50.2	Definitions	• Use some 10 CFR 50 definitions. • Use Framework for others needed.
50.3	Interpretations (Assigns legal interpretation authority to NRC General Counsel)	• Use 10 CFR 50 requirement.
50.4	Written Communications	• Use 10 CFR 50 requirement.
50.5	Deliberate Misconduct	• Use 10 CFR 50 requirement.
50.7	Employment Protection (Protects employees of licensees against discrimination and retribution for providing information to NRC, Congress, etc.)	• Use 10 CFR 50 requirement.
50.8	Information Collection Requirements: OMB Approval	• Use 10 CFR 50 requirement.
50.9	Completeness and Accuracy of Information	• Use 10 CFR 50 requirement.

H. Applicability of 10 CFR 50

Table H-1 Initial assessment of applicability of 10 CFR 50 requirements.

10 CFR Part 50		Applicability
Requirement of License, Exceptions		
50.10	License Required (Establishes license requirement Identifies facilities which are required to obtain an NRC license and which are not)	• Use 10 CFR 50 requirement.
50.11	Exceptions and Exemptions from License Requirements	• Use 10 CFR 50 requirement.
50.12	Specific Exemptions	• Use 10 CFR 50 requirement with modification to be risk-informed consistent with Framework risk criteria. See Framework draft Administrative Requirement #9.
50.13	Attacks by Enemies of the U.S.	• Use 10 CFR 50 requirement.
Clarification and Description of Licenses		
50.20	Two Classes of Licenses	• Use 10 CFR 50 requirement for Class 103 license.
50.21	Class 104 License (Medical facility and device manufacturer licenses)	• Not applicable to NPP licensing.
50.22	Class 103 License for Commercial and Industrial Facilities	• Use 10 CFR 50 requirement.
50.23	Construction Permits	• Use 10 CFR 50 requirement.
Applications for Licenses, Forms, Contents, Ineligibility of Certain Applications		
50.30	Filing of Application for License: Oath or Affirmation	• Use 10 CFR 50 requirement.
50.31	Combining Applications	• Use 10 CFR 50 requirement.
50.32	Elimination of Repetition	• Use 10 CFR 50 requirement.
50.33	Contents of Application (General Information)	• Replace with Framework draft Administrative Requirement #1.
50.33a	Information Requested by the Attorney General for Antitrust Review	• Use 10 CFR 50 requirement.
50.34	Contents of Application (Technical Requirements)	• Use 10 CFR 50 requirement except 50.34(f) and (h). 50.34(g) addressed in Framework draft Design Requirement #33. Also, see Framework draft Administrative Requirement #1.

Table H-1 Initial assessment of applicability of 10 CFR 50 requirements.

10 CFR Part 50		Applicability
50.34a	Design Objective Requirements for Equipment to Control the Release of Radioactive Material	• Use modified 10 CFR 50 requirement. See Framework draft Design Requirements #40 and #41 fro control of radioactive releases and Design Requirement #8 for the siting source term.
50.35	Issuance of Construction Permits	• Use 10 CFR 50 requirement.
50.36	Technical Specifications	• Use 10 CFR 50 requirement, supplement for use of risk information. See Framework draft Operating Requirement #12.
50.36a	Technical Specifications on Effluents from Nuclear Power Plants	• Use 10 CFR 50 requirement.
50.36b	Environmental Conditions	• Use 10 CFR 50 requirement.
50.37	Agreement Limiting Access to Classified Information	• Use 10 CFR 50 requirement.
50.38	Foreign Corporation or Individual Restriction	• Use 10 CFR 50 requirement.
50.39	Public Inspection of License Requirement	• Use 10 CFR 50 requirement.
Standards for Licenses and Construction Permits		
50.40	Common Standards (Part 51 compliance, Requirement for licensee to be technically and financially qualified, Operation does not infringe on defense or public health)	• Use 10 CFR 50 requirement.
50.41	Additional Standards for Class 104 Licenses	• Not applicable to licensing NPPs.
50.42	Additional Standards for Class 103 Licenses (Usefulness Requirement Antitrust Restriction Open Communication Requirement)	• Use 10 CFR 50 requirement.
50.43	Additional Standards and Provision Affecting Class 103 Licenses for Commercial Power Plants (NRC is required to inform of applications for licenses: 1. State and Local Authorities 2. Public via Federal Register 3. Other Cognizant Federal Agencies)	• Use 10 CFR 50 requirement.
50.44	Combustible Gas Control for Nuclear Power Reactors (BWR Containment Specifications Equipment Survivability Specifications Monitoring Requirements Analysis Requirements Requirement for Future Applicability)	• Partially applicable (use 10 CFR 50.44(a) and (d) words for non-LWR). See Framework draft Design Requirement #33.

H. Applicability of 10 CFR 50

Table H-1 Initial assessment of applicability of 10 CFR 50 requirements.

10 CFR Part 50		Applicability
50.45	Standards for Construction Permits	• Use 10 CFR 50 requirement.
50.46	Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Reactors	• Not applicable - LWR specific.
50.46a	Acceptance Criteria for Reactor Coolant System Venting System	• Not applicable - LWR specific.
50.47	Emergency Plans	• Use 10 CFR 50 requirement. Modify to ensure consistency with licensing analysis. See Framework draft Operations Requirement #13.
50.48	Fire Protection (General Description Specific Hazard Detection and Suppression Systems Administrative Controls Risk-informed Analysis Requirement)	• Use 10 CFR 50 GDC #3 for deterministic requirements modified to include liquid metal and graphite fire protection. Modify to use risk information to identify fire scenarios to be considered. Use 50.48 provisions as guidance, including a fire protection plan and allowing the use of a performance based, risk-informed approach. See Framework draft Design Requirement #26.
50.49	Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants	• Use Framework draft Design Requirement #6.
Issuance, Limitations, and Conditions of Licenses and Construction Permits		
50.50	Issuance of Licenses and Construction Permits	• Use of 10 CFR 50 requirement.
50.51	Continuation of License (Sets time limits on term of license Holds licensee responsible for site after permanent shutdown)	• Use 10 CFR 50 requirement.
50.52	Combining Licenses	• Use 10 CFR 50 requirement.
50.53	Jurisdictional Limitations	• Use 10 CFR 50 requirement.

Table H-1 Initial assessment of applicability of 10 CFR 50 requirements.

10 CFR Part 50		Applicability
50.54	Conditions of Licenses (Organizational Description Nuclear Material Control Restrictions Emergency and War Control Revocation, Suspension, Modification and Amendment Provisions Information Request Rules Antitrust Limitations Personnel Control Requirements Personnel Requalification Plans Licensed Operator Staffing Requirements Safeguards Contingency Plan Requirements Emergency Plan Requirements Physical Security Safeguards and Contingency Plan Requirements Insurance Requirements Clean Up Plan Requirements Restart and Decommissioning Authority Safety Deviation Allowance Fuel Storage Following Decommissioning Bankruptcy Notification Requirements National Security Technical Specification Allowance)	• Use 10 CFR 50 requirement except for fuel reprocessing and research reactor requirements. • For staffing requirements, use Framework draft Operating Requirement #7.
50.55	Conditions of Construction Permits (Failure and defect information and correction plan Time Limits for correction of defects and reporting requirements for failure to correct Defines conditions for required reports Report content requirements Directives of where to deliver reports Quality Assurance requirements SAR change reporting requirements)	• Use 10 CFR 50 requirement.
50.55a	Codes and Standards (Identifies acceptable Codes and Standards Sets Minimum Requirements for Specific Structural Materials)	• Needs to be modified to be technology-neutral. See Framework draft Design Requirement #9.
50.56	License Conversion	• Use 10 CFR 50 requirement.
50.57	Issuance of Operating License (Requirements to issue an operating license)	• Use 10 CFR 50 requirement.
50.58	Hearings and Report of the ACRS	• Use 10 CFR 50 requirement.
50.59	Changes, Tests, and Experiments	• Replaced by Framework draft Administrative Requirement #2.
50.60	Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation	• LWR specific. Use as guidance for LWRs.
50.61	Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events	• LWR specific. Use as guidance for LWRs

H. Applicability of 10 CFR 50

Table H-1 Initial assessment of applicability of 10 CFR 50 requirements.

10 CFR Part 50		Applicability
50.62	Requirements for Reduction of Risk from ATWS Events for Light Water Cooled Nuclear Power Plants	• Not applicable. LWR specific.
50.63	Loss of All Alternating Current Power	• Not applicable. LWR specific.
50.64	Limitation on the Use of Highly Enriched Uranium (HEU) in Domestic Non-power Reactors	• Not applicable.
50.65	Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	• Use 10 CFR 50 requirement with modification to conform to Framework. See framework draft Operating Requirement #2.
50.66	Requirements for Thermal Annealing of the Reactor Pressure Vessel	• Not applicable - LWR specific.
50.67	Accident Source Term (Defines applicability and requirements for existing LWRs wanting a license amendment to use a revised source term. Sets radiation exposure limits within defined areas around the plant)	• Not applicable.
50.68	Criticality Accident Requirements (Limits Concentrations of Storage Fuel Rods Limits Credit Taken for Moderation Limits Fuel Rod U-235 Purity)	• Not included since Framework does not cover fuel handling or storage.
50.69	Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants	• Use approach defined in the Framework. See Framework draft Design Requirement #5.
Inspections, Records, Reports, Notifications		
50.70	Inspections (Requires licensees to submit to routine inspection Requires licensee to provide reasonable space accommodation to inspectors)	• Use 10 CFR 50 requirement.
50.71	Maintenance of Records, Making Reports (Defines items which must be kept as records Sets requirements for quality of records Sets reporting periods for specific records)	• Use 10 CFR 50 requirement, supplemented to address record keeping of risk information. See Framework draft Administrative Requirement #3.
50.72	Immediate Notification Requirements for Operating Nuclear Power Reactors (Defines events and conditions which must be reported to the NRC Sets time limits for reporting Sets follow-up requirements)	• Use 10 CFR 50 requirement. Assess definition of types of events to be reported to ensure technology-neutral.

Table H-1 Initial assessment of applicability of 10 CFR 50 requirements.

10 CFR Part 50		Applicability
50.73	Licensee Event Report System (Defines events and conditions which must be reported via LER Sets time limits for reporting Sets follow-up requirements Sets content requirements for LER)	• Use 10 CFR 50 requirement. See Framework draft Administrative Requirement #5.
50.74	Notification of Change in Operator or Senior Operator Status Reporting Requirement	• Use 10 CFR 50 requirement.
50.75	Reporting and Record Keeping for Decommissioning Planning (Establishes reasonable assurance that funds will be available for decommissioning process)	• Use 10 CFR 50 requirement.
50.76	Licensee Change of Status, Financial Qualifications (Requires licensee to inform NRC 75 days before ceasing to exist)	• Use 10 CFR 50 requirement.
US/IAEA Safeguards Agreement		
50.78	Installation information and verification (Requires licensees to submit to IAEA inspection when directed by NRC)	• Use 10 CFR 50 requirement.
Transfers of Licenses, Creditors Rights, Surrender of Licenses		
50.80	Transfer of Licenses (Requires NRC to consent to license transfer to qualified licenses Defines requirements for new licensee to receive license)	• Use 10 CFR 50 requirement.
50.81	Creditor Regulations (Sets conditions under which a creditor may posses a lien on a utilization and production facility)	• Use 10 CFR 50 requirement.
50.82	Termination of License (Sets time limits for notifying NRC of intention to terminate a license Sets time limit for decommissioning once intention is announced Sets Funding Requirements for Decommissioning Sets Radiation Survey Requirements)	• Use 10 CFR 50 requirement.
50.83	Release of Part of a Power Reactor Facility or Site for Unrestricted Use (Defines planning and Notification Requirements Sets Radiation Exposure Limits Sets Inspection Requirements)	• Use 10 CFR 50 requirement.

H. Applicability of 10 CFR 50

Table H-1 Initial assessment of applicability of 10 CFR 50 requirements.

10 CFR Part 50		Applicability
Amendment of License or Construction Permit at Request of Holder		
50.90	Application for Amendment of License or Construction Permit	• Use 10 CFR 50 requirement.
50.91	Notice of Public Comment and State Consultation (Time requirements for announcing and holding public comment meetings Sets requirements for NRC to consult and inform state officials of license changes)	• Use 10 CFR 50 requirement.
50.92	Issuance of Amendment (Identifies issues which are to be considered when evaluating a request for a license change)	• Use 10 CFR 50 requirement except supplement 50.92(c) to develop a risk-informed alternative to allow the use of risk information. See Framework draft Administrative Requirement #8.
Revocation, Suspension, Modification, Amendment of Licenses and Construction Permits, Emergency Operations by the Commission		
50.100	Revocation, Suspension, and Modification of Licenses and Construction Permits for Cause	• Use 10 CFR 50 requirement.
50.101	Retaking Possession of Special Nuclear Material	• Use 10 CFR 50 requirement.
50.102	Commission Orders for Operation After Revocation	• Use 10 CFR 50 requirement.
50.103	Suspension and Operation in War or National Emergency	• Use 10 CFR 50 requirement.
Backfitting		
50.109	Backfitting	• Use 10 CFR 50 requirement. See Framework draft Administrative Requirement #7. Guidance draft on non-LWR risk metrics will be needed.
Enforcement		
50.110	Violations (Grants power to NRC to seek injunction for violations of Atomic Energy Act, NRC regulations, or violations of License)	• Use 10 CFR 50 requirement.
50.111	Criminal Penalties	• Use 10 CFR 50 requirement.
50.120	Training and Qualification of Nuclear Power Plant Personnel (Requirement to have a training program Training program standards Personnel required to receive training Training review and update requirements)	• Use 10 CFR 50 requirement. See Framework draft Operations Requirements #3 and #4.

Table H-1 Initial assessment of applicability of 10 CFR 50 requirements.

10 CFR Part 50		Applicability
Appendices		
Appendix A:	General Design Criteria for Nuclear Power Plants	• See Table H-2.
Appendix B:	Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants	• Use 10 CFR 50 requirement. See Framework draft Common Requirement #1.
Appendix C:	A Guide for the Financial Data and Related Information Required to Establish Financial Qualifications for Facility Construction Permits	• Use 10 CFR 50 requirement.
Appendix E:	Emergency Planning and Preparedness for Production and Utilization Facilities	• Use 10 CFR 50 requirement. See Framework draft Operating Requirement #13.
Appendix F:	Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities	• Not applicable to NPPs.
Appendix G:	Fracture Toughness Requirements	• Use as guidance for LWRs. Need to develop non-LWR guidance. See Framework draft Design Requirement #35.
Appendix H:	Reactor Vessel Material Surveillance Program Requirements	• Use 10 CFR 50, Appendix H in RG. See Framework draft Operational Requirement #9.
Appendix I:	Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion “As Low as is Reasonably Achievable” for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents	• Modify to be applicable to non-LWRs and also use as guidance for LWRs. See Framework draft Design Requirement #39 and Operational Requirement #1.
Appendix J:	Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors	• Not applicable - LWR specific.
Appendix K:	ECCS Evaluation Models	• Not applicable - LWR specific.
Appendix L:	Information Requested by the Attorney General for Antitrust Review of Facility Construction Permits and Initial Operating Licenses	• Use 10 CFR 50 requirement.
Appendix M:	Standardization of Design; Manufacture of Nuclear Power Reactors; Construction and Operation of Nuclear Power Reactors Manufactured Pursuant To Commission License	• Not needed in view of 10 CFR 52.
Appendix N:	Standardization of Nuclear Power Plant Designs; Licenses to Construct and Operate Nuclear Power Reactors of Duplicate Design at Multiple Sites	• Not needed in view of 10 CFR 52.

Table H-1 Initial assessment of applicability of 10 CFR 50 requirements.

10 CFR Part 50		Applicability
Appendix O:	Standardization of Design; Staff Review of Standard Designs	• Not needed in view of 10 CFR 52.
Appendix Q:	Pre-Application Early Review of Site Suitability Issues	• Use 10 CFR 50 requirement.
Appendix R:	Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979	• Not applicable. Only applies to plants prior to 1979.
Appendix S:	Earthquake Engineering Criteria for Nuclear Power Plants	• Use 10 CFR 50 requirement as guidance, supplemented to select SSE using risk-information. See Framework draft Design Requirements #2 and #11.

The GDCs have been reviewed to assess their relationship to the Framework with respect to:

- can the GDC be used directly (or with minor modification),
- is the intent of the GDC covered by the draft requirements contained in Appendix J,
- is the GDC not applicable due to its LWR specific nature, or
- are there topics not covered by the GDCs that the Framework addresses.

The following table summarizes the assessment of the 55 GDCs with respect to the above.

Overall, 16 GDCs are used directly (some with modifications to use Framework terminology or remove LWR specific parts) and 25 additional GDCs have their intent addressed by the draft example requirements in Appendix J of the Framework. This leaves 14 GDCs that are not applicable or outside the scope of the Framework. Appendix J contains the draft example requirements cross referenced to the GDC used. Also listed at the end of Table H-2 are the topics not addressed by the current GDCs but would need to be addressed in a risk-informed and performance-based licensing approach.

Table H-2 10 CFR 50, Appendix A - General Design Criteria (GDC)

GDC		Applicability
I)	**Overall Requirements**	
1)	Quality standards and records	Used directly in Framework with minor modification to use framework terminology (i.e., safety significant in lieu of important to safety). See Framework draft Common Requirement #1.
2)	Design basis for protection against natural phenomena	Used directly in Framework with modification to allow use of risk assessment to select natural phenomena to be considered in the design (see Framework draft Design Requirement #11).

Table H-2 10 CFR 50, Appendix A - General Design Criteria (GDC)

GDC	Applicability
3) Fire protection	Used directly in Framework with modification to address liquid metal and graphite fire protection and use risk assessment to identify fire scenarios which must be considered in the design as LBEs, provide a fire protection plan and allow for the use of a performance based, risk-informed approach, as defined in 10 CFR 50.48. (See Framework draft Design Requirement #26).
4) Environmental and dynamic effects design bases	Used directly in Framework with minor modification to use Framework terminology (i.e., safety significant in lieu of important to safety). See Framework draft Design Requirement #12.
5) Sharing of systems, structures and components	Used directly in Framework with minor modification to use Framework terminology (i.e., safety significant in lieu of important to safety). See Framework draft Design Requirement #13.
II) Protection by Multiple Fission Product Barriers	
10) Reactor design	Addressed by Framework draft Design Requirement #14.
11) Reactor inherent protection	Addressed by Framework draft Design Requirement #24.
12) Suppression of reactor power oscillations	Addressed by Framework draft Design Requirement #24.
13) Instrumentation and control	Used directly in Framework and supplemented by words on EQ and digital I and C. See Framework draft Design Requirement #38.
14) Reactor coolant pressure boundary	Used directly in Framework. See Framework draft Design Requirement #36.
15) Reactor coolant system design	Addressed by Framework draft Design Requirement #3.
16) Containment design	Addressed by Framework draft Design Requirement #16.
17) Electric power systems	Partially included in Framework. Requirements for electric power systems are only specified for electric power systems necessary to accomplish safety significant functions (e.g., respond to LBEs, provide instrumentation and control). See Framework draft Design Requirement #19.
18) Inspection and testing of electric power systems	Addressed in Framework draft Design Requirement #19.
19) Control room	Addressed in Framework draft Design Requirements #27 and #28.
III) Protection and Reactivity Control Systems	
20) Protection system functions	Addressed by Framework draft Design Requirements #14.
21) Protection systems reliability and testability	Addressed by Framework draft Design Requirements #14 and #30.
22) Protection system independence	Addressed by Framework draft Design Requirement #14.
23) Protection systems failure modes	Used directly in Framework draft Design Requirement #14.

Table H-2 10 CFR 50, Appendix A - General Design Criteria (GDC)

GDC	Applicability
24) Separation of protection and control systems	Not included since the Framework draft Design Requirement on protection systems (#14) will require a highly reliable reactor shutdown function, including consideration of interactions with control systems. Basis for an exemption is in Framework Appendix G (G.2.3).
25) Protection system requirements for reactivity control malfunctions	Addressed by Framework draft Design Requirement #14. PRA will determine which reactivity control malfunctions must be addressed in the design.
26) Reactivity control system redundancy and capability	Intent included in the draft Framework Design Requirements on reactor shutdown (#14) and response to LBEs (#3) cover reactivity control, shutdown and protection of the fuel. Basis for an exemption is in Framework Appendix G (G.2.3).
27) Combined reactivity control systems capability	Addressed by Framework draft Design Requirement #14. Stuck control rod requirement not included because some designs may not use control rods for shutdown and the requirement for two independent, redundant and diverse systems address the single failure.
28) Reactivity limits	Addressed by Framework draft Design Requirement #14.
29) Protection against anticipated operational occurrences	Addressed by Framework draft Design Requirement #3.
IV) Fluid Systems	
30) Quality of reactor coolant pressure boundary	Addressed by Framework draft Design Requirement #36.
31) Fracture prevention of reactor coolant pressure boundary	Used directly in the Framework. See Framework draft Design Requirement #35.
32) Inspection of reactor coolant pressure boundary	Addressed by Framework draft Design Requirement #35.
33) Reactor coolant makeup	Not included in the Framework due to its LWR specific nature. Design will determine need for coolant makeup.
34) Residual heat removal	Addressed by Framework draft Design Requirement #14. Single failure requirement met by requiring redundant, diverse and independent means of decay heat removal.
35) Emergency core cooling	Not included. LWR specific. LBEs selected using the PRA will be used to determine safety system requirements needed.
36) Inspection of ECCS	Not included. LWR specific. LBEs selected using the PRA will be used to determine safety system requirements needed.
37) Testing of ECCS	Not included. LWR specific. LBEs selected using the PRA will be used to determine safety system requirements needed.
38) Containment heat removal	Not included. LWR specific. LBEs selected using the PRA will be used to determine safety system requirements needed
39) Inspection of containment heat removal systems	Not included. LWR specific. LBEs selected using the PRA will be used to determine safety system requirements needed.

Table H-2 10 CFR 50, Appendix A - General Design Criteria (GDC)

GDC	Applicability
40) Testing of containment heat removal system	Not included. LWR specific. LBEs selected using the PRA will be used to determine safety system requirements needed.
41) Containment atmosphere cleanup	Used directly in the Framework, except for electric power requirements. LBEs, based upon the PRA, will determine electric power needs. Also, inspection and testing requirements added. See Framework draft Design Requirement #17.
42) Inspection of containment atmosphere cleanup systems	Addressed in Framework draft Design Requirement #17.
43) Testing of containment atmosphere cleanup systems	Addressed in Framework draft Design Requirement #17.
44) Cooling water	Not included - LWR specific.
45) Inspection of cooling water system	Not included - LWR specific.
46) Testing of cooling water system	Not included - LWR specific.
V) Reactor Containment	
50) Containment design basis	Addressed by Framework draft Design Requirement #16.
51) Fracture prevention of containment pressure boundary	Used directly in Framework. See Framework draft Design Requirement #18.
52) Capability for containment leak rate testing	Addressed by Framework draft Design Requirement #16.
53) Provisions for containment testing and inspections	Addressed by Framework draft Design Requirement #16.
54) Piping systems penetrating containment	Used directly in the Framework as draft Design Requirement #20.
55) Reactor coolant pressure boundary penetrating containment	Used directly in the Framework as draft Design Requirement #20.
56) Primary containment isolation	Addressed in Framework draft Design Requirement #20.
57) Closed system isolation valves	Used directly in the Framework as draft Design Requirement #21.
VI) Fuel and Radioactivity Control	
60) Control of radioactive material releases to the environment	Used directly in the Framework as draft Design Requirement #40.
61) Fuel storage and handling and radioactivity control	Not included in the Framework since the scope of the Framework does not include fuel handling and storage.
62) Prevention of criticality in fuel storage and handling	Not included in the Framework since the scope of the Framework does not include fuel handling and storage.
63) Monitoring fuel and waste storage	Not included in the Framework since the scope of the Framework does not include fuel handling and storage.
64) Monitoring radioactive releases	Used directly in the Framework (minus LWR specific words). See Framework draft Design Requirement #41.

In addition to the Framework draft example requirements referenced in the above tables, the Framework contains additional draft example requirements beyond 10 CFR 50 requirements to address the use of risk information in establishing the licensing basis, to address issues associated with non-LWRs, and to include other good engineering practices. These additional draft example requirements (identified by their Framework draft requirement number) are discussed in Appendix J and indicated below in Table H-3.

Table H-3 Example draft requirements beyond 10 CFR 50.

Req #	Requirement
Additional draft general requirements identified in Appendix J, Table J-1	
2	PRA Scope and Technical Acceptability
3	Use of Risk Information
4	Integration of Safety, Security and Preparedness
Additional draft security related requirements identified in Appendix J, Table J-3	
2	Security Performance Standards
Additional draft design requirements identified in Appendix J, Table J-5	
1	Plant Risk
2	Criteria for Selection of the Licensing Basis Events (LBEs)
4	Initiating Event Severity
6	Equipment Qualification
7	Licensing Analysis
10	Materials Qualification
15	Barriers to the Release of Radioactive Material
22	Vulnerability to a Single Human Action or Hardware Failure
23	Plant Aging and Degradation
25	Human Factors/Human Machine Interface
29	Reactor Core Flow Blockage and Bypass Prevention
30	Reliability and Availability
31	Research and Development
32	Use of Prototype Testing
34	Energetic Reaction Control
37	Reactor Coolant Activity Monitoring and Cleanup
39	Protection of Operating Staff
42	Qualified Analysis Tools
Additional draft construction requirements identified in Appendix J, Table J-7	
1	Use Accepted Codes, Standards, Practices
2	Security During Construction/Fabrication
3	NDE During Construction/Fabrication
4	Inspection During Construction/Fabrication
5	Testing During Construction/Fabrication

Table H-3 Example draft requirements beyond 10 CFR 50.

Req #	Requirement
Additional draft operational requirements identified in Appendix J, Table	
5	Use of Procedures
6	Use of Simulators
8	Aging Management Programs
10	ISI
11	Testing
14	Monitoring and Feedback
15	Work and Configuration Control
16	Maintenance of the PRA
17	Fuel and Replacement Part Quality
Additional draft administrative requirements identified in Appendix J, Table J-11	
2	Change Control Process
4	Document Control
6	Corrective Action Program

APPENDIX I
GUIDANCE FOR PERFORMANCE-BASED REQUIREMENTS

I. GUIDANCE FOR PERFORMANCE-BASED REQUIREMENTS

The following guidance provides a step-by-step approach to formulate a regulatory requirement that is focused on accomplishing a defined objective which corresponds to the result expected from performance-based regulation (see Chapter 5). An example of a typical performance objective is maintaining cladding integrity. In the conventional regulatory approach this objective is considered to be accomplished through a prescriptive approach of limiting cladding temperature and oxidation conditions to 2200 F and 17% respectively. In a performance-based approach, a different set of criteria, perhaps using a combination of qualitative and quantitative may be found to better fulfill the high-level guidelines.

I.1 Step 1 – Identifying the Performance Objective and its Context

Purpose – To define a performance objective for the structures, systems and components (SSC) and/or operator actions in such a way that one or more performance measures and criteria can be proposed for consideration.

Step 1a: What is the topic area with which the performance objective is associated?

This question is likely addressed during the review under Chapter 8, where the risk objectives are classified as falling under design, construction and operation. Additionally, from a regulatory standpoint, the objectives may fall under the categories public risk, worker risk and environmental risk. There could be significant differences in the information gathering and stakeholder identification depending on what is being addressed. A well defined performance objective is a pre-requisite for an effective performance measure. If a single performance objective will not be effective for establishing the requirements for the SSC, an Objectives Hierarchy (see NUREG/BR-0303) may need to be prepared.

Step 1b: Which of the NRC's performance goals does the performance objective address?

Clarifying the performance goal also improves the clarity with which NRC decision preferences may be incorporated in the consideration of performance measures or criteria. From the NRC's Strategic Plan (NUREG-1614, Vol. 3, August 2004) the two performance goals likely to be involved are "*Ensure protection of public health and safety and the environment*" and "*Ensure that NRC actions are effective, efficient, realistic, and timely*."

Step 1c: What are the expected outcomes and results from successful performance relative to the objective?

In general, the expected outcome is that the SSC performs its intended safety function adequately, and that the performance can be appropriately verified through regulatory oversight. In addition, this question addresses which part of the regulatory structure is appropriate for implementing the objective. In general, a regulation in the Code of Federal Regulations is likely to address higher level goals or objectives. Guidance documents are more likely to be directed at detailed or component level objectives.

I.2 Step 2 – Identifying the Safety Functions

Purpose – To identify the safety functions and systems that affect the performance objective (directly or indirectly).

Step 2a: What are the safety functions or concepts that can impact the performance objective?

The objective of this inquiry is to identify the most important functions. The probabilistic risk assessment (PRA) should be of help in this effort. However, some aspects of system performance may not be modeled in the PRA. Such aspects are generally those that cannot be easily quantified and must be considered qualitatively. It is key that the identification of important functions focus on successful outcomes rather than make assumptions because of inadequacies of the PRA model. In addition, consideration should be given to other aspects of the context which may include expected outcomes being fulfilled by other SSCs.

Step 2b: What equipment/systems/procedures are necessary to satisfy the safety function?

This addresses the technical evaluation that establishes the range of particular SSCs or support systems to be considered; for example, instrumentation, siting, safety conscious work environment, etc. Again, the evaluation can take advantage of the PRA where the modeling is adequate. Often, qualitative factors coupled with expert judgement can be as or more reliable than quantitative models that are not supported by sufficient data. This is especially the case when data from operating experience exists, even if the data is from a related but different industry.

Step 2c: What level of safety (based on appropriate metrics) is required to meet the performance objective?

This addresses the required level of safety that should have been addressed in the Chapter 6 evaluation. For example, the required level of safety for an accident within containment might be one that meets the objective of reducing, to an acceptable level, the risk of early containment failure. Hence, the metric in this case is the conditional containment failure probability. Another example might be that the required level of safety is to maintain at an acceptable level the core damage risk associated with certain configurations typical of specific modes of operations. Again, qualitative evaluations supported by expert judgement or operational data may be required.

I.3 Step 3 – Identifying Safety Margins

Purpose – To evaluate margins and identify performance measures (if any) that satisfy the performance objectives.

Step 3a: How much safety margin is available, and how robust is it, for performance monitoring to provide a basis for granting licensee flexibility?

The generic definition of a “margin” is that it is an expression of a difference between two system states. When the two states are associated with different levels of safety as reflected in the above evaluations related to outcomes, the “margin” becomes a safety margin. For regulatory purposes, the margin that is sought to be maintained is expressed by the first of these being the expected state and the other is one where a regulatory concern exists. The state of regulatory concern can be drawn from the frequency-consequence curve dealt with in Chapter 6 and the margin discussion in Chapter 6.

“Robustness” of a safety margin means that the margin between two performance levels is significantly greater than uncertainty and normal variability in performance. If this condition is met, a very low probability exists of the performance parameter crossing a set limit, unless performance changes in a very significant way. In any case, wherever there is substantial uncertainty, achieving robustness requires that nominal performance levels be set more conservatively than when there is less uncertainty. Depending on the situation, uncertainty can be assessed using explicit models (e.g., PRAs), expert judgment, or actuarial methods based on operating experience.

The identification of performance measures (natural, constructed or combination) begins as a search process within the overall context of the performance objective. It is likely to involve iteration through the steps in this guidance as well as consideration of the factors that were involved in the application of the viability guidelines. The flexibility aspects should include operational flexibility as well as the means to fulfill regulatory responsibilities.

Step3b: What observable characteristics, quantitative and qualitative, exist within the safety functions identified in Step 2?

For example, observable characteristics may come from the results of periodic servicing, testing, and calibration of certain instruments. The operating margin would be based on a comparison between these results and the target values established under a maintenance program. Another example would be observations based on verification (through testing) of design margins of structures.

Step 3c: Can the contemplated constructed measures provide qualitative expressions capable of observation with reasonable objectivity?

As explained in NUREG/BR-0303, natural measures are preferred, but appropriate constructed measures may also prove adequate with proper consideration given to verification and validation. In some cases, a binary constructed measure might well suffice where the measure reflects a positive or negative response to a question such as , “Does a particular attribute exist?”

I.4 Step 4 – Selecting Performance Measures and Criteria

Purpose – To select a complement of performance measures and objective criteria (if possible) that both satisfy the viability guidelines and accomplish the performance objective.

Step 4a: Can the identified observable characteristics, together with objective criteria, provide measures of safety performance and the opportunity to take corrective action if performance is lacking?

This step is a part of the search process. Many technically significant performance objectives will require engineering judgement for exploring qualitative and/or quantitative measures while keeping in mind operational (or other) constraints. Measures of safety performance considered as candidates should be associated with the desired outcomes as directly as possible. Sometimes, it may prove quite effective to use proxy measures. For example, if the accomplishment of a performance objective calls for an analysis, the cost of the analysis may be one of the measures considered as a proxy for efficiency of obtaining the outcome.

Another of the highly desirable features of a good performance measure is that it should be identified at as high a level as practicable. If this feature is not sought, all systems and sub-systems involved in, say, risk-significant configurations might have been targeted for monitoring. The management of risk when various configurations are being considered may include monitoring strategies that target all systems and sub-systems, or a higher-level measure that may prove to be simpler, but as effective. The process of searching for parameters at a high level directs the staff's attention to more cost-effective possibilities.

Step 4b: Can objective criteria be developed that are indicative of performance and that permit corrective action?

The search for threshold criteria that rely as little as possible on subjectivity is the next step in the search process. Parametric sensitivity analyses may help establish that the selected threshold is not in a region of highly unstable or non-linear behavior (so-called "cliff effects"). Some performance objectives are likely to be more difficult in the establishment of objective criteria that are indicative of performance than others. Also, selecting performance measures that permit sufficient time for corrective action may require probabilistic considerations and expert elicitation.

Step 4c: Is flexibility (for NRC and licensees) available consistent with level of margin?

The approach of setting criteria at as high a level as practicable can allow more flexibility. The benefits of flexibility must be balanced against assurance of opportunity to take appropriate corrective action and practicality of regulatory oversight. The basic principle involved is that more flexibility can be justified by higher levels and robustness of safety margin. Again, an iterative approach may be most suitable for optimum results. This is because questions of margin, corrective action, and flexibility strongly interact with one another. Strong linkages can exist between observable characteristics chosen as the performance measures to be used in a performance-based approach and the assessment of margin based on criteria applied to these parameters. For example, in the area of quality assurance, the quality of emergency backup power provided by a diesel generator would not necessarily be well-reflected just by the criteria that are applied to each component part of the diesel generator. Even if very strict quality criteria are applied to each of the component parts, the overall diesel generator performance may not meet regulatory standards. On the other hand, a diesel generator could adequately meet performance standards even if the component parts are only commercial grade.

I.5 Step 5 – Formulating a Performance-Based Requirement

Purpose – To determine the appropriate implementation of a performance-based approach within the regulatory structure.

Step 5a: Does the performance-based regulatory requirement provide necessary and sufficient coverage for the performance objective?

One of the important elements of coverage is consideration of defense-in-depth. As described in Chapters 4 and 8, NRC's defense-in-depth philosophy includes consideration of "prevention" and "mitigation" strategies which should operate in proper balance. Such considerations may require the use of more complex approaches based on decision theoretic concepts (also described in NUREG/BR-0303).

Step 5b: Of the performance parameters selected in Step 4, which of them requires that a prescriptive approach be used to meet regulatory needs? Can a combination of performance-based and prescriptive measures be implemented such that the resolution of the regulatory issue is as performance-based as possible?

The search process for performance measures and criteria may reveal various permutations and combinations of prescriptive, less-prescriptive and performance-based strategies for individual components or sub-systems. In some cases, specific prescriptive elements can be incorporated into a less prescriptive regulatory approach. The regulatory structure permits inclusion of prescriptive elements through Technical Specification or License Condition provisions.

Step 5c: Has the regulatory alternative been considered for implementation within each of the levels of the regulatory structure so that an optimum level is proposed?

For example, a prescribed parameter can be included in a Technical Specification or other license condition. It may be possible to provide flexibility in operation for parameters that do not have to be strictly controlled. Also, consideration should be given to incentives for licensees to increase the likelihood of improved safety outcomes.

Step 5d: Are licensees' incentives appropriately aligned, considering the overall complement of performance measures, criteria, the implementation, and the regulatory structure as a whole?

Licensees' flexibility can be coupled with positive and negative incentives. Examples of positive incentives occur when licensees may be able to reduce costs of operation if they meet specified levels of safety or trends in safety of operation. Examples of negative incentives occur when the enforcement policy may cause undesired consequences for the licensee when levels of safety or trends in safety are unfavorable.

Regulation that is based on sampling licensee performance needs to be designed with care, in order to avoid incentivizing performance in one important area at the expense of another, with a net adverse outcome. As a hypothetical example, regulation that sought only to minimize the unavailability of components might create an incentive to reduce maintenance to a level at which unreliability performance would be adversely affected. The regulatory structure itself should be subjected to critical scrutiny for inappropriate incentives.

Step 5e: Is it worth modifying the regulatory structure in the manner proposed, considering the particulars of the regulatory issue?

Among the high-level performance-based guidelines, the assessment guidelines are best suited to make this evaluation. A feedback process involving a wide range of stakeholders may be the most effective way to develop the required information. Such a process may explicitly consider the cost impacts of incorporating requirements in one or other part of the regulatory structure.

APPENDIX J
POTENTIAL RISK-INFORMED AND PERFORMANCE-BASED REQUIREMENTS

J. POTENTIAL RISK-INFORMED AND PERFORMANCE-BASED REQUIREMENTS

J.1 Introduction

This appendix contains initial drafts of example technical and administrative requirements for each of the Framework topics listed in Table 8-3 of the Framework. These topics were derived using the process described in Chapter 8 and implemented in Appendix G.

The draft example requirements contained in this appendix are for the purpose of illustrating how the process described in the Framework can be used to develop requirements. However, the draft example requirements contained in this appendix do not represent an NRC staff approach. Much additional work (e.g., see Appendix C) and iteration would be needed to arrive at a consensus set of requirements.

Since, in developing the requirements, it is the intent to utilize existing 10 CFR 50 requirements and General Design Criteria as much as practical, along with each draft requirement is an indication of whether or not an existing 10 CFR 50 requirement or General Design Criterion from 10 CFR 50, Appendix A, is fully used or partially used. In addition, tables are provided for each set of requirements that summarize what kind of implementation guidance could be necessary for each requirement, both in a technology-neutral and technology-specific fashion.

This appendix is organized by topic area to follow the topic structure in Chapter 8 as follows:

(A) General Requirements Common to Design, Construction and Operation
(B) Physical Protection Requirements
(C) Good Design Practices Requirements
(D) Good Construction Practices Requirements
(E) Good Operating Practices Requirements
(F) Administrative Requirements

J.2 Draft Example Requirements

This section contains a series of tables for the topic areas listed in Section J.1.

For each topic area there are two tables, one containing the draft example requirements for that topic area and one containing example information on the content of technology-neutral and, where necessary, technology specific regulatory guides to support implementation of each draft example requirement. The specific tables contained in this section are:

A. General Requirements Common to Design, Construction and Operation
- Table J-1 Draft Example Requirements
- Table J-2 Example Regulatory Guide Content

B. Physical Protection Requirements
- Table J-3 Draft Example Requirements
- Table J-4 Example Regulatory Guide Content

C. Good Design Practices Requirements
- Table J-5 Draft Example Requirements
- Table J-6 Example Regulatory Guide Content

D. Good Construction Practices Requirements
- Table J-7 Draft Example Requirements
- Table J-8 Example Regulatory Guide Content

E. Good Operating Practices Requirements
- Table J-9 Draft Example Requirements
- Table J-10 Example Regulatory Guide Content

F. Administrative Requirements
- Table J-11 Draft Example Requirements
- Table J-12 Example Regulatory Guide Content

Where the draft example requirements use the words "will" or "shall", they are for the purpose of illustration only.

Table J-1 Draft example general requirements (common to design, construction and operation).

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
1) QA/QC Safety significant structures, systems, and components need to be designed, fabricated, erected, tested, maintained and operated to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they need to be identified and evaluated to determine their applicability, adequacy, and sufficiency and need to be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance and quality control program needs to 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, testing, maintenance and operation of safety significant structures, systems, and components need to be maintained by or be under the control of the nuclear power unit licensee throughout the life of the unit.	**Yes** GDC #1 and 10 CFR 50 Appendix B

Table J-1 Draft example general requirements (common to design, construction and operation).

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
2) PRA Scope and Technical Acceptability Each application to construct and operate a NPP needs to include a design specific probabilistic risk-assessment (PRA) that: (a) analyzes the risk from full power and low power operation, shutdown, refueling, and spent fuel storage (except dry cask storage); (b) includes assessment of internal and external events and quantifies uncertainties; (c) includes assessment of all event sequences down to 10^{-8}/yr (mean value); and (d) is conducted in accordance with accepted standards appropriate for the reactor technology.	**No**
3) Use of Risk-Information Risk information from the design-specific PRA needs to be used in design, construction and operation to identify and ensure that the most important systems, structures, components, human actions and accident sequences are considered in: (a) LBE selection and analysis; (b) safety classification and special treatment; (c) technical specification development and implementation; (d) procedure development and implementation; (e) training programs; (f) plant configuration control; (g) plant staffing; (h) inspection, surveillance and monitoring programs; (i) reliability assurance programs; (j) aging management programs; (k) maintenance programs; and (l) events assessment.	**No**
4) Integration of Safety, Security and Preparedness In designing, constructing, operating and maintaining the NPP, safety, security and preparedness needs to be considered in an integral fashion such that plant design, construction, maintenance and operational decisions are made in consideration of the impact each has on the other.	**No**

Table J-2 Example regulatory guide content related to general requirements (common to design, construction and operation).

Draft Requirement	Technology-Neutral Regulatory Guide Material	Technology-Specific Regulatory Guide Material
1) QA/QC	Develop, based upon existing guidance as much as possible (e.g., 10 CFR 50, Appendix B).	None
2) PRA Scope and Technical Acceptability	Develop guidance based on Framework Chapter 7 and Appendix F.	Develop technology-specific guidance for, e.g.: • risk metrics • risk methods • data
3) Use of Risk-Information	Develop guidance on the uses of risk-information, e.g.: • design and operations optimization • ITACC prioritization • inspection • assessment of events • also, develop guidance on risk measures to be used; e.g., - importance measures - dominant risk scenarios	Technical Specifications • importance measures • risk metrics
4) Integration of Safety, Security and Preparedness	Explain intent of requirement and integrated decision-making guidelines.	None

Table J-3 Draft example requirements related to physical protection.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
1) General Security The design needs to comply with the requirements of 10 CFR 73. The resolution of security related issues needs to be by design, wherever practical.	**Yes** 10 CFR 73 plus the desire to resolve security related issues by design
2) Security Performance Standards Each application to construct and operate an NPP needs to contain a security assessment. The purpose of this security assessment is to demonstrate compliance with the following security performance standards; such that there is high assurance of protection of public health and safety: (a) reduce vulnerabilities to DBTs and a limited set of beyond DBTs; (b) ensure that the plant design, operation and security provide multiple lines of defense against each security related threat that could endanger public health and safety; (c) ensure that the plant design, operation and security provide both prevention and mitigation measures for each security related threat that could endanger public health and safety; and (d) ensure sufficient material control and accounting to detect the theft or diversion of significant amounts of material. This security assessment needs to be maintained up to date over the life of the plant to reflect changes in the threat situation.	**No**

Table J-4 Example regulatory guide content related to physical protection requirements.

Draft Requirement	Technology-Neutral Regulatory Guide Material	Technology-Specific Regulatory Guide Material
1) General Security	Reference existing regulatory guides for 10 CFR 73.	None
2) Security Performance Standards	Describe acceptable scope, approach and acceptance criteria (e.g., conditional risk decision matrix) for a security assessment from Chapter 6.	None
	Discuss assumptions and methodology for calculating QHOs. Reference 10 CFR 73 guidance for theft and diversion.	None

Table J-5 Draft example requirements related to good design practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
1) Plant Risk The design specific probabilistic risk assessment (PRA) needs to demonstrate that: (a) each accident sequence in the PRA meets the appropriate dose limit on the F-C curve (shown in the Figure J-1) using mean risk values. (b) the overall risk from the NPP (or if more than one NPP from all NPPs on site licensed after _______) meets the QHOs expressed in the Commission’s 1986 Safety Goal Policy (or approved surrogate risk objectives) using mean risk values.	**No**

Table J-5 Draft example requirements related to good design practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
2) Criteria for Selection of the Licensing Basis Event sequences from the design specific PRA which must be considered in the licensing analysis needs to be categorized as follows: • frequent $\geq 10^{-2}$/ry (mean frequency) • infrequent $<10^{-2}$/ry but $\geq 10^{-5}$/ry (mean frequency) • rare $<10^{-5}$/ry but $\geq 10^{-7}$/ry (mean frequency) Within each of these categories, the applicant/licensee need to designate those sequences of each event type (e.g., LOCA, external events, etc.) with the largest consequences as Licensing Basis Events (LBEs) which need to meet the acceptance criteria in Design Requirement #3. A postulated LBE for plant siting purposes needs to be selected in accordance with and meet the acceptance criteria in Design Requirement #8.	**No**
3) LBE Acceptance Criteria Event sequences selected as LBEs need to meet the following acceptance criteria: (a) LBEs in the frequent category need to: (1) not exceed the annual dose criteria represented by the F-C curve in Design Requirement #1, at the 95% confidence level. (2) not result in any fuel damage (no additional release of fission products or fuel beyond the initiating event and no loss of non-damaged fuel lifetime). (3) not result in any additional barrier damage or failure beyond the initiating event. (4) not result in the loss of any reactor shut down or decay heat removal functions.	No

Table J-5 Draft example requirements related to good design practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
(b) LBEs in the infrequent category need to: (1) not exceed the dose criteria represented by the F-C curve in Design Requirement #1 in the infrequent frequency range on a per event basis, at the 95% confidence level. (2) not result in loss of coolable core geometry (no fuel melting or other condition, such as fuel temperature, that could result in the uncontrolled movement of fission products and/or fuel from their intended location). (3) not result in the loss of containment functional capability to control the release of fission products or other radioactive material to the environment. (4) not result in the loss of all reactor shutdown or decay heat removal functions. (c) LBEs in the rare category need not exceed the dose criteria represented by the F-C curve in Design Requirement #1 in the rare frequency range on a per event basis, at the 95% confidence level. (d) A postulated LBE needs to be used for siting purposes as described in Design Requirement #8.	
4) Initiating Event Severity Any initiating events that have the potential to simultaneously fail multiple protective systems resulting in a release of radioactive material to the environment, such that the F-C curve limits in Design Requirement #1 would be exceeded, need not have a frequency of occurrence greater than 10^{-7}/ry. (This applies to all events except security related events).	**No**
5) Safety Classification and Special Treatment All plant systems, structures and components (SSCs) relied upon to function during LBEs and security related events need to be considered safety significant, and need to receive special treatment during design, construction and operation. The plant-specific risk assessment may be used in determining the special treatment needed consistent with their safety function. The special treatment may include sufficient testing, QA, EQ, maintenance and/or other controls that will help ensure they perform their intended function when called upon, consistent with assumptions in the licensing analysis. This needs to include ensuring reliability and availability goals are met consistent with Design Requirement #30.	**Replaces 10 CFR 50.69**

Table J-5 Draft example requirements related to good design practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
6) Equipment Qualification All equipment in the licensing analysis classified as safety significant needs to be qualified for the service conditions under which they are assumed to operate. These service conditions need to include those of normal operation, accident conditions and conditions associated with security related events. Acceptable qualification techniques include testing under actual service conditions or demonstrated service history under similar conditions.	**Replaces 10 CFR 50.49 and expands the scope of EQ**
7) Licensing Analysis The analysis to be used to support licensing (e.g., risk, design and safety analysis) needs to be done with analysis tools that are qualified for their particular applications in accordance with Design Requirement #42. The analysis needs to be conducted on a realistic basis and uncertainties need to be quantified. The uncertainties that need to be quantified include parameter and modeling uncertainties. Acceptance or success criteria used in the analysis needs to be consistent with acceptable codes and standards or needs to be set conservatively to ensure a less than 5% chance of failure, if not exceeded. A mechanistic source term may be used in the analysis, if justified. If a deterministic source term is used, then its chemical form, timing of release and magnitude should be sufficient to bound expected values.	**No**
8) Siting and Site Specific Considerations The design needs to comply with 10 CFR 100 (Part B) siting requirements. In addition, the plant design needs to consider all natural phenomena and man-made hazards associated with the site in accordance with the criteria stated in Design Requirements #1, #2 and #11. In addition, security and EP needs to be considered in site selection. For siting purposes, the plant's capability to prevent radiological releases to the environment needs to be calculated using a postulated LBE. This postulated LBE needs to be representative of a mechanistic event scenario and source term that results in a large amount of fission products being released from the reactor fuel and coolant system. The radiological containment functional capability (i.e., design pressure, temperature, isolation time and allowable leakage rate) must be sufficient to ensure that, when analyzed mechanistically, the postulated LBE does not cause the dose to an individual located at the EAB to exceed 25 rem TEDE during the "worst 2 hours" of the event or to exceed 25 rem TEDE to an individual located at the outer edge of the LPZ for the duration of the accident.	**Refers to 10 CFR 100 and replaces 10 CFR 50.34(a)(1)(ii)D**

Table J-5 Draft example requirements related to good design practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
9) Use of Consensus Codes and Standards for Design The design of safety significant system, structures and components (SSCs) needs to be based, to the extent possible, upon nationally accepted consensus codes and standards that are applicable to the materials, temperature, pressures and other service conditions to which the SSCs are subjected over their lifetime. Each code or standard used in the design must be submitted to NRC for review.	**Replaces 10 CFR 50.55a**
10) Materials Qualification All materials used in safety significant power supplies, structural, pressure boundary or radioactive material retention components needs to be qualified for the service conditions expected over the life of the plant. This needs to include normal, as well as off-normal service conditions and account for aging and degradation mechanisms. Qualification needs to be demonstrated by actual previous experience with the materials under the conditions expected or by sufficient testing under prototype conditions to demonstrate acceptable performance of the material.	**No**
11) Protection Against Natural Phenomena Safety significant structures, systems, and components needs to 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 needs to reflect: (1) the natural phenomena identified by the PRA; (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.	**Yes** GDC #2 with modification to reflect risk-derived nature of the Framework
12) Dynamic Effects Safety significant structures, systems, and components needs to be designed to 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.	**Yes** GDC #4 with modification to reflect risk-derived nature of the Framework

Table J-5 Draft example requirements related to good design practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
13) Sharing of Structures, Systems, and Components Safety significant structures, systems, and components need 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.	**Yes** GDC #5
14) Reactor Shutdown and Decay Heat Removal Each reactor design needs to have at least two redundant, diverse and independent means of highly reliable reactor shutdown and decay heat removal. Each reactor shutdown means needs to be capable of bringing the reactor to cold shutdown independent of the other means (assuming withdrawal of the highest worth control rod in each means) and each decay heat removal means needs to be capable of removing decay heat to an ultimate heat sink independent of the other means. The reactor shutdown means needs to 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. Each means needs to also be capable of periodic testing and inspection.	Replaces GDC #10, 20, 21, 22, 23, 25, 26, 27, 28 and 34
15) Barriers to Release of Radioactive Material Each reactor design needs to have at least two physical barriers to prevent the release of fission products from the reactor core to the surrounding area. These barriers needs to be independent from one another, diverse, redundant and needs to be capable of periodic testing and inspection.	**No**
16) Radiological Containment Functional Capability Each reactor design needs to have a radiological containment functional capability, independent from the fuel and reactor coolant system, to control the release of radioactive material. The design objective of the radiological containment functional capability is to be able to establish a controlled low leakage barrier, that meets the dose criteria specified in Design Requirement #1, assuming the occurrence all frequent and infrequent event category LBEs, any rare event category LBE and security related event for which containment functional capability is credited in the licensing analysis and the postulated LBE to be used for siting purposes (see Design Requirement #8). The controlled leakage barrier needs to be capable of being inspected and tested to ensure it meets its design objective (e.g., leakage) over the life of the plant.	Replaces GDC #16, 50, 52 and 53

Table J-5 Draft example requirements related to good design practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
17) Radiological Containment Atmosphere Cleanup Systems to control fission products, and other radioactive substances which may be released into the reactor radiological containment boundary needs to be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of radioactive material released to the environment following postulated accidents. The radiological containment atmosphere cleanup systems needs to be capable of periodic testing and inspection.	**Yes** GDC #41, 42 and 43 (except H_2 requirements which are addressed in Requirement #33)
18) Fracture Prevention of Radiological Containment Pressure Boundary The reactor radiological containment boundary needs to be designed with sufficient margin to assure that under operating maintenance, testing, and frequent and infrequent events (1) its ferritic materials behave in a non-brittle manner, and (2) the probability of rapidly propagating fracture is minimized. The design needs to reflect consideration of service temperatures and other conditions of the radiological containment boundary material during operation, maintenance, testing, frequent events, infrequent events, and the LBE described in Design Requirement #8, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.	**Yes** GDC #51 with modification to reflect the risk-derived nature of the Framework
19) Electric Power Systems An onsite electric power system and an offsite electric power system needs to be provided to permit functioning of safety significant structures, systems, and components. The safety function for each electric power system needed to power safety significant structures, systems and components needs to provide sufficient capacity and capability to assure that the LBE acceptance criteria specified in Requirement #3 above are met. The onsite electric power supplies, including the batteries, and the onsite electric distribution system, needs to have sufficient independence, redundancy, and testability to perform their safety functions, assuming a loss of offsite power. Safety significant electric power systems needs to be designed to permit appropriate periodic inspection and testing.	**Yes** GDC #17 minus prescriptive requirements to be consistent with risk-derived nature of the Framework

Table J-5 Draft example requirements related to good design practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
20) Piping Systems Penetrating Radiological Containment Boundary Each line that is part of the reactor coolant pressure boundary and that penetrates the reactor radiological containment boundary needs to be provided with isolation valves as follows, unless it can be demonstrated that the 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 the radiological containment boundary; or (2) One automatic isolation valve inside and one locked closed isolation valve outside the radiological containment boundary; or (3) One locked closed isolation valve inside and one automatic isolation valve outside the radiological containment boundary. A simple check valve may not be used as the automatic isolation valve outside the radiological containment boundary; or (4) One automatic isolation valve inside and one automatic isolation valve outside the radiological containment boundary. A simple check valve may not be used as the automatic isolation valve outside the radiological containment boundary. Isolation valves outside the radiological containment boundary needs to be located as close to the boundary as practical and upon loss of actuating power, automatic isolation valves needs to be designed to take the approach 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 needs to 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 inservice inspection, protection against more severe natural phenomena, and additional isolation valves needs to include consideration of the population density, use characteristics, and physical characteristics of the site environs.	**Yes** GDC #54, 55 and 56 modified to use Framework terminology
21) Closed System Isolation Valves Each line that penetrates reactor radiological containment boundary and is neither part of the reactor coolant pressure boundary nor connected directly to the atmosphere contained within the boundary needs to have at least one isolation valve which needs to be either automatic, or locked closed, or capable of remote manual operation. This valve needs to be outside the boundary and located as close to the boundary as practical. A simple check valve may not be used as the automatic isolation valve.	**Yes** GDC #57 modified to use Framework terminology

Table J-5 Draft example requirements related to good design practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
22) Vulnerability to a Single Human Action or Hardware Failure Compliance with Design Requirement #1 need not be dependent on any single human action (operator, maintenance, surveillance, security or emergency preparedness) or piece of hardware.	**No**
23) Plant Aging and Degradation The design needs to consider aging and degradation phenomena that may occur over the life of the plant, such that sufficient structural integrity and equipment performance is maintained to ensure the assumptions in the licensing analysis remain valid. The plant design needs to include sufficient provisions for inspection and testing to monitor aging and degradation.	**No**
24) Reactor Inherent Protection The reactor needs to be designed to have a negative power coefficient under all normal and off-normal conditions and to exhibit stable operation under all expected conditions of power and core flow rate. Control rod worth needs to be limited such that the inadvertent removal of one control rod need not cause the reactor to go critical. Control rods need to also be designed so as not to be subject to inadvertent ejection from the core during normal operation (i.e., power operation, shutdown or refueling).	Replaces GDC #11 and 12
25) Human Factors/Human Machine Interface Each reactor design needs to incorporate accepted human factors and man-machine interface practices into the design, maintenance and operation. This needs to include attention to such factors as accessibility, location, marking, lighting, environment, procedures, training and other important considerations to reduce the likelihood of human error. These practices needs to apply to both the control room and ex-control room portions of the plant.	**No**
26) Fire Protection Safety significant structures, systems, and components needs to be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials need to be used wherever practical throughout the unit, particularly in locations such as the control room. For liquid metal reactor designs, provisions (such as steel liners) needs to be provided in areas containing liquid metal to protect concrete from liquid metal impingement. For graphite moderated reactor designs, provision needs to be provided to extinguish or isolate the fire, without the generation of large quantities of combustible gas. Fire detection and fighting systems of appropriate capacity and capability need to be provided and designed to minimize the adverse effects of fires on safety significant structures, systems, and components. For	**Yes** GDC #3 modified to reflect the risk-derived nature of the Framework and to address liquid metal coolant and graphite fires

Table J-5 Draft example requirements related to good design practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
liquid metal reactor designs, fire suppression decks, inerting and other compatible fire fighting provisions need to be included. Fire fighting systems need to be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components. Fire protection systems need to ensure the fire suppression materials used do not cause adverse reactions with other plant materials. The PRA needs to be used to identify (consistent with Design Requirement #2) the fire scenarios which the design must assess and comply with Design Requirement #3. (10 CFR 50.48 requirements for a fire protection plan and allowing the use of a performance-based, risk-informed approach should also be incorporated into this requirement).	
27) Control Room Design The main control room needs to be designed with sufficient shielding and atmospheric control to ensure habitability by control room personnel for all accident sequences that have a frequency greater than 10^{-7}/ry (mean value). Habitability needs to encompass assuring the dose to control room operating personnel does not exceed 5 rem for the duration of the accident and that hazardous chemicals are not allowed entry in sufficient concentrations to affect the health and safety of control room personnel. The control room needs to have sufficient instrumentation, control and communication capability to allow all safety significant functions to be performed from this location.	Replaces GDC #19
28) Alternate Shutdown Location Each reactor design needs to have at least one location physically separate from the main control room where the reactor can be safely shut down, including establishment of decay heat removal, instrumentation to confirm key plant parameters and communications capability.	Replaces GDC #19
29) Reactor Core Flow Blockage and Bypass Prevention Each reactor design needs to provide measures to prevent bypass and blockage of flow through the reactor core that is sufficient to cause localized fuel damage.	**No**

Table J-5 Draft example requirements related to good design practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
30) Reliability and Availability Reliability and availability goals need to be established for each system, structure and component classified as safety significant under Design Requirement #5. These reliability and availability goals need to be consistent with assumptions in the licensing analysis and, for SSCs that do not have a sufficient operating history under the conditions expected to confirm their reliability and availability, a reliability assurance program needs to be conducted to establish the SSC reliability and availability. After plant operation begins, a monitoring and feedback program to assess SSC reliability and availability needs to be conducted in accordance with Operating Practices Requirement #15. To complement the above requirement on reliability and availability goals, goals on initiating event frequency needs to also be established, consistent with the assumptions in the licensing analysis. These goals need to also be monitored consistent with Operating Practices Requirement #14.	**No**
31) Research and Development Each applicant for a design certification or combined operating license under 10 CFR 52, or a construction permit and operating license, needs to be responsible for all research and development necessary to validate the assumptions in the licensing analysis. This needs to include assumptions related to normal and off-normal conditions, burnup, fluence and plant aging for all safety significant SSCs.	**No**
32) Use of Prototype Testing The use of a prototype reactor as part of research and development, to demonstrate the performance of a reactor design through the conduct of a comprehensive test program, is one acceptable way to provide data to validate analytical tools and confirm SSC performance, provided the following are met: • sufficient testing can be done to be applicable to all conditions of the licensing basis; and • fuel burn-up, fluence and plant aging effects can be accounted for.	**No**
33) Combustible Gas Control Use words from 50.44(a) and (d).	**Yes** Uses words from 50.44(a) and (d)

Table J-5 Draft example requirements related to good design practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
34) Energetic Reaction Control Reactor designs that have the potential for energetic reactions between the fuel, coolant or other material needs to include provisions to prevent or mitigate the effects of such reactions such that reactor shutdown, decay heat removal and coolable core geometry can be maintained.	**No**
35) Prevention of Reactor Coolant Boundary Brittle Fracture The reactor coolant pressure boundary needs to be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design needs to 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. The reactor coolant pressure boundary needs to also be designed to permit periodic inspection and testing. A reactor vessel materials surveillance program needs to also be developed and maintained in accordance with Operational Requirement #9.	**Yes** GDC #31and 32 10 CFR 50.60, 50.61 and Appendix G included as guidance for LWRs in RG
36) Reactor Coolant Pressure Boundary The reactor coolant pressure boundary needs to be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, or rapidly propagating failure, and of gross rupture as required by Requirement #35 above. The reactor coolant pressure boundary needs to be monitored for leakage such that, to the extent possible, the location of the leak can be located. Reactor designs that rely on the concept of leak before break to limit the pressure boundary leak size for which the plant has to be designed needs to: • demonstrate the material being used as the pressure boundary does, in fact, develop a thru-wall crack prior to developing a circumferential crack; and • have the capability to detect leakage from thru-wall cracks prior to the cracks exceeding assumptions in the licensing basis.	**Yes** GDC #14,and 30 supplemented with words on leak before break
37) Reactor Coolant Activity Monitoring and Cleanup Each reactor design needs to provide provisions for monitoring reactor coolant activity so as to detect abnormal conditions and remove impurities to restore and maintain the reactor coolant activity within normal levels.	**No**

Table J-5 Draft example requirements related to good design practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
38) Instrumentation and Control Systems Instrumentation needs to be provided to monitor variables and systems over their anticipated ranges for normal operation, for frequent and infrequent events 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 need to be provided to maintain these variables and systems within prescribed operating ranges. Instrumentation and control systems need to be designed in accordance with IEEE standards and need to be qualified to operate in the environmental conditions (both normal operation and accident conditions) under which they must function. Software based I and C systems must use software validated for the service intended using acceptable verification and validation techniques. These techniques must be submitted to the NRC for review.	**Yes** GDC #13 with modification to reflect the risk-derived nature of the Framework
39) Protection of Operating Staff The design needs to include provision for protection of the operating staff from harsh environments during normal and off-normal operations. For protection from radiation, the design needs to comply with 10 CFR 20, including ALARA. For protection from other hazards, (temperature, chemicals, inert gas, etc.), the design needs to comply with accepted standards.	**Yes** References 10 CFR 20 and use as guidance 10 CFR 50 Appendix I, modified to be applicable to other technologies as well as LWR
40) Control of Releases of Radioactive Materials to the Environment The nuclear power unit design needs to 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 frequent events. Sufficient holdup capacity needs to 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.	**Yes** 10 CFR 50.34a and GDC #60 with modification to use Framework terminology
41) Monitoring Radioactivity Releases Means need to be provided for monitoring the atmosphere within the radiological containment boundary, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including frequent, infrequent and rare events.	**Yes** 10 CFR 50.34a and GDC #64 with modification to use Framework terminology

Table J-5 Draft example requirements related to good design practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
42) Qualified Analysis Tools The analysis tools used in the licensing analysis needs to be qualified for use by validation against data obtained from acceptable test programs and/or actual operating experience. The analytical tools need to be shown to be validated for use over the range of conditions expected and need to be capable of quantifying uncertainties. The analysis tools, test data, program description and their validation process and results need to be submitted to NRC for review.	**No**

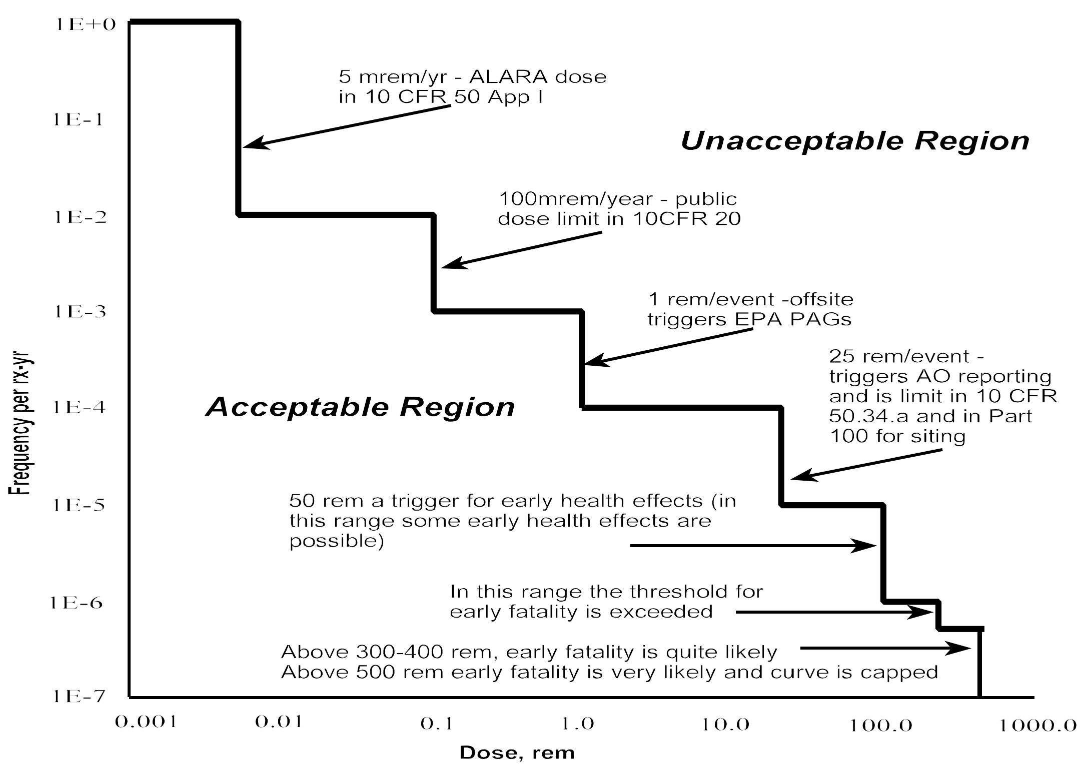

Figure J-1 Dose (rem) at the EAB and at the Outer Edge of the LPZ (EAB per year for event scenario frequencies greater than 10^{-3}/yr and for the worst 2 hours for event scenario frequencies less than 10^{-3}/yr and for the LPZ, the duration of the event scenario).

Table J-6 Example regulatory guide content related to good design practice requirements.

Draft Requirement	Technology-Neutral Regulatory Guide Material	Technology-Specific Regulatory Guide Material
1) Plant Risk	Discuss application of F-C curve and integrated risk. Develop QHO calculation guidelines.	Develop and discuss surrogate technology-specific risk objectives.
2) Criteria for Selection of the Licensing Basis	Develop guidance on how to select LBEs.	Develop types of LBE categories for different technologies (e.g., LOCA, on-line refueling).
3) LBE Acceptance Criteria	Describe dose criteria and confidence level. Describe reactor shutdown, decay heat removal and barrier integrity functionality.	Develop definitions for: • fuel damage • coolable core geometry Develop acceptable design limits for: • fuel (e.g., temperature, enthalpy, burnup, power density).
4) Initiating Event Severity	Discuss intent of requirement and develop guidance for implementation.	Develop and describe example events (e.g., PTS) that could affect multiple protective systems.
5) Safety Classification and Special Treatment	Describe SSC selection process, special treatment selection process and need to consider security.	Develop and discuss technology-specific importance measures and criteria for their use in establishing special treatment requirements.
6) EQ	Develop and describe what conditions equipment must be qualified for (e.g., LBE conditions) and what is expected to demonstrate qualification (e.g., testing). Also, need to consider security.	Discuss EQ considerations for: • HTGRs – graphite dust • LMRs – sodium aerosols, fuel qualification to run beyond clad breach

Table J-6 Example regulatory guide content related to good design practice requirements.

Draft Requirement	Technology-Neutral Regulatory Guide Material	Technology-Specific Regulatory Guide Material
7) Licensing Analysis	Develop and describe guidelines for safety analysis (LBE analysis, risk-assessment): • realistic calculations • use of a mechanistic or deterministic source term • uncertainties quantification • setting acceptance/ success criteria • dose calculations (meteorology, etc.) • code validation and documentation	Develop and describe guidance for technology-specific: • source term selection E.g., — HGTR – graphite dust in source term — LMR – activated sodium in source term • attenuation mechanisms
8) Siting and Site Specific Considerations	Describe: (1) How to select a deterministic LBE for siting, including NRC approval. (2) Mechanistic analysis of deterministic siting LBE. (3) Guidelines for dose calculations (e.g., source term, meteorology). (4) EP and security considerations Refer to RG 4.7 for other siting factors.	Develop and describe example deterministic LBEs for each technology that could be used for siting and for use in establishing radiological containment functional capabilities (see Design Requirement #16). Discuss technology-specific source term considerations: • HTGR – graphite dust • LMR – activated sodium aerosols
9) Use of Consensus Codes and Standards for Design	Describe the SSCs whose design needs to be based on consensus design codes and standards. Also identify acceptable TN codes and standards: • NQA • IEEE • ISO • etc.	Identify acceptable technology-specific codes and standards for: • graphite structures • RPV • etc. Consider use of 10 CFR 50.55a and its RG for guidance for LWRs.
10) Materials Qualification	Discuss intent of requirement and develop guidance for implementation.	Develop and describe material specific qualification guidance: • graphite • stainless steel • etc.

Table J-6 Example regulatory guide content related to good design practice requirements.

Draft Requirement	Technology-Neutral Regulatory Guide Material	Technology-Specific Regulatory Guide Material
11) Protection Against Natural Phenomena	Modify existing guidance for GDC #2 to reflect use of PRA and term “safety significant.” Also, use 10 CFR 50 Appendix S for earthquake engineering guidance.	None
12) Dynamic Effects	Modify existing guidance for GDC #4 to reflect use of PRA and term “safety significant.”	None
13) Sharing of Structures, Systems and Components	Modify existing guidance for GDC #5 to reflect use of PRA and term “safety significant.”	None
14) Reactor Shutdown and Decay Heat Removal	Discuss performance expectations. Define “means.” Discuss testing and inspection guidelines.	None
15) Barriers to Release of Radioactive Material	Develop and discuss performance expectations. Develop TN definition of “barriers.”	Develop technology-specific barrier definitions: • RCS • containment • coated particle • etc.
16) Radiological Containment Functional Capability	Discuss intent of requirement and develop guidance for implementation, including: • containment functional performance parameters that need to be established. • how to use the LBE, security related events and the deterministic LBE from Design Requirement #8 to set containment design requirements.	Describe example events that could be used to set containment design requirements.
17) Radiological Containment Atmosphere Cleanup	Discuss intent of requirement and use applicable existing guidance for GDC #41, 42 and 43.	Discuss atmosphere cleanup needs and methods for each reactor technology, e.g., • LWR - aerosols • HTGR - graphite dust • LMR - N_a and N_a aerosols

Table J-6 Example regulatory guide content related to good design practice requirements.

Draft Requirement	Technology-Neutral Regulatory Guide Material	Technology-Specific Regulatory Guide Material
18) Fracture Prevention of Radiological Containment Pressure Boundary	Discuss intent of requirement and use applicable existing guidance for GDC #51.	Develop and discuss fracture prevention guidance for different reactor technologies, e.g., • temperature • pressure • materials
19) Electric Power Systems	Use existing guidance for GDC #17, but reflect application for electric power only when needed for “safety significant” SSCs.	None
20) Piping Systems Penetrating Radiological Containment Boundary	Use existing guidance for GDC #54, 55, and 56.	None
21) Closed System Isolation Valves	Use existing guidance for GDC #57.	None
22) Vulnerability to a Single Human Action or Hardware Failure	Discuss intent of requirement and develop guidance for implementation.	None
23) Plant Aging and Degradation	Discuss intent of requirement and develop guidance for implementation.	None
24) Reactor Inherent Protection	Discuss intent of requirement and guidance for implementation.	Discuss what can cause positive power coefficients in: • LWRs (voids, boron dilution) • HTGRs (seismic) • LMRs (voids in core due to N_a boiling, flow blockage or cover gas entrainment). Discuss how to prevent these from occurring.
25) Human Factors/ Human Machine Interface	Describe acceptable guidelines: • NUREG-0700 • etc.	None
26) Fire Protection	Develop and describe general fire protection guidance. (Use current guidance for GDC #3 and 50.48).	Develop guidelines for: • Sodium fire protection • Graphite fire protection

Table J-6 Example regulatory guide content related to good design practice requirements.

Draft Requirement	Technology-Neutral Regulatory Guide Material	Technology-Specific Regulatory Guide Material
27) Control Room Design	Develop guidance on: • Dose criteria • Air quality	None
28) Alternate Shutdown Location	Develop and discuss guidance on location, environment and capability.	None
29) Reactor Core Flow Blockage and Bypass Prevention	Discuss intent of requirement and develop guidance for implementation.	Discuss what is meant by "fuel damage". Ensure consistency with design requirement #3 "fuel damage" definition. Discuss what can cause flow blockage and bypass: • LWR - debris • HTGR - graphite core block cracks • LMR - N_a plugging, debris Discuss how to prevent these from occurring.
30) Reliability and Availability	Describe how to establish reliability and availability goals. Describe content of the reliability assurance program. Describe IE frequency goals.	None
31) Research and Development	Describe responsibility of applicant and example R&D areas.	None
32) Use of Prototype Testing	Develop and discuss details of what prototype test programs must address. See Appendix G, Section G.2.2.1 for list of items.	None
33) Combustible Gas Control	Discuss intent of requirement. Use existing guidance for 50.44 where applicable.	Describe sources of combustible gas which are to be considered: • N_a/water reaction • N_a/concrete reaction • Graphite/water reaction • Zr/water reaction

Table J-6 Example regulatory guide content related to good design practice requirements.

Draft Requirement	Technology-Neutral Regulatory Guide Material	Technology-Specific Regulatory Guide Material
34) Energetic Reaction Control	Discuss intent of requirement and develop guidance for implementation.	Describe sources of energetic reactions: e.g.: • FCI • N_a/water reaction
35) Prevention of Reactor Coolant Boundary Brittle Fracture	Discuss intent of requirement and develop guidance for implementation.	Develop material specific limits. Use existing guidance for 50.60 and 50.61 and 10 CFR 50, Appendix G for LWR.
36) Reactor Coolant Pressure Boundary	Develop and describe what needs to be done to demonstrate leak before break: • material testing • leak detection • etc. Use existing guidance for GDC #14 and 30.	Describe acceptable materials, conditions for leak before break, and leak detection for each technology.
37) Reactor Coolant Activity Monitoring and Cleanup	Describe goals of coolant activity monitoring and cleanup and guidance for implementation. Describe technical basis to be used to set activity limits.	Describe type of monitoring required: • gamma (activation of impurities) • delayed neutron (fuel in coolant) Describe technology specific activity considerations (e.g., graphite dust collection and liftoff) Describe types of cleanup systems for each technology.
38) I and C Systems	Describe what needs to be done to qualify hardware and software. Use existing guidance for GDC #13.	None
39) Protection of Operating Staff	Use or reference guidance for 10 CFR 20 on radiation and environmental protection.	Develop and describe ALARA goals for each technology building upon 10 CFR 50, Appendix I.

Table J-6 Example regulatory guide content related to good design practice requirements.

Draft Requirement	Technology-Neutral Regulatory Guide Material	Technology-Specific Regulatory Guide Material
40) Control of Releases of Radioactive Materials to the Environment	Use LWR guidance for 10 CFR 50.34a and GDC #60.	None
41) Monitoring Radioactivity Releases	Use LWR guidance for 50. 34a and GDC #64 modified to remove LWR specific terminology.	None
42) Qualified Analysis Tools	Develop and discuss guidelines for how to qualify analytical tools. Can use RG 1.203 as a general reference.	None

Table J-7 Draft example requirements related to good construction practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
1) Use of Accepted Construction Codes, Standards and Practices On-site and off-site fabrication/construction need to be conducted in accordance with accepted codes, standards and practices applicable to the materials and construction techniques being used. Personnel performing the fabrication/construction techniques need to be qualified to conduct the fabrication/construction techniques used, or need to be under the supervision of qualified personnel.	**No**
2) Security During Construction/Fabrication During on-site and off-site fabrication/construction of safety significant SSCs, access to the SSCs need to be controlled, such that only authorized personnel have access. The applicant/licensee needs to be responsible for developing and implementing the access control and a personnel authorization program.	**No**

Table J-7 Draft example requirements related to good construction practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
3) Non-Destructive Examination (NDE) During Construction/ Fabrication NDE techniques need to be used to examine the quality of material and fabrication/construction (e.g., welds) on all safety significant SSCs. The NDE may be applied to a sample of the SSCs and fabrication/construction provided such sampling can be justified, or may cover all SSCs and fabrication/construction. The NDE techniques and personnel used need to be qualified for their intended applications. The techniques, qualification program and sampling plan need to be submitted to NRC for review.	**No**
4) Inspection During Construction and Fabrication The applicant/licensee needs to establish, implement and maintain an inspection program significant to ensure that the fabrication/construction of safety significant SSCs is accomplished consistent with the design and quality intended. The inspection personnel needs to be qualified to perform their assigned tasks and the inspection program needs to define what is to be done, how it is to be done and the acceptance criteria. The inspection program needs to be consistent with the licensing analysis and focus on those items most important to safety.	**No**
5) Testing of SSCs During Construction and Fabrication Upon completion of fabrication/construction of safety significant SSCs, testing needs to be performed to ensure that the SSCs perform their safety function. The testing may be coordinated and conducted in conjunction with other phases of plant testing. A description of the test program, including what is to be tested, how it is to be tested, when it is to be tested and the acceptance criteria needs to be submitted to the NRC for review.	**No**

Table J-8 Example regulatory guide content related to construction requirements (good practice)

Draft Requirement	Technology-Neutral Regulatory Guide Material	Technology-Specific Regulatory Guide Material
1) Use of Accepted Codes, Standards and Practices	Describe intent of requirement and guidance for implementation.	Describe and participate in the development of accepted codes and standards.
2) Security During Construction/Fabrication	Describe intent of requirement and guidance for implementation.	None

Table J-8 Example regulatory guide content related to construction requirements (good practice)

Draft Requirement	Technology-Neutral Regulatory Guide Material	Technology-Specific Regulatory Guide Material
3) NDE During Construction/Fabrication	Describe how to use PRA to develop sampling plan.	Describe acceptable NDE techniques.
4) Inspection During Construction/Fabrication	Describe how to use PRA to develop inspection plans (e.g., use of importance measures). Reference NUREG on construction inspection. Address inspection of factory fabrication and fabrication outside the U.S.	Develop and discuss technology-specific inspection needs (e.g., HTGR fuel quality).
5) Testing of SSCs During Construction/Fabrication	Describe how to use PRA to develop test program.	None

Table J-9 Draft example requirements related to good operating practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
1) Radiation Protection Operating procedures, controls and practices need to be developed, implemented and maintained to ensure that radiation exposure to operating personnel and the public from routine operation do not exceed the limits specified in 10 CFR 20 and comply with ALARA principles. For designs that use an intermediate heat transfer loop between the reactor coolant system (RCS) and the power generation system, the pressure in the intermediate loop needs to be maintained higher than the pressure in the RCS during power operation.	**Yes** 10 CFR 50, Appendix I, put in RG and modify to be applicable to other technologies, as well as LWRs.
2) Maintenance Program Use 50.65 words, plus the following: A maintenance program needs to be developed, implemented and maintained to ensure that the reliability, availability and performance of safety significant SSCs remain consistent with assumptions in the licensing analysis. The SSC reliability, availability and performance needs to be monitored and fed back into the licensing analysis in accordance with Operating Requirement #14.	**Yes** Use 50.65 modified to reflect use of risk information.

Table J-9 Draft example requirements related to good operating practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
3) Personnel Qualification Use 10 CFR 50.120 words.	**Yes** Use 10 CFR 50.120
4) Training Use 10 CFR 50.120 words.	**Yes** Use 10 CFR 50.120
5) Use of Procedures Plant security, operations, maintenance and emergency response needs to be controlled by the use of procedures. The procedures need to be developed integral with design, verified prior to use and maintained up to date over the life of the plant.	**No**
6) Use of Simulators Each plant needs to have a full scale control room simulator for the purpose of training operating personnel and verifying procedures. The simulator needs to be capable of simulating all LBEs in real time.	**No**
7) Staffing Each application needs to propose an operating staff level sufficient to perform all safety significant actions. Staffing levels for control room and ex-control room personnel needs to be subject to the NRC approval and, as a minimum, must include at least: (a) one RO per shift for each reactor, and (b) one senior RO per shift for each control room.	Replaces 10 CFR 50.54(k), (l) and (m) which could be used as guidance for LWRs in a RG.
8) Aging Management Program Each applicant to construct and operate a NPP under this Part needs to develop, implement and maintain an aging management program to detect and control aging of safety significant SSCs so as to maintain the plant within the assumptions used in the licensing analysis. A description of the aging management program needs to be submitted to the NRC for review.	**No**

Table J-9 Draft example requirements related to good operating practices.

<table>
<tr><th>Draft Example Requirements</th><th>Use Current GDC or 10 CFR 50 Regulation</th></tr>
<tr><td>9) Surveillance Program

Each applicant to construct and operate a NPP under this Part needs to develop, implement and maintain a surveillance and inspection program to:

(a) monitor the status of safety significant SSCs for degradation, proper alignment and other conditions which could adversely affect operability, and

(b) subject samples of key materials (e.g., RPV material, RPV internals material, cables, etc.) to the operating environment and periodically measure changes in their properties for comparison with assumptions in the licensing analysis.

A description of the program needs to be submitted to the NRC for review.</td><td>No

10 CFR Appendix H could be used as guidance in RG for material surveillance program.</td></tr>
<tr><td>10) In-Service Inspection

Each applicant to construct and operate a NPP under this Part needs to develop, implement and maintain an in-service inspection (ISI) program to inspect safety significant SSCs to ensure their integrity and consistency with assumptions in the licensing analysis. ISI techniques used need to be qualified for the materials, configurations and service conditions expected. A description of the ISI program needs to be submitted to the NRC for review.</td><td>No</td></tr>
<tr><td>11) In-Service Testing

Safety significant equipment needs to be tested periodically to demonstrate their performance. The testing intervals and test program need to be selected to be consistent with the assumptions in the licensing analysis.</td><td>No</td></tr>
<tr><td>12) Technical Specifications

Use 10 CFR 50.36 requirement, supplemented to address the role of risk information in identifying Tech Spec content.</td><td>Yes

10 CFR 50.36, supplement for the use of risk information.</td></tr>
</table>

Table J-9 Draft example requirements related to good operating practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
13) Emergency Preparedness Use 10 CFR 50.47 and Appendix E modified as follows: The emergency plans and procedures need to be developed integral with design and need to address both accident management and consequence mitigation, as well as off-site emergency actions, and the size of the EPZ, consistent with the projected source terms, timing, duration, dose and distance from the plant resulting from the release of radioactive material as identified by the licensing analysis and security considerations.	**Yes** 10 CFR 50.47 and Appendix E, supplemented with words on accident management, consequence mitigation and the use of the licensing analysis and security considerations in determining EP measures and EPZ size.
14) Monitoring and Feedback Each applicant to construct and operate an NPP needs to develop, implement and maintain a monitoring program to: (a) determine the reliability and availability of all safety significant equipment. This information needs to be periodically fed back into the licensing analysis so as to maintain the licensing analysis up to date. This information needs to also be compared to the reliability and availability goals established during design and, where these goals are not met, corrective action needs to be taken. (b) measure the release of radioactive material to the environment from normal operation and frequent events. This information needs to be compared to established limits and corrective action taken when limits are exceeded. (c) Measure the atmosphere within the radiological containment boundary for radioactivity and take corrective action if limits are exceeded.	**No**
15) Work and Configuration Control Each licensee to operate an NPP needs to develop and implement procedures to control plant work and configurations so as to minimize inadvertent plant challenges, situations hazardous to plant personnel and unanalyzed plant configurations. Criteria needs to be developed and implemented, based on the plant specific risk assessment, to guide what are acceptable plant configurations and the duration the plant can remain in each configuration.	**No**

Table J-9 Draft example requirements related to good operating practices.

Draft Example Requirements	Use Current GDC or 10 CFR 50 Regulation
16) Maintenance of the PRA Each licensee to operate an NPP needs to maintain its licensing analysis up to date. The plant specific PRA needs to be updated to reflect actual operating experience at least once every ________ years, or sooner if major unanalyzed situations are discovered. The information from the updated PRA needs to be used to update the plant's licensing basis including: • LBE selection and analysis • safety classification of SSCs • procedures • NDE, ISI, and IST programs • plant aging program • emergency preparedness Major changes resulting from these updates will require NRC approval in accordance with Administrative Requirement #2.	**No**
17) Fuel and Replacement Part Quality Each licensee to operate an NPP needs to develop and implement a program to ensure that over the life of the plant, new fuel and replacement parts used in the plant are of equal or better quality than the original. This program may use inspections, testing and other appropriate means to verify the quality. The licensee's program needs to be submitted to the NRC for review.	**No**
18) Security Each licensee to operate an NPP needs to comply with 10 CFR 73 requirements and any supplemental security requirements resulting from the security assessment specified in Security Requirement #2 conducted at the design stage and any updates of this assessment over the life of the plant.	**Yes** 10 CFR 73

Table J-10 Example regulatory guide content related to good operating practices.

Draft Requirement	Technology-Neutral Regulatory Guide Material	Technology-Specific Regulatory Guide Material
1) Radiation Protection	Description of intent of requirement and use existing guidance for 10 CFR 20.	Develop and describe ALARA for each technology. Use 10 CFR 50, Appendix I for LWR.
2) Maintenance Program	Use existing 10 CFR 50.65 guidance on how to set goals, what to do if goals are not met, supplemented with words on maintaining PRA assumptions.	None
3) Personnel Qualifications	Reference existing Regulatory Guides for 10 CFR 50.120 where possible.	None
4) Training	Reference existing Regulatory Guides for 10 CFR 50.120 where possible.	None
5) Use of Procedures	Describe intent of requirement and develop guidance for implementation.	None
6) Use of Simulators	Reference existing Regulatory Guides where possible. Develop and discuss guidance modular plant simulator.	None
7) Staffing	Develop and discuss guidance on how to select control room and ex-control room staffing levels, including for modular plants.	Develop and discuss technology-specific needs, such as a containment watch for LMRs. Consider using staffing requirements in 10 CFR 50.54 as guidance for LWRs.
8) Aging Management Program	Discuss intent of requirement and develop guidance on scope and approach for the program.	None

Table J-10 Example regulatory guide content related to good operating practices.

Draft Requirement	Technology-Neutral Regulatory Guide Material	Technology-Specific Regulatory Guide Material
9) Surveillance Program	Discuss intent of requirement and develop guidance for implementation. Discuss use of risk (e.g., importance measures) to optimize surveillance and inspection. Use non-LWR portions of 10 CFR 50, Appendix H.	Develop guidance on technology-specific material surveillance needs: e.g.: • HTGR - graphite • LWR - RPV (Use 10 CFR 50, Appendix H.) • LMR - RPV internals Develop guidance on surveillance of items key to the safety analysis (e.g., amount of graphite dust collecting in RCS using dose measured.
10) ISI	Discuss intent of requirement and develop guidance for implementation.	Develop guidance on technology-specific methods, needs, standards
11) In-Service Testing	Discuss intent of requirement and develop guidance for implementation.	Develop guidance on technology-specific testing needs: (e.g.: • HTGR - fuel integrity • LWR - ECCS • LMR - leak detection system
12) Technical Specifications	Use existing guidance for 10 CFR 50.36 supplemented to address the use of risk information. (See Framework Appendix G - Section G.2.2.3.)	Develop outline of content expected in technical specifications for: • LWRs • HTGRs • LMRs
13) Emergency Preparedness	Reference existing Regulatory Guides for 10 CFR 50.47 and Appendix E as applicable. Provide criteria that need to be considered in evaluating any adjustments to current requirements based upon the licensing analysis, including security related events. Develop guidance for accident management programs.	None

Table J-10 Example regulatory guide content related to good operating practices.

Draft Requirement	Technology-Neutral Regulatory Guide Material	Technology-Specific Regulatory Guide Material
14) Monitoring and Feedback	Discuss intent, scope and approach for monitoring and feedback. Develop guidance for implementation (e.g., use of risk importance measures).	None
15) Work and Configuration Control	Discuss intent, scope and approach for procedures and criteria. Develop guidance for implementation.	None
16) Maintenance of the PRA	Use guidance in Chapter 7.	None
17) Fuel and Replacement Part Quality	Discuss licensee responsibilities, including QA and inspection.	Discuss fuel fabrication and replacement part fabrication processes to be monitored.
18) Security	Reference existing Regulatory Guides for 10 CFR 73 where possible.	None

Table J-11 Draft example administrative requirements.

Draft Example Requirements	Use Current GDC or Regulation
1) Standard Format and Content of Applications Use 10 CFR 50.34 (except 50.34(f), (g) and (h) with the following supplement. Each application to construct and operate an NPP needs to contain the following information: (a) A summary of the PRA that shows _____________. (b) A proposed set of LBEs based upon design criterion #2. (c) Analysis showing the proposed LBEs meet design criterion #3. (d) A list of SSCs that are classified as safety significant in accordance with design criterion #5. (e) Analysis showing acceptable siting in accordance with design criterion #8. (f) Reliability assurance program. (g) Program for NDE and testing during construction. (h) Aging management program. (i) Surveillance and ISI program. (j) Program for assuring replacement part quality. (k) A description of how all design, construction, operation and administrative requirements are to be met. (l) Proposed set of technical specifications in accordance with operational criterion #12.	Replaces 10 CFR 50.33
2) Change Control Process Changes to the plant design, construction or operation subsequent to receiving an OL or COL requires NRC's review and approval if any of the following conditions are met: (a) the change results in any requirements of this Part not being met; (b) the change results in a revision to the approved LBEs, and/or list of safety significant SSCs; (c) the change results in a modification to the certified portion of a design that has been certified under 10 CFR 52; or (d) the change results in an increase in risk greater than _______.	Replaces 10 CFR 50.59
3) Record Keeping Utilize 10 CFR 50.71 supplemented with guidance on record keeping for risk-information.	**Yes** Use 10 CFR 50.71

Table J-11 Draft example administrative requirements.

Draft Example Requirements	Use Current GDC or Regulation
4) Documentation Control A program needs to be developed and implemented that ensures plant documentation (i.e., design, construction, operation and administrative) is controlled and kept up to date. **5) Reporting** Use licensee event report requirement from 10 CFR 50.73.	**No** **Yes** Use 10 CFR 50.73
6) Corrective Action Program A corrective action program needs to be developed and implemented that ensures when problems are discovered, they are corrected in a timely fashion.	**No**
7) Backfitting Use words from 10 CFR 50.109.	**Yes** Use 10 CFR 50.109 with guidance on risk metrics for non-LWRs
8) License Amendments Use 10 CFR 50.92, except for 50.92(c) which is replaced with the following: Changes to the plant design, construction or operation that affect conditions of the license requires NRC approval. License amendment requests need to address the impact of the proposed amendment on: • risk to public health and safety; • plant security and physical protection; and • defense-in-depth.	Replaces 10 CFR 50.92(c)
9) Exemptions Add to 10 CFR 50.12: • the impact of the exemption on plant risk, security and defense-in-depth.	**Yes** Use 10 CFR 50.12 with additional words on the use of risk information.
10) Legal and Process Items Use 10 CFR 50 sections identified in Appendix H.	**Yes**

Table J-12 Example regulatory guide content related to good operating practices.

Draft Requirement	Technology-Neutral Regulatory Guide Material	Technology-Specific Regulatory Guide Material
1) Standard Format and Content of Application	Develop guidance on application content.	None
2) Change Control Process	Develop guidance using risk criteria for requiring NRC approval and acceptance criteria (e.g., acceptable risk increases).	Develop risk metrics for each technology
3) Record Keeping	Use existing 10 CFR 50.71 guidance supplemented with additional description of what risk records need to be kept.	None
4) Document Control	Develop description of document control process.	None
5) Reporting	Utilize existing 10 CFR 50.73 guidance.	None
6) Corrective Action Program	Develop and describe guidance for an effective corrective action program.	None
7) Backfitting	Utilize existing guidance for back-fitting (50.109).	Need technology-specific guidance on risk metrics
8) License Amendments	Use existing guidance for 10 CFR 50.92 except for 50.92(c) which will require development of risk-derived acceptance criteria.	Develop risk informed acceptance criteria. Can use RG 1.174 for LWRs.
9) Exemptions	Use existing guidance for 10 CFR 50.12, supplemented with guidance on the use of risk information.	Develop guidance on risk metrics and acceptance criteria.
10) Legal and Process Items	Utilize existing guidance.	None

APPENDIX K
COMPLETENESS CHECK

K. COMPLETENESS CHECK

K.1 Introduction

As described in Chapter 8, a top down process has been used to identify the topics for which requirements are needed to have a stand alone risk-informed and performance-based approach for future plant licensing. The process started with the high level protective strategies (introduced in Chapter 2) and, through the use of structured logic diagrams for each protective strategy, identified the pathways that could lead to failure of that protective strategy. The topics that the requirements will need to address to prevent failure of the various pathways were then identified using experience and knowledge about reactor safety. Defense-in-depth was then considered for each protective strategy (to account for uncertainties) by applying the defense-in-depth principles described in Chapter 4 to each protective strategy. The end result of applying this process is summarized in Chapter 8 of the Framework (Table 8-3), which lists the technical topics which the requirements must address.

A similar process was followed for the administrative requirements; however, the defense-in-depth principles were not applied in the administrative area. The end result of applying the process to the administrative area resulted in the list of administrative topics discussed in Chapter 8 of the Framework.

To help ensure that the list of technical and administrative topics shown in Table 8-3 is complete, a check was made against other documents containing requirements for reactor safety. Specifically, the following documents were compared against Table 8-3 of the Framework:

- 10 CFR 50: “Domestic Licensing of Production and Utilization Facilities”
- IAEA Safety Standards Series NS-R-1: “Safety of Nuclear Power Plants: Design” [IAEA 2000a]
- IAEA Safety Standards Series NS-R-2: “Safety of Nuclear Power Plants: Operation” [IAEA 2000b]
- NEI 02-02: “A Risk-Informed, Performance-Based Regulatory Framework for Power Reactors” [NEI 2002]
- U.K. Health and Safety Executive Document, “Safety Assessment Principles for Nuclear Facilities” [UK 20006]

This appendix documents the results of the completeness check. The results of the comparisons are discussed in Sections K.2 through K.6.

In addition to reviewing the above documents, a December 2006 EPRI report “Technical Elements of Risk-Informed, Technology-Neutral Licensing Framework for New Nuclear Plants,” [EPRI 2006] was reviewed for its applicability to the identification of risk-informed requirements. However, the scope of the EPRI report was limited to a review and critique of a draft of the NRC Framework and the PBMR risk-informed licensing approach. As such, it did not identify any potential requirements and thus was not used in the completeness check discussed in this appendix.

K.2 Comparison Against 10 CFR 50

Table K-1 shows the results of the comparison against 10 CFR 50. Table K-1 addresses all requirements in 10 CFR 50 (technical and administrative). No technical requirements found in 10 CFR 50 were found missing in the Framework, except those unique to light water reactors (LWRs), which the Framework is not intended to include.

For the administrative topics, Table 8-3 identified those items necessary to control documentation, ensure sufficient record keeping and reporting, ensure sufficient information is included in applications and amendment requests and other items that document the plant condition. However, there are a number of administrative requirements (e.g., legal, process, etc.) that were not specifically identified by the application of the process described in Chapter 8, but rather were identified by comparison against 10 CFR 50. These 10 CFR 50 administrative requirements would be applicable to any risk-informed and performance-based licensing approach and are identified as such in Table K-1. These include:

- financial requirements
- process requirements
- employee protection requirements
- legal requirements

Table K-1 10 CFR 50 comparison.

10 CFR Part 50		Framework
General Provisions		
50.1	Basis, Purpose, and Procedures Applicable	• Applicable.
50.2	Definitions	• Review for applicability. Some will be applicable, some will need modification, some will not be applicable and new ones will need to be added to define selected Framework terms.
50.3	Interpretation (Assigns legal interpretation authority to NRC General Counsel)	• Applicable.
50.4	Written Communication	• Applicable.
50.5	Deliberate Misconduct	• Applicable.
50.7	Employment Protection (Protects employees of licensees against discrimination and retribution for providing information to NRC, Congress, etc.)	• Applicable.
50.8	Information Collection Requirements: OMB Approval	• Applicable.
50.9	Completeness and Accuracy of Information	• Applicable.

Table K-1 10 CFR 50 comparison.

10 CFR Part 50		Framework
Requirement of License, Exceptions		
50.10	License Required (Establishes license requirement Identifies facilities which are required to obtain an NRC license and which are not)	• Applicable.
50.11	Exceptions and Exemptions from License Requirements	• Applicable.
50.12	Specific Exemptions	• Included in Framework, supplemented with words on the use of risk information; see Framework draft Administrative Requirement #9.
50.13	Attacks by Enemies of the US	• Applicable.
Clarification and Description of Licenses		
50.20	Two Classes of Licenses	• Use 50.20 Class 103 license words.
50.21	Class 104 License (Medical facility and device manufacturer licenses)	• Not applicable to Framework.
50.22	Class 103 License for Commercial and Industrial Facilities	• Applicable.
50.23	Construction Permits	• Applicable.
Applications for Licenses, Forms, Contents, Ineligibility of Certain Applications		
50.30	Filing of Application for License: Oath of Affirmation	• Applicable.
50.31	Combining Applications	• Applicable.
50.32	Elimination of Repetition	• Applicable.
50.33	Contents of Application (General Information)	• Replaced with Framework draft Administrative Requirement #1.
50.33a	Information Requested by the Attorney General for Antitrust Review	• Applicable.
50.34	Contents of Application (Technical Requirements)	• Included in Framework, except for 10 CFR 50.34(f) and (h) which are LWR specific. 10 CFR 50.34(g) is addressed in Framework draft Design Requirement #33. Also see Framework draft Administrative Requirement #1.
50.34a	Design Objective Requirements for Equipment to Control the Release of Radioactive Active Material	• Applicable. See Framework draft Design Requirements #40 and #41.
50.35	Issuance of Construction Permits	• Applicable.

K. Completeness Check

Table K-1 10 CFR 50 comparison.

10 CFR Part 50		Framework
50.36	Technical Specifications	• Included in Framework as draft Operational Requirement #12, supplemented to require the use of risk-information.
50.36a	Technical Specifications on Effluent from Nuclear Power Plants	• Applicable.
50.36b	Environmental Conditions	• Applicable.
50.37	Agreement Limiting Access to Classified Information	• Applicable.
50.38	Foreign Corporation or Individual Restriction	• Applicable.
50.39	Public Inspection of License Requirement	• Applicable.
50.40	Common Standards (Part 51 Compliance, Requirement for licensee to be technically and financially qualified, Operation does not infringe on defense or public health)	• Applicable.
50.41	Additional Standards for Class 104 License	• Not applicable to Framework.
50.42	Additional Standards for Class 103 License (Usefulness Requirement Antitrust Restriction Open Communication Requirement)	• Applicable.
50.43	Additional Standards and Provision Affecting Class 103 Licenses for Commercial Power Plants (NRC is required to inform of applications for licenses: 1. State and Local Authorities 2. Public via Federal Register 3. Other Cognizant Federal Agencies)	• Applicable.
50.44	Combustible Gas Control for Nuclear Power Reactors (BWR Containment Specifications Equipment Survivability Specifications Monitoring Requirements Analysis Requirements Requirement for Future Applicability)	• Included in Framework. LWR specific provisions removed. See Framework draft Design Requirement #33.
50.45	Standards for Construction Permits	• Applicable.
50.46	Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Reactors	• Not applicable - LWR specific.
50.46a	Acceptance Criteria for Reactor Coolant System Venting System	• Not applicable - LWR specific.

Table K-1 10 CFR 50 comparison.

10 CFR Part 50		Framework
50.47	Emergency Plans	• Included in Framework, with supplemental words on the use of information from the licensing analysis. See Framework draft Operational Requirement #13.
50.48	Fire Protection (General Description Specific Hazard Detection and Suppression Systems Administrative Controls Risk-informed Analysis Requirement)	• Partially included in Framework. • Framework requirement is based primarily on GDC #3. • See Framework draft Design Requirement #26.
50.49	Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants	• Replaced by Framework draft Design Requirement #6.
50.50	Issuance of Licenses and Construction Permits	• Applicable.
50.51	Continuation of License (Sets time limits on term of license Holds licensee responsible for site after permanent shutdown)	• Applicable.
50.52	Combining Licenses	• Applicable.
50.53	Jurisdictional Limits	• Applicable.
50.54	Conditions of Licenses (Organized Description Nuclear Material Control Restrictions Emergency and War Control Revocation, Suspension, Modification and Amendment Provisions Information Request Rules Antitrust Limitations Personnel Control Requirements (staffing) Personnel Requalification Plans Licensed Operator Watch Requirements Safeguards Contingency Plan Requirements Emergency Plan Requirements Physical Security Safeguards and Contingency Plan Requirements Insurance Requirements Clean up Plan Requirements Restart and Decommissioning Authority Safety Deviation Allowance Fuel Storage Following Decommissioning Bankruptcy Notification Requirements National Security Technical Spec Allowance	• Applicable, except fuel reprocessing and LWR operator staffing requirement. For staffing requirements, use Framework draft Operating Requirement #7. • Fuel reprocessing and research reactor requirements are not applicable.

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Table K-1 10 CFR 50 comparison.

10 CFR Part 50		Framework
50.55	Conditions of Construction Permits (Failure and defect information and correction plan Time Limits for correction of defects and reporting requirements for failure to correct Defines conditions for required reports Report content requirements Directives of where to deliver reports Quality Assurance requirements SAR change reporting requirements)	• Applicable.
50.55a	Codes and Standards (Identifies acceptable Codes and Standards Sets Minimum Requirements for Specific Structural Materials)	• Replaced by Framework draft Design Requirement #9. • 50.55a could be used as guidance for LWRs.
50.56	License Conversion	• Applicable.
50.57	Issuance of Operating License (Requirements to issue an operating license)	• Applicable.
50.58	Hearings and report of the ACRS	• Applicable.
50.59	Changes, Tests, and Experiments	• Not included in Framework. • See Framework draft Administrative Requirement #2.
50.60	Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation	• Not applicable. • LWR specific. • Could be used in RG as guidance for LWRs.
50.61	Fracture toughness requirements for protection against pressurized thermal shock events	• Not Applicable - LWR specific. • Could be used in RG as guidance for LWRs. • Use Framework draft Design Requirement #35.
50.62	Requirements for reduction of risk from ATWS events for light water cooled nuclear power plants	• Not applicable - LWR specific.
50.63	Loss of all alternating current power	• Not applicable - LWR specific.
50.64	Limitation on the use of Highly Enriched Uranium (HEU) in Domestic Non-power Reactors	• Not applicable. • Applies to non-power reactor.
50.65	Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	• Included in Framework as draft Operational Requirement #2, supplemented by words on ensuring assumptions in the PRA are maintained.

Table K-1 10 CFR 50 comparison.

10 CFR Part 50		Framework
50.66	Requirements for Thermal Annealing of the Reactor Pressure Vessel	• Not applicable - LWR specific.
50.67	Accident Source Term (Defines applicability and requirements for existing plants that want a license amendment to use a revised source term Sets radiation exposure limits within defined areas around the plant)	• Not applicable.
50.68	Criticality Accident Requirements (Limits Concentrations of Storage Fuel Rods Limits Credit Taken for Moderation Limits Fuel Rod U-235 Purity)	• Not included in Framework. Scope of Framework does not currently cover fuel storage.
50.69	Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants	• Replaced by Framework draft Design Requirement #5.
Inspections, Records, Reports, Notifications		
50.70	Inspections (Requires licensees to submit to routine inspection Requires licensee to provide reasonable space accommodation to inspectors)	• Applicable.
50.71	Maintenance of Records, Making Reports (Defines items which must be records Sets requirements for quality of records Sets reporting periods for specific records)	• Included in Framework, supplemented with words of record keeping for risk information. See Framework draft Administrative Requirement #3.
50.72	Immediate Notification Requirements for Operating Nuclear Power Reactors (Defines events and conditions which must be reported to the NRC Sets time limits for reporting Sets follow up requirements)	• Applicable.
50.73	Licensee Event Report System (Defines events and conditions which must be reported via LER Sets time times for reporting Sets Follow-up requirements Sets Content requirements for LER)	• Included in Framework. See Framework draft Administrative Requirement #5.
50.74	Notification of Change in Operator or Senior Operator Status Reporting Requirement	• Applicable.
50.75	Reporting and Record Keeping for Decommissioning Planning (Establishes reasonable assurance that funds will be available for decommissioning process)	• Applicable.
50.76	Licensee Change of Status, Financial Qualifications (Requires licensee to inform NRC 75 days before ceasing to exist)	• Applicable.

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Table K-1 10 CFR 50 comparison.

10 CFR Part 50		Framework
US/IAEA Safeguards Agreement		
50.78	Installation information and verification (Requires licensees to submit to IAEA inspection when directed by NRC)	• Applicable.
Transfers of Licenses, Creditors Rights, Surrender of Licenses		
50.80	Transfer of Licenses (Requires NRC to consent to license transfer to qualified licenses Defines requirements for new licensee to receive license)	• Applicable.
50.81	Creditor Regulations (Sets conditions under which a creditor may posses a lien on a utilization and production facility)	• Applicable.
50.82	Termination of License (Sets time limits for notifying NRC of intention to terminate a license Sets time limit for decommissioning once intention is announced Sets Funding Requirements for Decommissioning Sets Radiation Survey Requirements)	• Applicable.
50.83	Release of Part of a Power Reactor Facility or Site for Unrestricted Use (Defines planning and Notification Requirements Sets Radiation Exposure Limits Sets Inspection Requirements)	• Applicable.
Amendment of License or Construction Permit at Request of Holder		
50.90	Application for Amendment of License or Construction Permit	• Applicable.
50.91	Notice of Public Comment and State Consultation (Time requirements for announcing and holding public comment meetings Sets requirements for NRC to consult and inform state officials of license changes)	• Applicable.
50.92	Issuance of Amendments (Identifies issues which are to be considered when evaluating a request for a license change)	• Included in Framework, supplemented with words on the use of risk information. See Framework draft Administrative Requirement #8.
Revocation, Suspension, Modification, Amendment of Licenses and Construction Permits, Emergency Operations by the Commission		
50.100	Revocation, Suspension, and Modification of Licenses and Construction Permits	• Applicable.
50.101	Retaking Possession of Special Nuclear Material	• Applicable.

Table K-1 10 CFR 50 comparison.

10 CFR Part 50		Framework
50.102	Commission Orders for Operation After Revocation	• Applicable.
50.103	Suspension and Operation in War or National Emergency	• Applicable.
Backfitting		
50.109	Backfitting	• Included in Framework as draft Administrative Requirement #7.
Enforcement		
50.110	Violations (Grants power to NRC to seek injunction for violations of Atomic Energy Act, NRC regulations, or violations of License)	• Applicable.
50.111	Criminal Penalties	• Applicable.
50.120	Training and Qualification of Nuclear Power Plant Personnel (Requirement to have a training program Training program standards Personnel required to receive training Training review and update requirements)	• Included in Framework. See Framework draft Operational Requirements #3 and #4.
Appendices		
A	General Design Criteria for Nuclear Power Plants	• See Table K-2.
B	Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants	• Included in Framework. See draft Common Requirement #1.
C	A Guide for the Financial Data and Related Information Required to Establish Financial Qualifications for Facility Construction Permits	• Applicable.
E	Emergency Planning and Preparedness for Production and Utilization Facilities	• Included in Framework. See draft Operation Requirement #13.
F	Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities	• Not applicable to Framework.
G	Fracture Toughness Requirements	• Not included in Framework. • Not applicable to all technologies. • Could be used for LWR in the form of a RG for draft Design Requirement #35.
H	Reactor Vessel Material Surveillance Program Requirements	• Use as RG for Framework draft Operational Requirement #9.

K. Completeness Check

Table K-1 10 CFR 50 comparison.

10 CFR Part 50		Framework
I	Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion “As Low as is Reasonably Achievable” for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents	• ALARA included in Framework. • Need technology-specific RGs to address ALARA guidance for each technology. Use Appendix I for LWR guidance. • See draft Framework Design Requirement #39 and Operational Requirement #1.
J	Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors	• Not applicable - LWR specific.
K	ECCS Evaluation Models	• Not applicable - LWR specific.
L	Information Requested by the Attorney General for Antitrust Review of Facility Construction Permits and Initial Operating Licenses	• Applicable.
M	Standardization of Design; Manufacture of Nuclear Power Reactors; Construction and Operation of Nuclear Power Reactors Manufactured Pursuant To Commission License	• Not needed. Superceded by 10 CFR 52.
N	Standardization of Nuclear Power Plant Designs; Licenses to Construct and Operate Nuclear Power Reactors of Duplicate Design at Multiple Sites	• Not needed. Superceded by 10 CFR 52.
O	Standardization of Design; Staff Review of Standard Designs	• Not needed. Superceded by 10 CFR 52.
Q	Pre-Application Early Review of Site Suitability Issues	• Applicable.
R	Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979	• Not applicable - LWR specific.
S	Earthquake Engineering Criteria for Nuclear Power Plants	• Applicable. Use as a guidance document. Supplement with words on use of Framework draft Design Requirement #2 to select SSE. Also see Framework draft Design Requirement #11.

Table K-2 10 CFR 50, Appendix A - General Design Criteria (GDC)

General Design Criteria	Framework/Draft Requirement #
1. Quality Standards and Records	included/Common #1
2. Design Bases for Protection Against Natural Phenomena	included/Design #11
3. Fire Protection	included/Design #26
4. Environmental and Dynamic Effects Design Bases	included/Design #12
5. Sharing of Structures, Systems and Components	included/Design #13
10. Reactor Design	intent included/Design #14
11. Reactor Inherent Protection	intent included/Design #24
12. Suppression of Reactor Power Oscillations	intent included/Design #24
13. Instrumentation and Control	included/Design #38
14. Reactor Coolant Pressure Boundary	included/Design #36
15. Reactor Coolant System Design	intent included/Design #3
16. Containment Design	intent included/Design #16
17. Electric Power Systems	partially included/Design #19
18. Inspection and Testing of Electric Power Systems	partially included/Design #19
19. Control Room	intent included/Design #27 and #28
20. Protection System Functions	intent included/Design #14
21. Protection System Reliability and Testability	intent included/Design #14 and #30
22. Protection System Independence	intent included/Design #14
23. Protection System Failure Modes	intent included/Design #14
24. Separation of Protection and Control Systems	not included (design specific)
25. Protection System Requirements for Reactivity Control Malfunctions	intent included/Design #14
26. Reactivity Control System Redundancy and Capability	intent included/Design #3 and #14
27. Combined Reactivity Control System Capability	intent included/Design #14
28. Reactivity Limits	intent included/Design #14

Table K-2 10 CFR 50, Appendix A - General Design Criteria (GDC)

General Design Criteria	Framework/Draft Requirement #
29. Protection Against AOOs	intent included/Design #3
30. Quality of Reactor Coolant Pressure Boundary	intent included/Design #36
31. Fracture Prevention of Reactor Coolant Pressure Boundary	included/Design #35
32. Inspection of Reactor Coolant Pressure Boundary	included/Design #35
33. Reactor Coolant Makeup	not included - LWR specific
34. Residual Heat Removal	intent included/Design #14
35. Emergency Core Cooling	not included - LWR specific
36. Inspection of Emergency Core Cooling System	not included - LWR specific
37. Testing of Emergency Core Cooling System	not included - LWR specific
38. Containment Heat Removal	not included (design specific)
39. Inspection of Containment Heat Removal System	not included (design specific)
40. Testing of Containment Heat Removal System	not included (design specific)
41. Containment Atmosphere Cleanup	included/Design #17
42. Inspection of Containment Atmosphere Cleanup System	intent included/Design #17
43. Testing of Containment Atmosphere Cleanup System	intent included/Design #17
44. Cooling Water	not included - LWR specific
45. Inspection of Cooling Water System	not included - LWR specific
46. Testing of Cooling Water System	not included - LWR specific
50. Containment Design Basis	intent included/Design #16
51. Fracture Prevention of Containment Pressure Boundary	included/Design #18
52. Capability for Containment Leakrate Testing	intent included/Design #16
53. Provisions for Containment Testing and Inspection	intent included/Design #16

Table K-2 10 CFR 50, Appendix A - General Design Criteria (GDC)

General Design Criteria	Framework/Draft Requirement #
54. Piping Systems Penetrating Containment	included/Design #20
55. Reactor Coolant Pressure Boundary Penetrating Containment	included/Design #20
56. Primary Containment Isolation	intent included/Design #20
57. Closed System Isolation Valves	included/Design #21
60. Control of Releases of Radioactive Materials to the Environment	included/Design #40
61. Fuel Storage and Handling and Radioactivity Control	not currently in Framework scope
62. Prevention of Criticality in Fuel Storage and Handling	not currently in Framework scope
63. Monitoring Fuel and Waste Storage	not currently in Framework scope
64. Monitoring Radioactivity Releases	included/Design #41

K.3 Comparison Against IAEA NS-R-1

Table K-3 shows the results of the comparison against International Atomic Energy Agency (IAEA) document NS-R-1. The IAEA document differs from 10 CFR 50 in that it is written to be more general (i.e., many of the requirements are stated in the form of objectives or principles). Like 10 CFR 50, the IAEA document is written to be applicable to LWRs and covers technical as well as administrative topics.

Table K-3 does not track directly with the organization of the IAEA standard. The table has been organized to group similar IAEA requirements by broad categories for easier comparison with the Framework.

In reviewing Table K-3 it can be seen that most of the topics included in NS-R-1 have also been identified in Chapter 8 of the Framework. However, NS-R-1 does include some design topics not found in Chapter 8. These are:

- automatic safety actions in initial stage of accidents
- escape routes
- design fuel assemblies to permit inspection

Accordingly, these need to be assessed as to whether or not they should be incorporated into the Framework as part of implementation of the Framework.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
Objectives, Purposes, and Bases	
General Nuclear Safety Objective: To protect individuals, society, and the environment from harm by establishing and maintaining in nuclear installations effective against radiological hazards. Radiation Protection Objective: To ensure that all operational states radiation exposure within the installation or due to planned release of radioactive material from the installation is kept below prescribed limits and as low as reasonably achievable, and to ensure the mitigation radiological consequences of any accidents.	• Included in principle.
Defense-in-Depth	
Level 1: defense to prevent deviations from normal operation, and to prevent system failures. Level 2: defense to detect and intercept deviations from normal operational states in order to prevent anticipated operational occurrences from escalating to accident conditions. Level 3: Anticipate unlikely escalations in the design basis for the plant and to achieve stable and acceptable plant states following such events. Level 4: defense to address severe accidents in which the design basis may be exceeded and to ensure that radioactive releases are kept as low as practical. Level 5: mitigation of the radiological consequences of potential releases of radioactive materials that may result from accident conditions.	• DID discussed in Framework. DID principles applied to identify needed requirements and DID provisions are included in the requirements.
Safety functions	
The objective of the safety approach needs to be to provide adequate means to maintain the plant in a normal operational state. At all levels of operation and accidents design needs to: • Control radioactivity • Remove heat from the core • Confine radioactive materials and control operational discharges A systematic approach needs to be followed to identify structures, systems, and components that are necessary to fulfill the safety function.	• Included in principle through protective strategies.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
Management Requirements/Confidence	
Responsibility in Management	
Have a clear division of responsibility with corresponding lines of authority and communication. Ensure that it has sufficient technically qualified and appropriately trained staff at all levels. Establish clear interfaces between the groups engaged in different parts of the design, and between designers, utilities, suppliers, constructors and contractors as appropriate.	• Organization and management not included. These are management issues, not design issues and should not be included in design requirements.
Develop and strictly adhere to sound procedures.	• Procedures are included.
Review, monitor and audit all safety related design matters on a regular basis. Ensure that a safety culture is maintained.	• Safety culture is not included. This is a management issue and should not be included in design requirements.
Management of Design	
Ensure that characteristics, specifications, and materials can provide adequate protection for the life of the design. Ensure that the requirements of the operating organization are met and that due account is taken of the human capability and limitations. Design should take into account deterministic and complementary probabilistic safety analyses. Design needs to ensure that the generation of radioactive waste is kept to the minimum practicable.	• Included in principle.
Tracking and Records Requirements	
Safety Classification	
All structures, systems and components including software that are important to safety needs to be identified and classified according to their safety function. The method for classifying safety significant equipment needs to be based primarily on deterministic analysis with complementary probabilistic analysis. System interfaces need to be designed such that systems with lower safety significance need never propagate failure to systems of greater safety significance.	• Included in principle.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
Safety Objectives	
Independent Verification of the Safety Assessment	
Accident Prevention and Plant Safety Characteristics	
Plants need to be designed such that sensitivity to accidents is minimized. Postulated Initiating Events (PIE) produce no significant safety related effect or produce only a change in the plant towards a safe condition by inherent characteristics. Following a PIE, the plant is rendered safe by passive safety features or by the action of safety systems that are continuously operating in the state necessary to control the PIE. Following a PIE, the plant is rendered safe by the action of safety systems that need to be brought into service in response to a PIE. Following a PIE, the plant is rendered safe by specified procedural actions.	• Included in principle.
General Design Basis	
The design basis needs to specify the necessary capabilities of the plant to cope with a specified range of operational states and design basis accidents. Conservative design measures need to be applied and sound engineering practices need to be adhered to in the design basis for normal, abnormal, and accident operation. Performance of the plant in situations beyond design basis need to be addressed in the design.	• Included in principle.
General Requirements for Instrumentation and Control Systems Important to Safety	
Instrumentation needs to be provided to monitor plant variables and systems over the respective ranges for normal operation, anticipated operational occurrences, design basis accidents, and severe accidents. Instrumentation and recording equipment need to be provided to ensure that essential information is available for monitoring the course of design basis accidents and the status for essential equipment. Appropriate and reliable controls need to be provided to maintain the plant parameters within specified operational ranges.	• Included in principle.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
Confidence in Personnel	
Proven Engineering Practices	
Wherever possible, structures, systems and components important to safety need to be designed according to the latest or currently applicable approved standards. Where an unproven design or feature is introduced or there is a departure from an established engineering practice, safety needs to be demonstrated to be adequate by appropriate research and testing. In the selection of equipment, consideration needs to be given to both spurious operation and unsafe failure modes.	• Included in principle.
Operational Experience and Safety Research	
Design needs to take into account relevant operational experience.	• Included in principle.
Safety Assessment	
A comprehensive safety assessment needs to be carried out to confirm that the design as delivered meets the safety requirements. Safety Assessment needs to be part of the design process. The basis for safety assessment needs to have data derived from safety analysis, operational experience, research and proven engineering practice.	• Included in principle.
Human Factors	
The design needs to be operator friendly and needs to be designed to minimize the potential for operational error. The working areas and working environment of the site personnel need to be designed according to ergonomic principles. Systematic consideration of human factors and human machine interface need to be included throughout the design process. The human-machine interface needs to be designed in order to provide operators comprehensive but easily manageable information. Verification and validation of aspects of human factors need to be included at appropriate stages to confirm that the design adequately accommodates all necessary operator actions. Operators need to be considered to have dual roles, that of equipment operators and systems managers. Operators need to be provided with information which permits an understanding of the overall condition of the plant, and the determination of the appropriate operator initiated safety actions to be taken. As equipment operator, operators need to be provided with sufficient information on parameters associated with individual plant systems and equipment to confirm that the necessary safety actions can be initiated safely.	• Included in principle.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
The design should be aimed at promoting the success of operator actions with due regard for time, physical environment, and physiological demands.	
Control Room	
A control room needs to be provided from which the plant can be safely operated in all its operational states, and from which measures can be taken to maintain the plant in a safe state or to bring it back into such a state after the onset of anticipated operational occurrences, design basis accidents and severe accidents. Special attention needs to be given to identifying those events, both internal and external to the control room, which may pose a direct threat to continued operation. The layout of the control room needs to be such that personnel can have an overall picture of the status and performance of the plant. Devices need to be provided to give visual and if appropriate audible indication of the operating state and processes that have deviated from normal and could affect safety.	• Included in principle.
Emergency Control Center	
An on-site emergency control center separated from the plant control room needs to be provided for use by emergency staff.	• Included in principle.
Confidence in Engineering	
Quality Assurance	
A quality assurance program that describes the overall arrangements for the management, performance and assessment of the plant design needs to be prepared and implemented. Design, including subsequent changes or safety improvements needs to be carried out in accordance with established procedures that call on appropriate engineering. Adequacy of design needs to be verified or validated by individuals or groups separate from those originating the design.	• Included in principle.
Operational States	
Plants need to be designed to operate within a specific set of physical parameters with a minimum set of supporting safety features in operational condition. The potential for accidents at low power and shutdown states need to be addressed in the design. The design process needs to establish a set of requirements and limitations for safe operation. These requirements and limitations need to be a basis for the establishing of operational limits and conditions.	• Included in principle.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
Common Cause Failures	
The potential for common cause failures of items important to safety needs to be considered to determine where the principle of diversity, redundancy, and independence should be applied to achieve the necessary reliability.	• Included in principle.
Fail-Safe Design	
Fail-safe design needs to be considered and incorporated into the design of systems and components.	• Included in principle.
Auxiliary Services	
Auxiliary services supporting safety systems need to be considered part of the safety systems and need to be classified accordingly.	• Included in principle.
Provision for In-Service Testing, Maintenance, Repair, Inspection and Monitoring	
SSCs need to be inspected, tested, and repaired in a manner commensurate with their safety importance such that sufficient reliability of the safety function can be maintained. Where it is not possible to performance testing and inspection, alternate or indirect surveillance need to be utilized and conservative safety margins need to be applied.	• Included in principle.
Equipment Qualification	
A qualification procedure needs to be adopted to confirm that the items important to safety are capable of meeting demands for performing their function throughout their design operational lives. Any unusual environmental conditions that can reasonably be anticipated needs to be included in the qualification program.	• Included in principle.
Aging	
Appropriate margin needs to be provided to incorporate aging into SSCs designs throughout the design life.	• Included in principle.
Interactions of Systems	
When there is a significant probability that it will be necessary for safety systems to operate simultaneously, possible interaction whether direct or indirectly needs to be evaluated.	• Included in principle.
Interactions between the electrical power grid and the plant	
Account needs to be taken of the power plant to grid interaction including independence of and number of power supply lines to the plant relative to necessary reliability of outside power to safety systems.	• Included in principle.
Safety Analysis	
A safety analysis of the plant design needs to be conducted in which methods of both deterministic and probabilistic analysis needs to be applied.	• Included in principle.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
Deterministic Approach	
Deterministic safety analysis needs to include the following: Confirmation that operational limits and conditions are in compliance with the assumptions and intent of the design for normal operation of the plant; Characterization of the PIEs that are appropriate for the design and site of the plant; Analysis and evaluation of event sequences that result from PIEs; Comparison of the results of the analysis with radiological acceptance criteria and design limits; Establishment and confirmation of the design basis; Demonstration that the management of anticipated operational occurrence and design basis accidents is possible by automatic response of safety systems in combination with prescribed actions of the operators; and Applicability of the analytical assumptions, methods and degree of conservatism needs to be verified.	• Included in principle.
Probabilistic Approach	
A probabilistic safety analysis of the plant needs to be carried out in order to: Provide a systematic analysis to give confidence that the design will comply with the general safety objectives; Ensure that no particular PIE has a disproportionately large contribution to overall risk; Provide confidence that small deviations in plant parameters that could give rise to severely abnormal plant behavior will be prevented; Provide assessment of the probabilities of occurrence of severe core damage states; Provide assessment of the probabilities of occurrence and the consequence of external hazards; Identify systems for which design improvements could reduce the probability of severe accidents; Assess adequacy of plant emergency procedures; and Verify compliance with probabilistic targets.	• More extensive use of PRA is included in the Framework.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
In-service Inspection of the Reactor Coolant Pressure Boundary	
The reactor coolant system pressure boundary needs to be designed, manufactured and arranged in a manner that adequate inspections and tests can be made at appropriate intervals. It needs to be ensured that it is possible to inspect or test either directly or indirectly the components of the reactor coolant pressure boundary. Indicators for the integrity of the reactor coolant pressure boundary need to be monitored. If safety analysis of the nuclear power plant indicates that particular features in the secondary cooling system may result in serious consequences, it needs to be ensured that it is possible to inspect relevant pars of the secondary cooling systems.	• Included in principle.
Use of Computer Based Systems in Systems Important to Safety	
Computer systems required by safety systems need to be subject to standards and practices for the development and testing of the hardware and software. The level of reliability needs to be commensurate with the safety importance of the system. The level of reliability assumed in the safety analysis for a computer based system needs to include a specified conservatism to compensate for the inherent complexity of the technology.	• Included in principle.
Automatic Control	
Various safety actions need to be automated so that operator action is not necessary within a justified period of time from the onset of anticipated operational occurrences or design basis accidents.	• Not included.
Functions of the Protection System	
The protection system needs to be designed: To initiate automatically the operation of appropriate systems, including, as necessary, the reactor shutdown system, in order to ensure that design limits are not exceeded; To detect design basis accidents and initiate the operation of necessary systems; and To be capable of overriding unsafe actions of the control system.	• Included in principle.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
Reliability and Testing of the Protection System	
The protection system needs to be designed for high functional reliability and periodic testability commensurate with the safety function of the system. Design needs to ensure that: No single failure results in a loss of protective function; and The removal from service of any component or channel does not result in loss of the necessary minimum redundancy. Protection systems need to be designed to ensure that the effects of all operating conditions do not result in loss of function or that the loss is acceptable. Protection systems need to be designed to permit periodic testing of its function when the reactor is in operation. Protection systems need to be designed to minimize the likelihood that operator actions could defeat the effectiveness of the protection system.	• Included in principle.
Use of Computer Based Systems in Protection	
Where a computer based system is intended to be used in protection systems: The highest quality of and best practices for hardware and software need to be used; The whole development process needs to be systematically documented and reviewable; An assessment of the computer based system needs to be undertaken by independent expert personnel; and When the integrity of the system cannot be demonstrated with high confidence, a diverse means of fulfilling the protection function needs to be provided.	• Included in principle.
Contingency Planning	
Requirements for Defense-in-Depth	
Multiple physical barriers to uncontrolled release of RAM. Needs to be conservative, and construction needs to be of high quality. Needs to provide for control of the plant behavior during and following an PIE using inherent and engineered features.	• Framework DID has different objectives, scope and approach. Framework includes DID principles and requirements reflect DID provisions.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
Needs to provide for supplementing control of the plant, by the use of automatic activation of safety systems and operator actions. Needs to provide for equipment and procedures to control the course and limit the consequences of accidents. Needs to provide multiple means for ensuring that each of the fundamental safety functions is performed. Design needs to prevent as far as practicable: Challenges to the integrity of physical barriers; Failure of a barrier when challenged; and Failure of a barrier as a consequence of failure of another barrier. The first and second level of defense needs to prevent all but the most improbable events. Design needs to take into account the fact that the existence of multiple levels of defense is not a sufficient basis for continued power operation in the absence of one level of defense.	
Categories of Plant States	
Plant states need to be identified and grouped into a limited number of categories according to their probability of occurrence.	• Included.
Postulated Initiating Events	
Plant design needs to acknowledge that plant challenges can occur at all levels of defense-in-depth and design measures need to be provided to ensure that the necessary safety functions are maintained.	• Included.
Internal Events	
All those internal events which could affect plant safety need to be identified including: Fires and explosion, and Other internal hazards.	• Included.
External Events	
A combination of deterministic and probabilistic methods needs to be used to select a subset of external events which the plant is designed to withstand. Human caused and nature caused external events need to be considered in the design.	• Included in principle.
Site Related Characteristics	
Where combinations of randomly occurring events could credibly lead to abnormal or accident conditions, they need to be taken into account in the design.	• Included.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
Design Rules	
The engineering design rules for structures, systems, and components need to be specified and need to comply with the appropriate accepted national, or international or foreign engineering standards. Designs need to maintain sufficient margin to safety against seismic events.	• Included in principle.
Design Basis Accidents	
A set of design basis accidents need to be derived from potential accidents for the purpose of setting the boundary conditions for SSCs. Where prompt and reliable action is required, automatic systems need to be incorporated into the design. Provision for adequate instrumentation need to be provided where operator diagnosis and action is required to put the plant in a stable long term condition. Any equipment necessary in manual response and recovery processes need to be placed in the most suitable location to ensure its ready availability.	• Included in principle.
Severe Accidents	
Certain very low probability events arising due to failure of multiple safety systems which lead to significant core degradation and jeopardize the integrity of many or all barriers are referred to as severe accidents. Assessment and mitigation of these events need to be performed using best estimate techniques. Combinations of safety and non-safety systems may be considered in the mitigation of severe accidents.	• Included in principle.
Single Failure Criterion	
The single failure criterion needs to be applied to each safety group incorporated in the plant design. Spurious action needs to be considered a mode of failure. Single failure is considered to have been satisfied when any harmful consequence of an event are assumed to have occurred and the worst possible configuration of safety systems performing the necessary safety function is assumed. Single failure need not to be required for high quality passive components.	• Not included, except in a few key areas (i.e., reactor shutdown, decoy heat removal, barriers). Framework uses PRA.
Systems containing fissile and radioactive materials needs to be designed to be adequate in operational and design basis accidents	• Included in principle.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
Escape Routes and Means of Communication	
Nuclear power plants need to be designed with a sufficient number of safe escape routes, clearly and durable marked, with reliable emergency lighting, ventilation and other building service essential to safe escape. Suitable alarm systems and means of communications need to be provided so that all personnel on site can be warned and instructed. Availability of communications necessary for safety within the immediate vicinity of the site and to off site agencies need to be ensured at all times.	• Not included.
Decommissioning	
Consideration needs to be given to incorporating features that will facilitate the decommissioning and dismantling of the plant.	• Included in principle. (See 10 CFR 20.1406)
In particular: Choice of materials such that radioactive waste needs to be minimized; Access capabilities that may be necessary; and Facilities necessary for storing radioactive waste generated in both operation and decommissioning of the plant.	
Internal Structures of the Containment	
The design needs to provide for ample flow routes between separate compartments inside the containment. Consideration needs to be given to the internal structures during severe accidents.	• Not included - LWR specific.
Control and Cleanup of the Containment Atmosphere	
Systems to control fission products and other substances that may be released into the containment atmosphere. Systems for cleaning up the containment atmosphere needs to have suitable redundancy in components and features. Consideration needs to be given to the clean up of containment atmosphere during severe accidents.	• Included in principle.
Engineering Prescriptives	
Sharing of Safety Related Reactor Systems needs to be Avoided	
When systems are shared, systems need to be demonstrated that safety requirements are met of all reactors under all conditions.	• Included in principle.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
Power Plants used for Cogeneration	
Power plants used for cogeneration, heat generation or desalination need to be designed to prevent radioactive material from the nuclear plant to the desalination or district heating unit under all conditions.	• Not applicable.
General Design	
Reactor core and associated coolant, control and protection systems need to be designed to ensure that appropriate margins and radiation safety standards are applied in all operational states. Reactor core and associated internal components located within the reactor vessel need to be designed and mounted in such a way that they will withstand the static and dynamic loading expected in operational states. The maximum degree of positive reactivity and its maximum rate of increase by insertion in operational states and design basis accidents need to be limited so that no resultant failure of the reactor pressure boundary will occur, no cooling capability will be maintained and no significant damage will occur to the reactor core. The possibility of recriticality or reactivity excursion following PIE need to be minimized. The core and coolant and control and protection systems need to be designed to enable adequate inspection and testing.	• Included in principle.
Fuel Elements and Assemblies	
Fuel elements and assemblies need to be designed to withstand satisfactorily the anticipated irradiation and environment conditions in the reactor core. The deterioration considered needs to include that arising from differential expansion and deformation, irradiation, internal and external pressure, static and dynamic loading including vibration, and chemical effects. Specified fuel design limits need not be exceeded in normal operation and significant occurrences need not cause further deterioration.	• Included in principle.
Fuel assemblies need to be designed to permit adequate inspection of their structure and component parts after irradiation.	• Not included.
Requirements need to be maintained in the event fuel management strategy is changed.	• Included in principle.
Control of Reactor Core	
Reactivity, criticality and fuel assembly integrity need to be maintained for all levels and distributions of neutron flux in all modes of operation. Provision needs to be made for the removal of non-radioactive substances including corrosion products which may compromise safety systems.	• Included in principle.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
Reactor Shutdown	
Means need to be provided to ensure that there is a capability to shut down the reactor in operational states and design basis accidents and that shutdown conditions can be maintained in the most reactive core conditions.	• Included.
There needs to be at least two different systems available to shutdown reactor.	• Included.
At least one of the systems needs to be, on it's own, capable of quickly rendering the nuclear reactor subcritical by an adequate margin from operational states and in design basis accidents on the assumption of a single failure.	• Included in principle.
In judging the adequacy of the means of shutdown, considerations need to be given to failures arising anywhere in the plant which could prevent shutdown systems from operating.	• Included in principle.
The means of shutdown need to be adequate to prevent or withstand inadvertent increases in reactivity by insertion during the shutdown including during refueling.	• Included in principle.
Instrumentation needs to be provided and tests need to be specified to ensure that the shutdown means are always in the state stipulated for the given plant conditions.	• Included in principle.
In the design of reactivity control devices, account needs to be taken of wear-out, and the effects of radiation.	• Included in principle.
Reactor Coolant System	
Reactor coolant systems and associated auxiliary systems, controls and protection systems need to be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded in operational states. Component parts containing the reactor coolant need to be designed in such a way as to withstand the static and dynamic loads anticipated in all operational states. The reactor vessel and the pressure tubes need to be designed and constructed to be of the highest quality. The pressure retaining boundary for reactor coolant needs to be designed so that flaws are very unlikely to be initiated, and any flaws that are initiated would propagate in a regime of high resistance to unstable fracture with fast crack propagation. The design needs to reflect consideration of all conditions of the boundary material in operational states, testing, maintenance, and design basis accidents. The design of the components contained inside the reactor coolant pressure boundary needs to be such as to minimize the likelihood of failure.	• Included in principle.
Inventory Control	
Provisions need to be made for controlling the inventory and pressure of coolant to prevent exceeding specified design limits.	• Included in principle.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
Removal of Residual Heat from the Core	
Means for removing residual heat need to be provided. Interconnection and isolation capabilities need to be provided to ensure reliability of residual heat removal systems.	• Included in principle.
Emergency Core Cooling	
Core cooling needs to be provided in the event of a loss of coolant accident so as to minimize fuel damage and limit the escape of fission products from the fuel. The limiting parameters for the cladding and fuel integrity will not exceed acceptable values. Possible chemical reactions are limited to an allowable level. Alteration in the fuel and internal structural alterations will not significantly reduce the effectiveness of the means of emergency core cooling. The cooling of the core will be ensured for a sufficient time. Design features and suitable redundancy and diversity in components need to be provided. Adequate consideration needs to be given to extending the capability to remove heat from the core following a severe accident.	• Included in principle.
Inspection and Testing of Emergency Core Cooling Systems	
The emergency core cooling system needs to be designed to permit appropriate periodic inspection of important components and to permit periodic testing.	• Included in principle.
Heat Transfer to an Ultimate Heat Sink	
Systems need to be provided to transfer residual heat from structures, systems, and components important to safety to an ultimate heat sink. Reliability of the systems need to be achieved by an appropriate choice of measures. Natural phenomena and human induced events need to be taken in account in the design of the systems in the consideration of diversity of an ultimate heat sink. Adequate consideration needs to be given to extending the capability to transfer residual heat from the core to an ultimate heat sink in consideration of severe accident.	• Included in principle.
Design of the Containment System	
A containment system needs to be provided in order to ensure that any release of radioactive materials to the environment in a design basis accident. All identified design basis accidents need to be taken into account in the design of the containment system.	• Included in principle.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
Strength of the Containment Structure	
The strength of the containment structure, including access openings and penetrations and isolation valves need to be designed with sufficient safety margins on the basis of: Internal overpressure Internal underpressure Temperatures Dynamic effects Reaction forces Chemical actions Radiolytic actions Provision needs to be made to maintain the integrity of containment in a severe accident.	• Included in principle.
Capability for Containment Pressure Tests	
Containment needs to be designed to allow for pressure testing.	• Included in principle.
Containment Leakage	
Containment needs to be designed so that maximum leakage is not exceeded in design basis accidents. Containment needs to be designed and constructed so that leak rate can be tested at the design pressure. Consideration needs to be given to controlling leakage in the event of a severe accident.	• Included in principle.
Containment Penetrations	
The number of penetrations through the containment needs to be kept to a minimum. Penetrations need to meet the same design requirements as the containment structure. Resilient seals or expansion bellows need to be designed to have the capability for leak testing at design pressure. Consideration needs to be given to penetrations remaining functional in the event of severe accidents.	• Included in principle.
Containment Isolation	
Each line that penetrates the containment as part of the reactor coolant pressure boundary or that is connected directly to the containment atmosphere needs to be automatically and reliably sealable in the event of a design basis accident. Each line that penetrates the primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere needs to have at least one adequate containment isolation valve.	• Included in principle.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
Consideration needs to be given to isolation devices remaining functional during sever accident.	
Containment Air Locks	
Access to the containment needs to be through airlocks equipped with doors that are interlocked to ensure isolation during operations and accidents. Consideration needs to be given to severe accidents.	• Not included - design specific.
Removal of Heat from the Containment	
The capability to remove heat from the reactor containment needs to be ensured. Consideration needs to be given to removing heat from the containment during severe accidents.	• Included in principle.
Coverings and Coatings	
Coverings and coatings need to be selected in order to minimize interference with other safety functions and fulfill their own safety functions even with deterioration.	• Included in principle.
Supplementary Control Room	
Sufficient instrumentation and control equipment need to be available, preferably at a single location, that is physically and electrically separate from the control room such that the reactor can be shut down and maintained in a long term safe state.	• Included.
Separation of Protection and Control Systems	
Interface between the protected system and the control systems need to be prevented.	• Included in principle.
Emergency Power Supplies	
It needs to be ensured that the emergency power supply is able to supply the necessary power in any operational state or in a design basis accident. The combined means to provide emergency power needs to have a reliability and form that are consistent with all the requirements of the safety systems to be supplied. It needs to be possible to test the functional capability of the emergency power supply.	• Included in principle.
Security of Material and Facilities Requirements	
Control of Access	
Plans need to be isolated from the surroundings by suitable layout of structural elements in such a way as to be permanently controlled to guard against unauthorized access. Unauthorized access to SSCs need to be prevented.	• Included in principle.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
Containment and Exposure Requirements	
Radiation Protection and Acceptance Criteria	
In the design of plants, all actual and potential sources of radiation need to be identified, properly considered, and strictly controlled. Measures need to be taken in design to ensure that radiation protection and doses to the public and site personnel do not exceed prescribed limits and are kept as low as reasonably achievable. Designs need to have as an objective the prevention and subsequent mitigation of radiation exposures. Plant states that could potentially result in high radiation doses or radioactive release need to be restricted to a very low likelihood of occurrence.	• Included in principle.
Transport and Packaging	
Transport and packaging for fuel and radioactive waste need to be incorporated into plant designs.	• Framework currently does not address fuel handling.
Removal of Radioactive Substance	
Adequate facilities need to be provided for the removal of radioactive substances from the reactor coolant, including corrosion and fission products.	• Included in principle.
Waste Treatment and Control Systems	
Adequate systems need to be provided to treat radioactive liquid and gaseous effluents in order to keep the quantities radioactive discharges as low as reasonably achievable. Adequate systems need to be provided for the handling of radioactive wastes and for storing waste on site for extended periods of time until disposal.	• Included in principle.
Control of Release of Radioactive Liquids to the Environment	
Design needs to include suitable means to control the release of radioactive liquids to the environment.	• Included in principle.
Control of Airborne Radioactive Material	
Ventilation systems with appropriate filtration need to: Prevent unacceptable dispersion of airborne radioactive substance; Reduce the concentration of airborne radioactive substances to levels compatible with the need for access to the particular area; Keep levels of airborne radioactive substances in the plant below prescribed limits during normal, abnormal, and accident conditions; and Ventilate rooms containing inert or noxious gases without impairing the capability to control radioactive substances.	• Included in principle.

K. Completeness Check

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
Control of Release of Gaseous Radioactive Material to the Environment	
Ventilation needs to contain appropriate filtration to control the release of airborne radioactive substances to the environment. Filter systems need to be sufficiently reliable and achieve necessary retention factors.	• Included in principle.
Handling and Storage of Non-Irradiated Fuel	
Handling and storage systems for non-irradiated fuel need to be designed: To prevent criticality by a specified margin by physical means or processes; To permit appropriate maintenance, inspection, and testing of components; and To minimize the probability of loss or damage to the fuel.	• Not included. Framework scope does not currently include fuel storage.
Handling and Storage of Irradiated Fuel	
Handling and storage for irradiated fuel need to be designed: To prevent criticality by physical means; To provide adequate heat removal in operational and accident conditions; To permit inspection of irradiated fuel; To permit inspection and testing of components important to safety; To prevent dropping of spent fuel in transit; To prevent unacceptable handling stresses on the spent fuel assemblies; To adequately identify individual fuel assemblies; To control soluble absorber levels if used; To facilitate maintenance and decommissioning of the fuel storage areas and handling facilities; To facilitate decontamination of fuel handling and storage areas and equipment; and To ensure that adequate operating and accounting procedure can be implemented to prevent loss of fuel. When using a water pool system for fuel storage, the design needs to provide:	• Not included. Framework scope does not currently include fuel handling or storage.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
A means for controlling chemistry and activity of any water in which fuel is stored; A means for monitoring and controlling the water level in the fuel storage pool and for detecting leakage; and A means to prevent emptying of the pool in the event of a pipe break (anti-syphon).	
General Requirements	
Radiation protection is directed to preventing any avoidable radiation exposure and to minimize unavoidable exposures with: Appropriate layout and shielding of structures, systems, and components; Giving attention to the design of the plant and equipment so as to minimize the number and duration of human activities undertaken in radiation fields; Making provision for the treatment of radioactive materials in an appropriate form and condition; and Making arrangements to reduce the quantity and concentration of radioactive materials produced and dispersed. Account needs to be taken of the potential buildup of radiation levels with time in areas of personnel occupancy.	• Included in principle.
Design for Radiation Protection	
Suitable provision needs to be made in the design and layout of the plant to minimize exposure and contamination from all sources. The shielding design needs to be such that radiation levels in operating areas do not exceed the prescribed limits, and needs to facilitate maintenance and inspection so as to minimize exposure of maintenance personnel. Plant layout and procedures need to provide for the control of access to radiation areas and areas of potential contamination. Provision needs to be made for appropriate decontamination facilities for both personnel and equipment and for handling any radioactive waste.	• Included in principle.
Means of Radiation Monitoring	
Equipment needs to be provided to ensure that there is adequate radiation monitoring in operational and accident states. Stationary dose rate meters need to be provided for monitoring the local radiation dose rate at places routinely occupied by operating personnel. Monitors need to be provided for measuring the activity of radioactive substances in the atmosphere in those areas routinely occupied by personnel.	• Included in principle.

Table K-3 NS-R-1 comparison.

IAEA Safety Standards	Framework
Stationary equipment and laboratory facilities need to be provided for the determination in a timely manner the concentration of selected radionuclides in fluid process systems as appropriate in operational states and in accident conditions. Stationary equipment needs to be provided for monitoring the effluents prior to or during discharge to the environment. Instruments need to be provided for measuring radioactive surface contamination. Facilities need to be provided for the monitoring of individual doses to and contamination of personnel. In addition to monitoring within the plant, arrangements need to also be made to determine radiological impacts, if any, in the vicinity of the plant, with particular reference to: Pathways to the human population, including the food chain; The radiological impact, if any, on local ecosystems; The possible accumulation of radioactive materials in the physical environment; and The possibility of any unauthorized discharge routes.	
Regulation Burden Mitigation	
Equipment Outages	
Plants need to be designed such that reasonable on-line maintenance and testing of systems important to safety can be conducted without the necessity to shut down.	• Included in principle.

K.4 Comparison Against IAEA NS-R-2

Table K-4 shows the results of the comparison against IAEA document NS-R-2. Similar to IAEA document NS-R-1, NS-R-2 states the requirements as general objectives or principles and includes administrative as well as technical items. The relevant topics included in NS-R-2 are also included in Chapter 8 of the Framework.

Table K-4 NS-R-2 comparison

IAEA Safety Standards	Framework
Operating Organization	
— functions	• not included (This is a management issue.)
— responsibilities	• not included (This is a management issue.)
— staffing	• included
— procedures	• included
— interface with regulator	• included
— QA program	• included
— feedback of operator experience	• included
— physical protection	• included
— fire safety	• included
— EP	• included
Qualification and Training	
— definition of qualification needed	• included
— training program	• included
— use of simulators	• included
— AM training	• included
— Operator experience feedback	• included
Commissioning Program	
— testing	• included in principle
— baseline data collection	• included in principle
Plant Operations	
— operational limits (tech spec)	• included
— procedures	• included
— core management and fuel handling	• not included (outside current scope of Framework)
Maintenance, Testing, Surveillance and Inspection	
— periodic inspection and testing	• included
— set frequency of maintenance, inspection, and testing to ensure reliability	• included
— procedures	• included
— work planning and control	• included
— record keeping	• included
— spare parts procurement, storage and dissemination	• partially included
— restart after abnormal occurrences	• included
Plant Modifications	
— regulatory approval	• included
— work control	• included
— update documentation	• included

Table K-4 NS-R-2 comparison

IAEA Safety Standards	Framework
Radiation Protection and Waste Management — radiation protection program — waste management program — ALARA — effluent monitoring	 • included • included • included • included
Records and Reports — document control	 • included
Periodic Safety Review — update safety analysis — impact of operator experience — use of PSA	 • included (maintain PRA) • included • included
Decommissioning — funding arrangements — preparation for decommissioning	 • included • included

K.5 Comparison Against NEI 02-02

In 2002, the Nuclear Energy Institute (NEI) prepared and submitted to the NRC for information a document (NEI 02-02) describing a way to risk-inform the NRC licensing process. NEI 02-02 was written to suggest a risk-informed, performance-based alternative to 10 CFR 50, which NEI called Part 53.

The NEI document is a high-level document describing a concept, structure, approach and content for their proposed Part 53, including examples of how to develop risk-informed alternatives to 10 CFR 50. The examples provided focused on LWR technology but acknowledged that other technologies could also be addressed if a technology-neutral approach were taken. Very little technical basis was provided for the examples and there were many technical areas that were incomplete. Nevertheless, it is useful to compare the Framework topics identified in Chapter 8 against the content of NEI 02-02. This comparison is shown in Table K-5 below.

As can be seen from Table K-5, many technical items are not included in NEI 02-02. NEI 02-02 does, however, include a thorough listing of the administrative items which should be included in their proposed Part 53. It does list one item which is not included in the Framework and that is in the area of selective implementation, which is a policy issue, not a design or operational issue.

Table K-5 NEI 02-02 comparison.

Framework Topic	NEI 02-02
(A) General Topics Common to Design, Construction and Operation	
1) QA/QC	Included
2) PRA scope and technical acceptability	Minimally included
3) Uses of risk information	Minimally included
4) Integrated safety, security and preparedness	Not included
(B) Physical Protection	
1) General (10 CFR 73)	Included
2) Security performance standards	Not included
(C) Good Design Practices	
1) Plant Risk: • Frequency-Consequence curve • QHOs (including integrated risk)	Not included
2) Criteria for selection of LBEs	Included
3) LBE acceptance criteria: • frequent events (dose, plant damage) • infrequent events (dose, plant damage) • rare events (dose) • link to siting	Partially included
4) Initiating event severity	Not included
5) Safety classification and special treatment	Partially included
6) Equipment Qualification	Included
7) Licensing analysis • realistic analysis, including failure assumptions • source term	Partially included
8) Siting and site-specific considerations	Partially included
9) Use consensus design codes and standards	Not included
10) Materials qualification	Not included
11) Protection against natural phenomena	Not included
12) Dynamic effects	Not included
13) Sharing of structures, systems and components	Not included
14) Reactor shutdown and decay heat removal	Partially included
15) Barriers to release of radioactive material	Partially included

Table K-5 NEI 02-02 comparison.

Framework Topic	NEI 02-02
16) Radiological containment functional capability	Partially included
17) Radiological containment atmosphere cleanup	Not included
18) Fracture prevention of radiological containment pressure boundary	Not included
19) Electric power systems	Not included
20) Piping systems penetrating radiological containment boundary	Not included
21) Closed systems	Not included
22) Vulnerability to a single human action or hardware failure	Not included
23) Plant aging and degradation	Not included
24) Reactor inherent protection (i.e., no positive power coefficient, limit control rod worth, stability, etc.)	Partially included
25) Human factors/man-machine interface	Not included
26) Fire protection	Included
27) Control room design	Partially included
28) Alternate shutdown location	Not included
29) Reactor core flow blockage and bypass prevention	Not included
30) Reliability and availability: • establish Reliability Assurance Program • specify goals on initiating event frequency	Not included
31) Research and Development	Not included
32) Use of prototype testing	Not included
33) Combustible gas control	Not included
34) Energetic reaction control	Not included
35) Prevention of reactor coolant boundary brittle fracture	Not included
36) Reactor coolant pressure boundary	Not included
37) Reactor coolant activity monitoring and cleanup	Not included
38) I and C System • analog • digital • HMI	Not included
39) Protection of operating staff	Not included

Table K-5 NEI 02-02 comparison.

Framework Topic	NEI 02-02
40) Control of releases of radioactive materials to the environment	Not included
41) Monitoring radioactivity releases	Not included
42) Qualified analysis tools	Partially included
(D) Good Construction Practices	
1) Use accepted codes, standards, practices	Not included
2) Security during construction/fabrication	Included
3) NDE during construction/fabrication	Not included
4) Inspection during construction/fabrication	Not included
5) Testing during construction/fabrication	Not included
(E) Good Operating Practices	
1) Radiation protection	Included
2) Maintenance program	Not included
3) Personnel qualification	Not included
4) Training	Included
5) Use of procedures	Not included
6) Use of simulators	Not included
7) Staffing	Included
8) Aging management program	Included
9) Surveillance program	Included
10) ISI	Not included
11) Testing	Included
12) Technical specifications	Included
13) Emergency preparedness	Included
14) Monitoring and feedback	Included
15) Work and configuration control	Included
16) Maintain PRA	Not included
17) Fuel and replacement part quality	Not included
18) Security	Included

Table K-5 NEI 02-02 comparison.

Framework Topic	NEI 02-02
(F) Administrative	
1) Standard format and content of applications	Included
2) Change control process	Included
3) Record keeping	Included
4) Documentation control	Included
5) Reporting	Included
6) Corrective action program	Not included
7) Backfitting	Included
8) License amendments	Included
9) Exemptions	Included
10) Other legal, financial and process items	Included

K.6 Comparison Against U.K. Document ("Safety Assessment Principles for Nuclear Facilities")

In 2006 the United Kingdom (U.K.) Health and Safety Executive (H&SE) issued a document titled, "Safety Assessment Principles for Nuclear Facilities." This document is used by the Nuclear Installations Inspectorate (a branch of the H&SE) to guide regulatory decision making in their licensing process. The document contains a number of principles, each followed by detailed guidance on its implementation. The scope of the document is broader than the scope of the Framework in that it addresses waste management, decommissioning and land contamination.

The authors of this NUREG performed a high-level review of the H&SE document for comparison with the Framework. The detailed guidance for implementation of each of the principles was not included in the comparison since it contains much more detail than the Framework is intended to include. However, at the principle level, the Framework and the H&SE safety assessment principles have much in common. The list of engineering principles from the H&SE document is shown in Table K-6. As can be seen, the H&SE engineering principles cover many of the same elements as does the Framework.

Table K-6 H&SE engineering principles.

Key principles (e.g., QA)	Civil engineering
Safety classification and standards	Graphite components and structures
Equipment qualification	Safety systems
Design for reliability	Control and instrumentation of safety-related systems

Table K-6 H&SE engineering principles.

Reliability claims	Essential services
Commissioning	Human factors
Maintenance, inspection and testing	Control of nuclear matter
Aging and degradation	Containment and ventilation
Layout	Reactor core
External and internal hazards	Heat transport systems
Pressure systems	Criticality safety
Integrity of metal components and structures	

Major differences are limited to the following:

- The approach to defense-in-depth taken in the H&SE document follows more closely the IAEA approach than does this NUREG.

- The H&SE document calls for no human action in the early stages of an accident, passive safety features and the application of a single failure criterion to safety systems to address random failures. This NUREG does not propose any constraint on human actions, or passive safety features, so as to leave the designer flexibility to meet the acceptance criteria. This document relies more on PRA analysis, in lieu of the single failure criterion, to establish the number of failures to be considered in the safety analysis.

The H&SE document also contains principles on fault analysis, radiation protection, accident management and emergency planning, similar to this NUREG. However, the cutoff frequency for initiating events which must be considered in the design appears to be higher than that proposed in this NUREG.

In summary, much of what is contained in the H&SE document is consistent with the Framework. The level of detail contained in the H&SE document make it a valuable resource that could support developing detailed guidance to implement the Framework if, and when, a decision is made to develop any guidance.

K.7 References

[EPRI 2006] Electric Power Research Institute, Report #1013582, "Technical Elements of Risk-Informed, Technology-Neutral Licensing Framework for Nuclear Power Plants," December 2006.

[IAEA 2000a] International Atomic Energy Agency, "Safety of Nuclear Power Plants: Design Safety Requirements," Safety Standards Series No. NS-R-1, October, 2000.

[IAEA 2000b] International Atomic Energy Agency, "Safety of Nuclear Power Plants: Operation Safety Requirements," Safety Standards Series No. NS-R-2, October, 2000.

[NEI 2002] Nuclear Energy Research Institute, “A Risk-Informed, Performance-Based Regulatory Framework For Power Reactors,” NEI 02-02, May 2002.

[UK 2006] United Kingdom Health and Safety Executive, “Safety Assessment Principles for Nuclear Facilities,” 2006 Edition, Version 1.

APPENDIX L
STAKEHOLDER AND ACRS COMMENTS

L. STAKEHOLDER AND ACRS COMMENTS

L.1 Introduction

The Framework document was issued for public review and comment as part of the advance notice of proposed rulemaking (ANPR) regarding "Approaches to Risk-Informed and Performance-Based Requirements for Nuclear Power Reactors," (71 FR 26267) [NRC 2006]. In response to issuing this NUREG for public review and comment, the staff received the following:

- Stakeholders' comments directly responding to the questions raised in the ANPR;
- Letter submitted by the Advisory Committee on Reactor Safeguards (ACRS) to the Commission [ACRS 2007];
- Letter submitted by Dr. Graham Wallis to the Executive Director of Operations [Wallis 2007]; and
- Report issued by the Electric Power Research Institute (EPRI) addressing the Framework document [EPRI 2006].

The responses to each of the above are discussed in the subsequent sections.

L.2 Stakeholders' Comments in Response to the ANPR

In the ANPR, 10 separate topics (and associated questions) were included. Although the Framework document was a separate topic, the other topics and associated questions (and stakeholder responses) were related to the Framework. Consequently, all the comments received under each topic were reviewed and responses developed.

The topics addressed in the ANPR included the following:

- Plan to Risk-Inform 10 CFR 50
- Integration of Safety, Security, and Emergency Preparedness
- Level of Safety
- Integrated Risk
- ACRS views
- Containment Performance Standards
- The Framework
- Defense-in-Depth
- Single Failure Criterion
- Continue Individual Rulemaking to Risk-Inform 10 CFR 50 Requirements

The questions associated with each topic are provided in Section L.2.11.

In response to the ANPR, the staff received comments from 10 stakeholders (four of whom also submitted a preliminary set of comments). The stakeholders providing comments included:

- Nuclear Energy Institute (NEI) [NEI 2006a,b]
- American Society of Mechanical Engineers (ASME) [ASME 2006a,b]
- American Nuclear Society (ANS) [ANS 2006a,b]
- Institute of Electrical and Electronics Engineers (IEEE) [IEEE 2006]

- Areva NP [Areva 2006]
- General Electric Company [GE 2006]
- Westinghouse [West 2006]
- Pebble Bed Modular Reactor (Pty) Ltd. [PBMR 2006a,b]
- Strategic Teaming and Resource Sharing (STARS) Alliance [STARS 2006]
- Nuclear Equipment Forum [NEQ 2006]

A categorization of the comments received from the above organizations is provided in Section L.2.12.

Although detailed comments on each topic were provided by groups such as NEI, ASME, ANS and Areva NP, some stakeholders only provided specific comments on the ANPR plan (i.e., Topic A) or indicated that they agreed with the comments submitted by NEI.

A summary of stakeholder comments is provided below for each topic. In the ANPR, a short introduction of the issue was provided with each topic. Those introductory remarks are included below also with a summary of the questions for that topic to provide background and context for the topic.

In reviewing the stakeholders comments, the authors' response falls into one of the following categories:

- Stakeholder comment(s) is an observation and there is no need to modify the NUREG.

- Stakeholder comment(s) involves implementation of the NUREG. The authors do not necessarily disagree with the comment(s), however, any further study will be determined based on future use of the NUREG. These are noted in Appendix C for further study.

- Stakeholder comment(s) is a suggested clarification which has been included in the NUREG, as appropriate.

- Stakeholder comment(s) involves issues that the authors disagree with and no modification was made to the NUREG; the basis for the disagreement is provided.

L.2.1 ANPR Topic A: Plan to Risk-Inform 10 CFR 50 (Questions 1-7)

Issue —

The NRC proposed a plan to develop an integrated risk-informed and performance-based (RI/PB) alternative to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," that would cover power reactor applications including non-light-water reactor (non-LWR) designs. Safety, security, and preparedness will be integrated into this effort to provide one cohesive structure. This structure will ensure that the reactor regulations and staff processes and programs are built on a unified safety concept and are properly integrated so that they complement one another. Based on the above, the overall objectives of a RI/PB alternative to 10 CFR 50 are to: (1) enhance safety and security by focusing NRC and licensee resources in areas commensurate with their importance to public health and safety, (2) provide the NRC with a framework that uses risk information in an integrated manner, (3) use risk information to provide flexibility in plant design and operation while maintaining or enhancing safety and security, (4) ensure that risk-informed activities are coherently and properly integrated such that they complement one another and continue to meet the

Commission's 1995 Probabilistic Risk Assessment (PRA) Policy Statement, and (5) allow for different reactor technologies in a manner that will promote stability and predictability in the long term. The proposed plan addresses risk-informed power reactor activities and the associated guidance documents. Risk-informed activities addressing non-power reactors, nuclear materials and waste are not addressed. The NRC's proposed approach is to create a new Part in 10 CFR (10 CFR 53) that can be applied to any reactor technology as an alternative to 10 CFR 50. Two major tasks are proposed: (1) develop the technical basis for rulemaking for 10 CFR 53, and (2) develop the regulations and associated guidance for 10 CFR 53.

As part of the ANPR, stakeholders were asked to provide feedback on the merit of a new 10 CFR 53, whether it should be technology neutral, when would it be needed, and whether they would be willing to develop needed guidance.

Stakeholder Comments –

- In general, all the stakeholders were supportive of the plan to develop RI/PB requirements for future reactors, and indicated that the NRC should not begin rulemaking immediately.

  Response:

  Stakeholder comments are observations and there is no need to modify the NUREG. However, as indicated in SECY-07-0101, the staff agrees with the stakeholders that new rulemakings are not warranted at this time. It was further noted that the results of the development of the licensing strategy for the Next Generation Nuclear Plant (NGNP) and the Pebble Bed Modular Reactor (PBMR) pre-application review will help determine how to proceed to rulemaking. It is believed this approach is appropriate, in part, because rulemaking is not needed for the near-term LWR licensing applications expected in the 2007-2010 time frame.

- Stakeholders suggested that, before initiating rulemaking, draft requirements based on the Framework should be developed and made available for information and discussion, and that the draft requirements should be tested against the licensing of a non-LWR, under 10 CFR Parts 50 and 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," as a pilot.

  Response:

  Example potential requirements have been developed and included in Appendix J. However, as indicated in SECY-07-0101, the staff agrees that draft requirements would benefit from being applied as a test case against the licensing of a non-LWR.

- Most stakeholders indicated that the NRC needs to maintain a high priority on supporting the licensing and certifications of the next generation of near-term LWRs.

  Response:

  Stakeholder comments are observations and there is no need to modify the NUREG.

- Stakeholders had mixed views as to whether requirements should be technology neutral or specific and thought a "test case" applying draft requirements would help inform this topic.

  Response:

  These comments involve further study to support implementation of the NUREG. The authors do not necessarily disagree with the comment(s), however, any further study or effort will be determined based on future use of the NUREG. These are noted in Appendix C for further study.

L.2.2 ANPR Topic B: Integration (of Safety, Security and Emergency Preparedness (Questions 8-12))

Issue —

The Commission believes that safety, security, and emergency preparedness should be integrated in developing a RI/PB set of requirements for nuclear power reactors (i.e., in this context, 10 CFR 53). The NRC has proposed to establish security performance standards for new reactors (see SECY-05-0120, "Security Design Expectations for New Reactor Licensing Activities" July 6, 2005, ML051100233). Under the proposed approach, nuclear plant designers would analyze and establish, at an earlier stage of design, security design aspects so that there would be a more robust and effective (intrinsic) security posture and less reliance on operational (extrinsic) security programs (guns, guards, and gates). This approach takes advantage of making plants more secure by design rather than by adding security components on after the design is complete.

As part of the ANPR questions, stakeholders were asked to provide feedback on the proposed approach for integration, views on principles for security standards, and if security and emergency preparedness should be risk-informed.

Stakeholder Comments –

- Stakeholders indicated that insights derived from risk assessments on safety should be used to develop a coordinated approach to safety and security. One stakeholder argued that security requirements should credit future designs that have low intrinsic risks and that there was a need to risk-inform security. All stakeholders, however, agreed that emergency preparedness should be risk-informed based on all available risk information and insights, using a graded approach.

  Response:

  Stakeholder comments are observations and there is no need to modify the NUREG.

- Two stakeholders argued strongly that safety and security should be integrated in the management process.

  Response:

  Stakeholder comments are observations and there is no need to modify the NUREG.

- A majority of stakeholders expressed concern with the integration of safety, security, and emergency preparedness:

  — One stakeholder stated that it is premature to integrate safety and security until the several ongoing rulemakings on security are complete.

  Response:

  Integration of safety and security are addressed in Appendix J as an issue to be pursued as part of implementation. The results of any rulemakings will be taken into consideration at the time of implementation.

  — Another concern expressed was that the public exchange of information on a new reactor's safety design philosophy could lead to compromising its protection against threats to physical security and vice versa. It is believed that, full integration of safety and security could conflict with the need to limit public discussion of strategies to protect against threats to security so that the inherent security in a given plant design is not compromised.

  Response:

  The authors disagree with the comment. Integration of safety and security is not the same as making safety and security information public. The authors agrees that it would be inappropriate to release security information publicly; sensitive information would continue to be classified.

  — Most stakeholders indicated that application of PRA methods to the issue of security risk was premature because of the large uncertainties involved.

  Response:

  The authors disagree with the comment. The large uncertainties are associated with the initiating event frequencies. In fact, these vary day by day and are the province of security experts. Application of PRA methods, given a threat, has a long history in security analysis and does not involve greater uncertainties than in safety analyses.

L.2.3 ANPR Topic C: Level of Safety (Questions 13-20)

Issue —

The staff, in SECY-05-0130, "Policy Issues Related to New Plant Licensing and Status of the Technology-Neutral Framework for New Plant Licensing," issued July 21, 2005, ML051670388, proposed options for establishing a regulatory standard that would be applied during licensing to enhance safety for new plants consistent with the Commission's policy statement, "Regulation of Advanced Nuclear Power Plants." Four options were evaluated which included: (1) perform a case-by-case review, (2) use the Quantitative Health Objectives (QHOs) in the Commission's policy statement on "Safety Goals for the Operation of Nuclear Power Plants," (3) develop other risk objectives for the acceptable level of safety, and (4) develop new QHOs. The NRC is soliciting stakeholder views on these options. In the ANPR, stakeholders were also asked to discuss any alternative options and their benefits. Subsidiary risk objectives could also be developed to implement the Commission's expectation regarding enhanced safety for new plants. These subsidiary risk objectives could be a useful way to: focus more on plant design, provide quantitative criteria for accident prevention and mitigation, and provide high-level goals to assist in establishing plant system and equipment reliability and availability targets. Currently, subsidiary risk objectives of 10^{-5}/plant year and 10^{-6}/plant year that could be applicable to all reactor designs are being considered for accident prevention, i.e., preventing major fuel damage, and accident mitigation, i.e., preventing releases of radioactive material offsite such that no early fatalities occur from acute radiation doses.

As part of the ANPR questions, stakeholders were asked to provide feedback on staff options, subsidiary objectives, need for a Level 3 PRA, and if the QHOs could be met by prevention or mitigation alone.

Stakeholder Comments –

- Stakeholders agreed that the QHOs established in the Commission's Safety Goal Policy should be used to establish the minimum level of safety for new plants. No alternative options were brought up by the stakeholders.

  Response:

  Stakeholder comments are observations and there is no need to modify the NUREG.

- One stakeholder opined that, while the use of QHOs is appropriate, margins and defense-in-depth vis a vis the QHOs should be a design-specific consideration.

  Response:

  These comments involve further study to support implementation of the NUREG. The authors do not necessarily disagree with the comment(s), however, any further study or effort will be determined based on future use of the NUREG. These are noted in Appendix C for further study.

- Overall, there was no consensus concerning the establishment of subsidiary risk objectives similar to core-damage frequency (CDF) and large, early release frequency (LERF) for LWRs. Some stakeholders indicated that subsidiary risk objectives should be established to facilitate the development of industry standards and regulatory guidance, and to provide an approach for demonstrating that a new plant meets the QHOs without performing a Level III risk assessment. Most stakeholders stated that subsidiary risk objectives should be established in technology-specific regulatory guidance. Two stakeholders argued that it is not technically possible to develop meaningful technology-neutral subsidiary risk objectives that could be successfully applied to gas-cooled reactors. Most stakeholders questioned the need to establish subsidiary risk objectives for accident prevention and accident mitigation; rather, it was expressed that both preventive and mitigative measures should be taken into account when evaluating the capability of a plant to meet the QHOs.

Response:

These comments involve further study to support implementation of the NUREG. The authors do not necessarily disagree with the comment(s), however, any further study or effort will be determined based on future use of the NUREG. These are noted in Appendix C for further study.

L.2.4 ANPR Topic D: Integrated Risk (Questions 21-23)

Issue —

For new plant licensing, potential applicants have indicated interest in locating new plants at new and existing sites. In addition, potential applicants have indicated interest in locating multiple (or modular) reactor units at new and existing sites. The NRC is evaluating the issue of integrated risk. The staff, in SECY-05-0130, evaluated three options that included: (1) no consideration of integrated risk, (2) quantification of integrated risk at the site only from new reactors (i.e., the integrated risk would not consider existing reactors), and (3) quantification of integrated site risk for all reactors (new and existing) at that site. Another aspect of this issue is the level of safety associated with the integrated risk. The NRC is presently considering whether the integrated risk should be restricted to the same level that would be applied to a single reactor. If this approach were adopted, the integrated risk resulting from adding multiple reactors to an existing site would not be allowed to exceed the level of safety expressed by the QHOs in the Commission's Safety Goal Policy Statement.

As part of the ANPR questions, stakeholders were asked to provide feedback on staff options and whether a minimum risk threshold should be specified in the regulations.

Stakeholder Comments –

- There was no consensus concerning the consideration of integrated risk. One stakeholder argued that comparisons to the QHOs must include all site risks (existing plants and new plants). However, other stakeholders observed that plants are licensed individually, and that the NRC has traditionally considered risk on a per-reactor basis.

  Response:

  These comments involve further study to support implementation of the NUREG. The authors do not necessarily disagree with the comment(s), however, any further study or effort will be determined based on future use of the NUREG. These are noted in Appendix C for further study.

- There was some support for comparing the integrated risk from all new plants at a site to the QHOs.

  Response:

  These comments involve further study to support implementation of the NUREG. The authors do not necessarily disagree with the comment(s), however, any further study or effort will be determined based on future use of the NUREG. These are noted in Appendix C for further study.

L.2.5 ANPR Topic E: ACRS Views (Question 24)

Issue —

In a September 21, 2005 letter, the ACRS raised a number of questions related to new plant licensing. The ACRS discussed issues related to requiring enhanced safety and how the risk from multiple reactors at a single site should be accounted for. The details of the ACRS discussion could be found in the September 21, 2005, letter, that was attached to the ANPR. The Commission, in a September 14, 2005 Staff Requirements Memorandum (SRM), directed the staff to consider ACRS views in developing a subsequent notation vote paper addressing these policy issues.

As part of the ANPR, the ACRS letter was included and stakeholders were asked to provide feedback on the ACRS views.

Stakeholder Comments –

- Stakeholders expressed a variety of opinions about the views of the ACRS on the appropriate level of safety and treatment of integrated risk for new plants. One stakeholder commented that the ACRS had raised important and relevant points about these issues that warrant further consideration by the staff.

  Response:

  These comments involve further study to support implementation of the NUREG. The authors do not necessarily disagree with the comment(s), however, any further study or effort will be determined based on future use of the NUREG. Level of safety and integrated risk are noted in Appendix C for further study.

- Another stakeholder concluded that the points raised by the ACRS had already been adequately addressed in the Framework.

  Response:

  Stakeholder comments are observations and there is no need to modify the NUREG.

L.2.6 ANPR Topic F: Containment Performance Standards (Questions 25-30)

Issue —

The Commission has directed the staff to develop options for containment functional performance requirements and criteria that take into account such features as core, fuel, and cooling system design.

As part of the ANPR questions, stakeholders were asked to provide feedback regarding how to define containment, its safety functions, and its functional performance standards, including physical security considerations.

Stakeholder Comments –

- Stakeholders generally believed that containment performance standards should be developed, at a high level, on a technology-neutral basis and should be viewed as a plant-wide safety function, not a predetermined specific barrier or set of barriers separate from other aspects of the design. Stakeholders believed that technology-specific guidance could then be provided to support implementation, and that the resulting design features that perform the containment function would be design specific and could range from pressure retaining to non-pressure retaining structures, provided the release criteria are met. In addition, stakeholders believed that risk considerations should be used in developing the requirements and implementing guidance so as to facilitate design-specific implementation.

  <u>Response</u>:

  These comments involve further study to support implementation of the NUREG. The authors do not necessarily disagree with the comment(s), however, any further study or effort will be determined based on future use of the NUREG. These are noted in Appendix C for further study.

- Stakeholders also stated that the physical security safety functions of containment should be design specific, as the design (fuel characteristics, below ground siting, etc.) will play a major role and will need to be considered.

  <u>Response</u>:

  The authors do not disagree with the comment; however, it involves containment, which is a policy issue, and is noted in Appendix C for further study.

- Stakeholders considered that the frequency categories and the process for selection of licensing basis events contained in the Framework were reasonable for assessing containment functional performance; however, the application of the Framework defense-in-depth principles should be applied on a design specific basis. Other comments were made with respect to how the Framework uses an frequency-consequence (F-C) curve and the QHOs with regard to containment performance.

  <u>Response</u>:

  The authors disagree with the comments and no modifications were made to the NUREG. It is the authors' view that plant designs should have a containment functional capability to prevent an unacceptable release of radioactive material to the public. The principle of defense- in-depth should ensure that, regardless of the features incorporated in the plant to prevent an unacceptable release of radioactive material from the fuel and the reactor coolant system (RCS), additional means should be employed to prevent an unacceptable release to the public should a release from the fuel and RCS occur that has the potential to exceed the dose acceptance criteria.

  The purpose of this defense-in-depth principle is to protect against unknown phenomena and threats, i.e., to compensate for completeness uncertainty affecting the magnitude of the source term. In doing so, threats from selected low probability, but credible, events with the potential for a large source term and a significant radionuclide release to the environs are considered. This performance-based principle is consistent with 10 CFR 50.34 (a) (1) (i) which states that special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. 10 CFR 50.34 (a) (1) (i) requires that a fission product release from the core into the containment be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, rather than limiting the selection of events from the design-basis frequency category.

Nonetheless, this issues is a policy issue for Commission consideration and is discussed in Appendix C.

L.2.7 ANPR Topic G: The Framework (Questions 31-54)

The questions in the ANPR on the Framework document were divided into seven different groups. The summary of the comments are presented for each of these groups.

L.2.7.1 ANPR Topic G(a): Approach/Structure (Questions 31-34)

Issue —

In support of determining the requirements for these alternative regulations, the NRC is developing a technology-neutral Framework. This Framework provides one approach in the form of criteria and guidelines that could serve as the technical basis for 10 CFR 53 that is technology-neutral and RI/PB.

The questions pertaining to the Framework approach and structure were intended to solicit stakeholder feedback on the overall top down organization of the document, the use of the Commission's Safety Goals and defense-in-depth as the starting point for deriving a unified set of risk-derived requirements and their technical bases. Also, feedback was solicited on whether or not the Framework should now be applied to a specific reactor design and if the scope and application of the Framework are clear.

Stakeholder Comments –

- Stakeholders generally agreed that the top down organization taken in the Framework is a reasonable approach. Stakeholders generally agreed that the Framework scope and application is clear.

  Response:

  Stakeholder comments are observations and there is no need to modify the NUREG.

- Stakeholders agreed that the next step should be to apply the Framework to a specific reactor design, with an operating LWR and / or a high-temperature gas-cooled reactor (HTGR) being reasonable candidates.

  Response:

  These comments involve further study to support implementation of the NUREG. The authors do not necessarily disagree with the comment(s), however, any further study or effort will be determined based on future use of the NUREG. These are noted in Appendix C for further study.

- Stakeholders generally agreed on the unified approach taken in the Framework to derive requirements using the Commission's Safety Goals and defense-in-depth. Additional clarification was suggested in describing the process to select licensing basis events and that this question should be reassessed after the stakeholders comments are addressed and draft requirements written.

  Response:

  Stakeholder comment(s) is a suggested clarification which has been included in the NUREG, as appropriate.

L.2.7.2 ANPR Topic G(b): Emergency Preparedness (Questions 35-36)

Issue —

The Commission believes that safety, security, and emergency preparedness should be integrated. The approach in the Framework to achieve this integration is to define the safety, security, and preparedness expectations that are needed and to define protective strategies and defense-in-depth principles for each area in an integrated manner.

The questions pertaining to emergency preparedness were intended to solicit stakeholder feedback on the factors that should play a role in integrating emergency preparedness requirements for future plants and what should the emergency preparedness requirements for future plants be, including their generic and technology-specific nature.

Stakeholder Comments –

- Stakeholders suggested that risk insights could form the basis for smarter protective actions, including consideration of smaller source terms and defense-in-depth, which could significantly reduce pubic risk. One stakeholder suggested that technology-specific emergency preparedness requirements would not be practical and supported generic risk-informed requirements. Two stakeholders suggested that the detailed emergency preparedness requirements should be technology-specific and should not be based on deterministic requirements.

  Response:

  These comments involve further study to support implementation of the NUREG. The authors do not necessarily disagree with the comment(s), however, any further study or effort will be determined based on future use of the NUREG. These are noted in Appendix C for further study.

L.2.7.3 ANPR Topic G(c): Defense-in-depth (Questions 37-39)

Issue —

The core of the NRC's safety philosophy has always been the concept of defense-in-depth, and defense-in-depth remains basic to the safety, security, and preparedness expectations of the technology-neutral Framework. Defense-in-depth is the mechanism used to compensate for uncertainty. This includes uncertainty in the type and magnitude of challenges to safety, as well as in the measures taken to assure safety.

As part of the ANPR questions, stakeholders were asked to provide feedback regarding (1) whether the approach used in the Framework for how defense-in-depth treats uncertainty is well-described and reasonable, (2) whether the defense-in-depth principles are clearly stated, (3) whether additional principles would be needed, and (4) whether the guidance on safety margin should be enlarged to provide more quantitative guidance in a technology-neutral fashion.

Stakeholder Comments –

- One stakeholder agreed with the approach to defense-in-depth as it related to uncertainty, but felt that additional criteria were needed to ensure that a design feature introduced for defense-in-depth purposes also served to reduce uncertainty, as there were other motivations besides uncertainty for incorporating defense-in-depth in the design.

  Response:

  The Framework provides guidance on the issue of defense-in-depth for NRC staff. It is not meant to provide a complete set of criteria for defense-in-depth that licensees may wish to employ.

- Another stakeholder felt that the approach needed improvement to clarify the interdependence of defense-in-depth, design criteria, and protective strategies and that some aspects of defense-in-depth were more appropriately described as design principles.

  Response:

  Clarification of the relationship of defense-in-depth and the protective strategies has been added to the NUREG. However, the defense-in-depth principles in the Framework are for potential staff use in developing requirements; they are not developed as design principles. In addition, further work is expected to be performed in this area as the staff develops a policy statement on defense-in-depth for Commission consideration.

- Stakeholders generally agreed that the defense-in-depth principles were clearly stated. One stakeholder felt that a clearer definition of defense-in-depth in relation to prevention and mitigation was needed.

  Response:

  These comments involves implementation of the NUREG. The authors do not necessarily disagree with the comment(s), however, any further study or effort will be determined based on future use of the NUREG. These are noted in Appendix C for further study. In addition, the staff is initiating an effort to develop a policy statement on defense-in-depth for Commission consideration. This comment will be addressed in development of the supplementary information supporting the policy statement.

- Most stakeholders agreed that the treatment of safety margin in the Framework was reasonable. One stakeholder, however, disagreed on the ground that the treatment was inconsistent with codes and standards for addressing margins.

  Response:

  The treatment of safety margins in the Framework is meant to illustrate the concepts involved, it is not intended as a substitute for the quantitative criteria relating to margins embedded in various codes and standards.

L.2.7.4 ANPR Topic G(d): Protective Strategies (Questions 40-41)

Issue —

The Framework introduces five protective strategies as safety fundamentals that satisfy the structuralist expectations for defense-in-depth. Chapter 8 applies a logic diagram to each protective strategy to develop requirements to prevent failure, i.e., loss of defense-in-depth.

As part of the ANPR questions, stakeholders were asked to provide feedback regarding the capability of the protective strategies to defend against all challenges and to address the issue of relative importance among them.

Stakeholder Comments –

- Stakeholders agreed that the protective strategies defend against all challenges so far foreseen. They would like some flexibility in setting the emphasis among them, depending on design capabilities.

  Response:

  Stakeholder comment is an observation and there is no need to modify the NUREG.

- One stakeholder suggested restructuring level of safety, defense-in-depth, and protective strategies into a set of fundamental technology-neutral safety principles (FSP) and associated fundamental design principles (FDP), and suggested a candidate set of each.

  Response:

  The authors do not believe a restructuring is warranted. The hierarchal structure of the Framework evolved over time. The authors believe the current structure, relative to its objective and based on comments from numerous other stakeholders, is appropriate.

L.2.7.5 ANPR Topic G(e): Probabilistic Approach (to Licensing Basis (Questions 42-46))

Issue —

In the Framework, risk information is used in two basic parts of the licensing process: (1) Identification and selection of those events that are used in the design to establish the licensing basis, and (2) the safety classification of selected systems, structures, and components.

As part of the ANPR questions, stakeholders were asked to provide feedback on: (1) the basis for selection of licensing-basis events (LBEs), (2) the cut-off for the rare event frequency of 1E-7 per year, (3) the approach used to select and classify safety systems, structures, and components, (4) the approach and basis to the construction of the frequency-consequence curve, (5) the deterministic criteria for LBEs in the infrequent and rare event categories, and (6) the use of a 95% confidence value for a mechanistic source term to calculate doses for both the PRA sequences and the LBEs.

Stakeholder Comments –

- Most stakeholders agreed that the approach to LBE selection was reasonable, and that the deterministic criteria for LBEs and the use of a 95% confidence value for a mechanistic source term provided a reasonable approach.

  Response:

  Stakeholder comment is an observation and there is no need to modify the NUREG

- Most stakeholders agreed with the approach to safety classification of structures, systems and components (SSCs). A few, however, disagreed.

  — One stakeholder claimed that the approach did not credit safety equipment qualified to nuclear safety codes and standards and that it used the same failure rate for commercial equipment and qualified safety-related equipment.

    Response:

    There is no statement in the Framework document that implies that safety class equipment would not need to be qualified to nuclear safety codes and standards, nor is there any implication regarding failure rates of safety and non-safety class equipment.

  — Another stakeholder felt that the approach would be more restrictive than those in use today, because all SSCs that were needed to maintain the frequency of a sequence within the acceptable portion of the frequency-consequence curve would thereby become safety-class.

    Response:

    The Framework approach is more risk focused than the current regulatory structure for identifying the “safety-related” SSCs and results in a more realistic safety classification. With the Framework approach, SSCs are either “safety significant” or "non-safety significant," based on their role in meeting LBE acceptance criteria. In addition, non-risk significant SSCs that were classified as "safety-related" in the current regulatory structure become non-safety significant. In addition, for those SSCs that are classified as "safety significant," in the Framework, a graded approach is allowed for their special treatment based on their risk significance.

- There was a range of views expressed on the F-C curve. Some felt the approach was reasonable although some minor modifications, such as decreasing the number of steps or changing the frequency basis from per reactor-year to per plant-year to accommodate modular plants in the future, were warranted. Some stakeholders were in favor of considering a complementary cumulative distribution function (CCDF) curve. Stakeholders felt that the use of a CCDF would prevent one event sequence causing the entire design to be unacceptable. The use of the CCDF would then allow designers more flexibility.

  Response:

  These comments involve further study to support implementation of the NUREG. The authors do not necessarily disagree with the comment(s), however, any further study or effort will be determined based on future use of the NUREG. These are noted in Appendix C for further study.

- One stakeholder argued that testing and comparison with existing and advanced LWRs is needed along with greater clarity on the aggregation of sequences to develop LBEs. Another stakeholder felt that additional testing of the deterministic dose criterion was needed.

  Response:

  These comments involve further study to support implementation of the NUREG. The authors do not necessarily disagree with the comment(s), however, any further study or effort will be determined based on future use of the NUREG. Framework testing is noted in Appendix C.

L.2.7.6 ANPR Topic G(f): PRA Technical Acceptability (Questions 47-49)

Issue —

The approach proposed in the Framework requires a full-scope "living" PRA that would incorporate operating experience and performance-based requirements in the periodic re-examination of events designated as LBEs that were originally selected based on the design, and structures, systems, and components that were characterized as safety-significant.

Stakeholder Comments –

- Stakeholders generally agreed that PRA quality assurance requirements were necessary.

  Response:

  Stakeholder comment is an observation and there is no need to modify the NUREG

- Stakeholders suggested that not all requirements of 10 CFR 50, Appendix B are practical or necessary for application to PRA. One stakeholder stated that it is unclear as to whether the approach to PRA quality assurance is the same as 10 CFR 50, Appendix B, which was not written for PRA.

  Response:

  Stakeholder comments are suggested clarifications which have been included in the NUREG, as appropriate. Section 7.3.2, “Quality Assurance Criteria,” list the applicable quality control requirements for a PRA that is supporting a Framework analysis. This list was derived from 10 CFR 50, Appendix B which was customized to reflect the unique quality requirements of a PRA.

- Stakeholders suggested that NRC Regulatory Guide 1.174 provides a discussion of the elements of Appendix B that would generally be applicable to the PRA and should be considered adequate for a PRA supporting the Framework.

  Response:

  The authors disagree with the comments. The integration of PRA into the design and licensing process creates new challenges in the construction and maintenance of PRAs, and causes completeness, defensibility and transparency to be more important than in the past. Guidance that is included in Regulatory Guide 1.174 is for using risk information in support of licensee-initiated licensing basis changes. It is written with the expectation that PRA is one element in a multi-element risk-informed decision process. The regulatory guide includes pertinent quality control expectations and lists four specific provisions that describe methods acceptable to NRC staff. The requirements listed in Section 7.3.2 of the Framework are adapted from the current 10 CFR 50, Appendix B and include additional requirements to those listed in Regulatory Guide 1.174. These additional requirements reflect the increased role the PRA as part of the licensing basis and include: a PRA quality assurance program applicable to the life cycle of the PRA, measures to ensure that applicable requirements and standards are specified and included in the development and maintenance of the PRA, measures to ensure the control of PRA interfaces, measures to ensure that conditions adverse to PRA quality are promptly identified and corrected, and measures to ensure that a comprehensive system of planned and periodic audits is carried out to verify compliance.

- One stakeholder indicated that it is important that the deterministic elements of the approach be applied in a manner that the licensing basis is not sensitive to the kind of PRA update changes that can be expected. The stakeholder further stated that in applying the process of 10 CFR 50.69, it is noted that the nature of the PRA change evaluation process might be different than simply calculating changes in CDF and LERF as these risk metrics may not be used for a given new reactor. If that is accounted for by the process under 50.59, then the stakeholder agrees that this approach is reasonable.

Response:

Stakeholder comments are observations and there is no need to modify the NUREG.

Changes that reduce design margin but do not impact the Framework's regulatory safety margin will not require a re-assessment of the LBEs or the defense-in-depth measures. Designs that have large design margins will be less sensitive to PRA update changes than those with small design margins.

It should be noted that 10 CFR 50.69, "Risk-informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors," is not directly applicable to the Framework. 10 CFR 50.69 risk categorization considers design bases functions and functions credited for mitigation and prevention of severe accidents, and PRA functions; and employs a four-quadrant safety categorization process. The Framework uses the PRA to determine the design, mitigation and prevention functions and uses two categories to classify SSCs: safety significant and non-safety significant.

Section 7.3.9, "Configuration Control," discusses a process similar to 10 CFR 50.59 where proposed changes would be evaluated consistent with the Framework's acceptance criteria prior to implementation of the proposed change.

- One stakeholder indicated that a major concern is that the available consensus standards from which to draw supporting technical requirements are LWR specific and are highly focused on operation plants for which the as-built and as-operated design and operational characteristics are well known.

  Response:

  These comments involve further study to support implementation of the NUREG. The authors do not necessarily disagree with the comment(s), however, any further study or effort will be determined based on future use of the NUREG. These are noted in Appendix C for further study.

- Stakeholders suggested that consideration be given to establishing approaches which are not fully quantitative, similar to those used for advanced light-water reactor (ALWR) designs certified using 10 CFR 52, for addressing hazards such as seismic and other external hazards where PRA is not needed to demonstrate an adequate safety case.

  Response:

  These comments involve further study to support implementation of the NUREG. The authors do not necessarily disagree with the comment(s), however, any further study or effort will be determined based on future use of the NUREG. These are noted in Appendix C for further study.

- Stakeholders suggested that some hazards may have such minor potential consequences that the quantitative frequency of an anticipated operational occurrence (AOO) or design-basis event (DBE) is not important. That is, the hazard can be screened from the PRA and treated deterministically.

Response:

Comment is a suggested clarification which has been included in the NUREG, as appropriate. The Framework was always intended to apply to accidents rather than routine operations, including minor spills. The text has been clarified. Future trials may lead to sharpened definition of scenarios to be included in the PRAs for new reactors..

L.2.7.7 ANPR Topic G(g): Process to Develop Requirements (Questions 50-54)

Issue —

Chapter 8 describes and applies a process to identify the topics which the requirements must address to ensure the success of the protective strategies and administrative controls. This process is based upon: developing and applying a logic diagram for each protective strategy to identify the pathways that can lead to failure of the strategy and then, through a series of questions, identify what needs to be done to prevent the failure; applying the defense-in-depth principles from Chapter 4 to each protective strategy; developing and applying a logic diagram to identify the needed administrative controls; and providing guidance on how to write the requirements.

The questions pertaining to the process to develop requirements were intended to solicit stakeholder feedback on the clarity of the process, its approach and adequacy, the list of topics identified as needing requirements and the scope and results of the completeness check made on the list of topics.

Stakeholder Comments –

- Stakeholders generally agreed the process for developing requirements is clear and reasonable.

  Response:

  Stakeholder comment is an observation and there is no need to modify the NUREG.

- Stakeholders generally agreed that the list of topics needing requirements is reasonable, but should be reassessed after it is tested on an actual design.

  Response:

  These comments involve further study to support implementation of the NUREG. While the authors agree with the comments, any further study or effort will be determined based on future use of the NUREG. These are noted in Appendix C for further study. In addition, as indicated in SECY-07-0101, the staff agrees that draft requirements would benefit from being applied as a test case against the li censing of a non-LWR.

- Two stakeholders suggested that the completeness check be expanded to include a check against ANS, ASME, ASCE and IEEE standards.

  Response:

  The authors disagree. The appropriate codes and standards need to be developed before a check can be performed.

- A third stakeholder suggested a group of independent experts be assembled to try the process for reproducibility.

  Response:

  While, the authors agree with the comment, this effort would be addressed in support of implementation of the NUREG, which is noted in Chapter 8.

- Two stakeholders suggested that justification be provided for excluding any item found via the completeness checks.

  Response:

  While, the authors agree with the comment, this effort would be addressed in support of implementation of the NUREG, which is noted in Appendix C.

- One stakeholder noted that the use of any 10 CFR 50 requirements in the set of risk-informed requirements developed using the Framework, should be based on their being technology-neutral and compatible with the risk-informed nature of the Framework.

  Response:

  Stakeholder comment is an observation and is in agreement with the approach in the Framework.

L.2.8 ANPR Topic H: Defense-in-Depth (Questions 55-59)

Issue —

In SECY-03-0047, "Policy Issues Related to Licensing Non-Light-Water Reactor Designs," issued March 28, 2003 (ML030160002), the staff recommended that the Commission approve the development of a policy statement or description (e.g., a white paper) on defense-in-depth for nuclear power plants to describe: the objectives of defense-in-depth (philosophy); the scope of defense-in-depth (design, operation, etc.); and the elements of defense-in-depth (high level principles and guidelines). The policy statement or description would be technology neutral and risk informed and would be useful in providing consistency in other regulatory programs (e.g., Regulatory Analysis Guidelines). In the SRM to SECY-03-0047, issued June 26, 2003, the Commission directed the staff to consider whether it can accomplish the same goals in a more efficient and effective manner by updating the Commission policy statement, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," to include a more explicit discussion of defense-in-depth, risk-informed regulation, and performance-based regulation.

As part of the ANPR questions, stakeholders were asked to provide feedback on whether a better defense-in-depth definition for future plants should be included as a separate policy statement, a revision to the PRA policy statement, or as an update to Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," and whether such a description should be completed on the same schedule as 10 CFR 53.

Stakeholder Comments –

- Most stakeholders felt that a new policy statement on defense-in-depth for future plants was needed. A holistic and technology-neutral defense-in-depth statement that recognizes the role of inherent safety and passive approaches, in addition to the traditional use of redundant and diverse active systems, would be helpful in advance of developing the requirements of a new 10 CFR 53. The Framework definition was regarded as a good start but further iteration was advocated by a number of stakeholders. The policy statement should address the interdependency between defense-in-depth, protective strategies, and design criteria in relation to the safety margins incorporated in a new design. Because the scope of defense-in-depth is broader than just PRA, stakeholders felt a separate policy statement on defense-in-depth is needed. The stakeholders also noted that modifying Regulatory Guide 1.174 for new plants may be difficult because it is focused on existing deterministic requirements as well as LWR risk metrics like CDF and LERF.

  Response:

  The Commission, in their SRM in response to SECY-03-0047, agreed that a policy statement on defense-in-depth should be developed. In SECY-07-0101, it was noted that the staff is initiating an effort to develop a policy statement on defense-in-depth for Commission consideration which was approved by the Commission in the subsequent SRM. This comment will be addressed in development of the supplementary information supporting the policy statement.

L.2.9 ANPR Topic I: Single Failure Criterion (Questions 60-63)

Issue —

In SECY-05-0138, "Risk-Informed and Performance-Based Alternatives to the Single-Failure Criterion," dated August 2, 2005, (ML051950619) the staff forwarded to the Commission a draft report entitled, "Technical Report to Support Evaluation of a Broader Change to the Single Failure Criterion," and recommended to the Commission that any followup activities to risk-inform the single-failure criterion (SFC) should be included in the activities to risk-inform the requirements of 10 CFR 50. The Commission directed the staff to seek additional stakeholder involvement. The report provides the following options: (1) maintain the SFC as is, (2) risk-inform the SFC for design bases analyses, (3) risk-inform the SFC based on safety significance, and (4) replace the SFC with risk and safety function reliability guidelines.

As part of the ANPR questions, stakeholders were asked to provide feedback on any other options for risk-informing the SFC that they wished to be considered.

Stakeholder Comments –

- With one exception, all stakeholders agreed that the SFC should be eliminated and/or risk informed as part of the broader effort to risk-inform the regulations as follows:

  — Some stakeholders indicated that the general approach of using the PRA along with the F-C curve and licensing basis events eliminates the need for any kind of arbitrary redundancy requirement like the SFC.

  Response:

  Stakeholder comment is an observation and there is no need to modify the NUREG.

  — Another stakeholder felt that the use of PRA to risk-inform based on safety significance was also an acceptable approach to eliminating the

  Response:

  Stakeholder comment is an observation and there is no need to modify the NUREG.

  — One stakeholder felt that the SFC should be preserved and maintained until such time as standards committees revised their current nuclear codes and standards based on risk insights.

  Response:

  The SFC as it appears currently in 10 CFR 50 is not being revised. As noted in SECY-07-0101, the staff proposed that "the NRC should not undertake new risk-informed and performance-based revisions of 10 CFR 50 until specific rules are identified as needed. This approach will allow industry and the NRC to focus resources on maintaining the safety of existing reactors and on the expedient licensing of new reactors to existing requirements. The staff will propose candidate rulemakings after the staff and industry have had time to identify appropriate candidates." Further, any revisions carried out by code and standard committees are outside the scope of this NUREG.

- Some stakeholders indicated that changes to the SFC should be carried out as part of the effort to develop the proposed 10 CFR 53, while others felt that these efforts should be pursued separately (i.e., a separate 10 CFR 50 rulemaking).

  Response:

  Stakeholder comment is an observation and there is no need to modify the NUREG. However, with regard to changing the SFC in the current 10 CFR 50, (as noted in SECY-07-0101), the staff will consider this option once the ongoing rulemaking efforts are completed (see response to Topic J).

L.2.10 ANPR Topic J: Continue Individual Rulemaking (to Risk-Inform 10 CFR 50 Requirements (Questions 64-67))

Issue —

The NRC has for some time been revising certain provisions of 10 CFR 50 to make them more RI/PB. Examples are: (1) a revision to 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants;" (2) a revision of 10 CFR 50.48, "Fire Protection," to allow licensees to voluntarily adopt National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition" (NFPA 805); and (3) issuance of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," as a voluntary alternative set of requirements. These actions have been effective but they required extensive NRC and industry efforts to develop and implement. The NRC plans to continue the current risk-informed rulemaking actions, e.g., 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," that are ongoing, and would undertake new risk-informed rulemaking only on an as-needed basis.

As part of the ANPR questions, stakeholders were asked to provide input on whether to initiate risk-informing other regulations in 10 CFR 50, which regulations and when. Stakeholders were also asked whether to risk-inform specific regulatory guides, which ones and when.

Stakeholder Comments –

- The majority of the stakeholders who responded to this topic were in general agreement that the priority focus should be on completing ongoing efforts on specific 10 CFR Parts 50 and 52 rulemakings. With regard to future endeavors, stakeholders provided views in three areas:

    — Future rulemakings on select regulations in 10 CFR 50 – Some stakeholders commented that it was not cost-beneficial to undertake security rulemakings and that NRC should wait on the completion of the first reviews on the combined construction and operating licenses, other stakeholders thought it was too difficult to make a recommendation until successful implementation of the 10 CFR 50.69 and 10 CFR 50.46a actions.

    — Revising the supporting regulatory guides to be RI/PB – Some stakeholders suggested NRC should complete including RI/PB considerations in the ongoing revisions to the regulatory guides, others suggested that NRC should make no RI/PB changes as long as the underlying regulation is deterministic, and others suggested developing a policy statement for achieving a RI/PB regulation.

    — Time frame for initiating any new endeavors – Some stakeholders suggested NRC should start new endeavors immediately, while other stakeholders suggested waiting until successful implementation of ongoing efforts had been demonstrated.

Response:

In SECY-07-0101, the staff noted that (1) new rulemakings are not warranted at this time., (2) the NRC should not undertake new risk-informed and performance-based revisions of 10 CFR 50 until specific rules are identified as needed, (3) this approach will allow industry and the NRC to focus resources on maintaining the safety of existing reactors and on the expedient licensing of new reactors to existing requirements, and (4) the staff will propose candidate rulemakings after the staff and industry have had time to identify appropriate candidates.

L.2.11 ANPR Questions

In the ANPR, 10 topics were included with a set of questions with each topic. The detailed questions in the ANPR are provided below in Table L-1.

Table L-1 Questions in ANPR.

Topic A. Plan	
1.	Is the proposed plan to make a risk-informed and performance-based alternative to 10 CFR 50 reasonable? Is there a better approach than to create an entire new 10 CFR 53 to achieve a risk-informed and performance-based regulatory Framework for nuclear power reactors? If yes, please describe the better approach?
2.	Are the objectives, as articulated above in the proposed plan section, understandable and achievable? If not, why not? Should there be additional objectives? If so, please describe the additional objectives and explain the reasons for including them.
3.	Would the approach described above in the proposed plan section accomplish the objectives? If not, why not and what changes to the approach would allow for accomplishing the objectives?
4.	Would existing licensees be interested in using risk-informed and performance-based alternative regulations to 10 CFR 50 as their licensing basis? If not, why not? If so, please discuss the main reasons for doing so.
5.	Should the alternative regulations be technology-neutral (i.e., applicable to all reactor technologies, e.g., light-water reactor or gas-cooled reactor), or be technology-specific? Please discuss the reasons for your answer. If technology-specific, which technologies should receive priority for development of alternative regulations?
6.	When would alternative regulations and supporting documents need to be in place to be of most benefit? Is it premature to initiate rulemaking for non-LWR technologies? If so, when should such an effort be undertaken? Could supporting guidance be developed later than the alternative regulations, e.g. phased in during plant licensing and construction?
7.	The NRC encourages active stakeholder participation through development of proposed supporting documents, standards, and guidance. In such a process, the proposed documents, standards, and guidance would be submitted to and reviewed by NRC staff, and the NRC staff could endorse them, if appropriate. Is there any interest by stakeholders to develop proposed supporting documents, standards, or guidance? If so, please identify your organization and the specific documents, standards, or guidance you are interested in taking the lead to develop?

L. Stakeholder and ACRS Comments

Table L-1 Questions in ANPR.

B.	Integration of Safety, Security, and Emergency Preparedness
8.	In developing the requirements for this alternative regulatory Framework, how should safety, security, and emergency preparedness be integrated? Does the overall approach described in the technology-neutral Framework clearly express the appropriate integration of safety, security, and preparedness? If not, how could it better do so?
9.	What specific principles, concepts, features or performance standards for security would best achieve an integrated safety and security approach? How should they be expressed? How should they be measured?
10.	The NRC is considering rulemaking to require that safety and security be integrated so as to allow an easier and more thorough understanding of the effects that changes in one area would have on the other and to ensure that changes with unacceptable impacts are not implemented. How can the safety-security interface be better integrated in design and operational requirements?
11.	Should security requirements be risk-informed? Why or why not? If so, what specific security requirements or analysis types would most benefit from the use of Probabilistic Risk Assessment (PRA) and how?
12.	Should emergency preparedness requirements be risk-informed? Why or why not? How should emergency preparedness requirements be modified to be better integrated with safety and security?
C.	Level of Safety
13.	Which of the options in SECY-05-0130 with respect to level of safety should be pursued and why? Are there alternative options? If so, please discuss the alternative options and their benefits.
14.	Should the staff pursue developing subsidiary risk objectives? Why or why not? Are there other uses of subsidiary risk objectives that are not specified above? If so, what are they?
15.	Are the subsidiary risk objectives specified above reasonable surrogates for the QHOs for all reactor designs?
16.	Should the latent fatality QHO be met by preventive measures alone without credit for mitigative measures, or is this too restrictive?
17.	Are there other subsidiary risk objectives applicable to all reactor designs that should be considered? What are they and what would be their basis?
18.	Should a mitigation goal be associated with the early fatality QHO or should it be set without credit for preventive measures (i.e., assuming major fuel damage has occurred)?
19.	Should other factors be considered in accident mitigation besides early fatalities, such as latent fatalities, late containment failure, land contamination, and property damage? If so, what should be the acceptance criteria and why?
20.	Would a level 3 PRA analysis (i.e., one that includes calculation of offsite health and economic effects) still be needed if subsidiary risk objectives can be developed? For a specific technology, can practical subsidiary risk objectives be developed without the insights provided by level 3 PRAs?
D.	Integrated Risk
21.	Which of the options in SECY-05-0130 with respect to integrated risk should be pursued and why? Are there alternative options? If so, what are they?
22.	Should the integrated risk from multiple reactors be considered? Why or why not?
23.	If integrated risk should be considered, should the risk meet a minimum threshold specified in the regulations? Why or why not?

Table L-1 Questions in ANPR.

E.	ACRS Views on Level of Safety and Integrated Risk
24.	Should the views raised in the ACRS letter and by various members of the Committee be factored into the resolution of the issues of level of safety and integrated risk? Why or why not?
F.	Containment Functional Performance Standards
25.	How should containment be defined and what are its safety functions? Are the safety functions different for different designs? If so, how?
26.	Should the containment functional performance standards be design and technology specific? Why or why not?
27.	What approach should be taken to develop technology-neutral containment performance standards that would be applicable to all reactor designs and technologies? Should containment performance be defined in terms of the integrated performance capability of all mechanistic barriers to radiological release or in terms of the performance capability of a means of limiting or controlling radiological releases separate from the fuel and reactor pressure boundary barriers?
28.	What plant physical security functions should be associated with containment and what should be the related functional performance standards?
29.	How should PRA information and insights be combined with traditional deterministic approaches and defense-in-depth in establishing the proposed containment functional performance requirements and criteria for controlling radiological releases?
30.	How should the rare events in the range 10^{-4} to 10^{-7} per year be considered in developing the containment functional performance requirements and criteria? Should events less than 10^{-7} per year in frequency be considered in developing the containment functional performance requirements and criteria?
G.	Technology-Neutral Framework
G(a)	Approach/Structure
31.	Is the overall top-down organization of the Framework, as illustrated in Figure 2-6 a suitable approach to organize the approach for licensing new reactors? Does it meet the objectives and principles of Chapter 1? Can you describe a better way to organize a new licensing process?
32.	Do you agree that the Framework should now be applied to a specific reactor design? If not, why not? Which reactor design concept would you recommend?
33.	The unified safety concept used in the Framework is meant to derive regulations from the Safety Goals and other safety principles (e.g., defense-in-depth). Does this approach result in the proper integration of reactor regulations and staff processes and programs such that regulatory coherence is achieved? If not, why not?
34.	The Framework is proposing an approach for the technical basis for an alternative risk-informed and performance-based 10 CFR 50. The scope of 10 CFR 50 includes sources of radioactive material from reactor and spent fuel pool operations. Similarly, the Framework is intended to apply to this same scope. Is it clear that the Framework is intended to apply to all of these sources? If not, how should the Framework be revised to make this intention clear?

Table L-1 Questions in ANPR.

G(b)	Emergency Preparedness
35.	What role should the following factors play in integrating emergency preparedness requirements (as contained in 10 CFR 50.47) in the overall Framework for future plants: • the range of accidents that should be considered? • the extent of defense-in-depth? • operating experience? • federal, state, and local authority input and acceptance? • public acceptance? • security-related events?
36.	What should the emergency preparedness requirements for future plants be? Should they be technology-specific or generic regardless of the reactor type?
G(c)	Defense-in-Depth
37.	Is the approach used in the Framework for how defense-in-depth treats uncertainties well described and reasonable? If not, how should it be improved?
38.	Are the defense-in-depth principles discussed in the Framework clearly stated? If not, how could they be better stated? Are additional principles needed? If so, what would they be? Are one or more of the stated principles unnecessary? If so, which principles are unnecessary and why are they unnecessary?
39.	The Framework emphasizes that sufficient margins are an essential part of defense-in-depth measures. The Framework also provides some quantitative margin guidance with respect to LBEs in Chapter 6. Should the Framework provide more quantitative guidance on margins in general in a technology-neutral way? What would be the nature of this guidance?
G(d)	Protective Strategies
40.	The Framework stresses that all of the Protective Strategies must be included in the design of a new reactor but it does not discuss the relative emphasis placed on each strategy compared to the others. Are there any conditions under which any of these protective strategies would not be necessary? Should the Framework contain guidelines as to the relative importance of each strategy to the whole defense-in-depth application?
41.	Are the protective strategies well enough defined in terms of the challenges they defend against? If not, why not? Are there challenges not protected by these five protective strategies? If so, what would they be?
G(e)	Probabilistic Approach to Licensing Basis
42.	Is the approach to and the basis for the selection LBEs reasonable? If not, why not? Is the cut-off for the rare event frequency at 1E-7 per year acceptable? If not, why not? Should the cut-off be extended to a lower frequency?
43.	Is the approach used to select and to safety classify structures, systems, and components reasonable? If not, what would be a better approach?
44.	Is the approach and basis to the construction of the proposed frequency-consequence (F-C) curve reasonable? If not, why not?
45.	Are the deterministic criteria proposed for the LBEs in the various frequency categories reasonable from the standpoint of assuring an adequate safety margin? In particular, are the deterministic dose criteria for the LBEs in the infrequent and rare categories reasonable? If not, why not?
46.	Is it reasonable to use a 95% confidence value for the mechanistic source term for both the PRA sequences and the sequences designated as LBEs to provide margin for uncertainty? If not, why not? Is it reasonable to use a conservative approach for dispersion to calculate doses? If not, why not?

Table L-1 Questions in ANPR.

G(f)	PRA Technical Acceptability
47.	The approach proposed in the Framework does not predefine a set of LBEs to be addressed in the design. The LBEs are plant specific and identified and selected from the risk-significant events based on the plant-specific PRA. Because the plant design and operation may change over time, the risk-significant events may change over time. The licensee would be required to periodically reassess the risk of the plant and, as a result, the LBEs may change. This reassessment could be performed under a process similar to the process under 10 CFR 50.59. Is this approach reasonable? If not, why not?
48.	The Framework provides guidance for a technically acceptable full-scope PRA. Is the scope and level of detail reasonable? If not, why not? Should it be expanded and if so, in what way?
49.	Because a PRA (including the supporting analyses) will be used in the licensing process, should it be subject to a 10 CFR 50, Appendix B approach to quality assurance? If not, why not?
G(g)	Process to Develop Requirements
50.	Is this process clear, understandable, and adequate? If not, why not? What should be done differently?
51.	Is the use of logic diagrams to identify the topics that need to be addressed in the requirements reasonable? If not, what should be used?
52.	Is the list of topics identified for the requirements adequate? Is the list complete? If not, what should be changed (added, deleted, modified) and why?
53.	A completeness check was made on the topics for which requirements need to be developed for the new 10 CFR 53 (identified in Chapter 8) by comparing them to 10 CFR 50, NEI 02-02, and the International Atomic Energy Agency (IAEA) safety standards for design and operation. Are there other completeness checks that should be made? If so, what should they be?
54.	The results of the completeness check comparison are provided in Appendix G. The comparison identified a number of areas that are not addressed by the topics but that are covered in the IAEA standards. Should these areas be included in the Framework? If so, why should they be included? If not, why not?
H.	Defense-in-Depth
55.	Would development of a better description of defense-in-depth be of any benefit to current operating plants, near-term designs, or future designs? Why or why not? If so, please discuss any specific benefits.
56.	If the NRC undertakes developing a better description of defense-in-depth, would it be more effective and efficient to incorporate it into the Commission's Policy Statement on PRA or should it be provided in a separate policy statement? Why?
57.	RG 1.174 assumes that adequate defense-in-depth exists and provides guidance for ensuring it is not significantly degraded by a change to the licensing basis. Should RG 1.174 be revised to include a better description of defense-in-depth? Why or why not? If so, would a change to RG 1.174 be sufficient instead of a policy statement? Why or why not?
58.	How should defense-in-depth be addressed for new plants?
59.	Should development of a better description of defense-in-depth (whether as a new policy statement, a revision to the PRA policy statement, or as an update to RG 1.174) be completed on the same schedule as 10 CFR 53? Why or why not?

Table L-1 Questions in ANPR.

I.	Single Failure Criterion
60.	Are the proposed options reasonable? If not, why not?
61.	Are there other options for risk-informing the SFC? If so, please discuss these options.
62.	Which option, if any, should be considered?
63.	Should changes to the SFC in 10 CFR 50 be pursued separate from or as a part of the effort to create a new 10 CFR 53? Why or why not?
J.	Continue Individual Rulemakings to Risk-Inform 10 CFR 50
64.	Should the NRC continue with the ongoing current rulemaking efforts and not undertake any effort to risk-inform other regulations in 10 CFR 50, or should the NRC undertake new risk-informed rulemaking on a case-by-case priority basis? Why?
65.	If the NRC were to undertake new risk-informed rulemakings, which regulations would be the most beneficial to revise? What would be the anticipated safety benefits?
66.	In addition to revising specific regulations, are there any particular regulations that do not need to be revised, but whose associated regulatory guidance documents, could be revised to be more risk-informed and performance-based? What are the safety benefits associated with revising these guides? Which ones in particular are stakeholders interested in having revised and why?
67.	If additional regulations and/or associated regulatory guidance documents were to be revised, when should the NRC initiate these efforts, e.g., immediately or after having started implementation of current risk-informed 10 CFR 50 regulations?

L.2.12 Categorization of Stakeholder Comments

Tables L-2 and L-3 lists where each comment submitted by the various stakeholders in response to each question in the ANPR are addressed by the authors. The comments provided by the stakeholders are listed with the applicable ANPR question; however, where a stakeholder comment addresses multiple ANPR questions, that comment is listed for each applicable ANPR question. For example:

- NEI Comment #8 only addresses ANPR Question #8, under Topic B; therefore the comment is summarized under Topic B.
- NEI Comment #30 addresses ANPR Questions #30 and #44, under Topic F and Topic G, Sub-topic (e); therefore the comment is summarized under Topic F and Topic G, Sub-topic (e).
- NEI Comment #12, does not relate to Question #12 listed in the ANPR under Topic B, and actually addresses ANPR Question #35 under Topic G, Sub-topic (b); therefore the comment is summarized under Topic G, Sub-topic (b).

Table L-2 Categorization of stakeholders comments to ANPR. (Note 1)

ANPR Topics / Stakeholders		A. Plan to Risk Inform 10 CFR 50	B. Int. of Safety, Security and EP	C. Level of Safety	D. Integrated Risk	E. ACRS Views	F. Cont. Performance Standards	G.a Approach	G.b Emerg. Prep
IEEE 12/23/06		1-7	8-11	NC	NC	NC	NC	NC	12
NEQ 12/28/06		1-7	8-11	NC	NC	NC	NC	NC	12
GE 12/29/06 (Note 8)		L, refer to NEI	refer to NEI	refer to NEI	refer to NEI	refer to NEI	refer to NEI	refer to NEI	refer to NEI
WEST 12/22/06 (Note 7)		L	NC	NC	NC	NC	NC	NC	NC
PBMR 12/28/06		1,7 refer to NEI	refer to NEI	refer to NEI	refer to NEI	refer to NEI	refer to NEI	refer to NEI	36, refer to NEI
PBMR 9/11/06		pg 1-3 & 5th para on pg 4 of letter	pg 4 of letter, 2nd para	NC	NC	NC	pg 4 of letter, 3rd para	NC	pg 4 of letter, 4th para
ANS 10/05/06 (Note 6)	ANS 28	1,2,3,6,7	8, 9, 11	13-16, 18, 19	21, 22	24	25-28, 35	31	36
	ANS 22	1,2,5,6,7	8, 11	13	21-23	NC	25, 26	NC	NC
ANS 9/12/06 (Note 5)		L1-L5	NC	NC	NC	NC	NC	NC	NC
NEI 12/20/06		1-7	8-11	13-20	21-23	24	25-29	31-34	12, 35, 36
NEI 9/11/06		1-7	8-11	13-20	21-23	24	25-30	31-34	12, 35, 36
STARS 8/17/06 (Note 4)		GC, SC1, SC2, refer to NEI	refer to NEI	refer to NEI	refer to NEI	refer to NEI	refer to NEI	refer to NEI	refer to NEI
ASME 12/27/06 (Note 3)		L1-L4, 1-7	8-12	L3, 13-20	22	25-30	25-30	3, 5, 31-34	refer to NEI, 35, 36
ASME 8/25/06 (Note 2)		L1-L7, EA	EB	L3, EC	ED	EF	EF	EG	EG
AREVA 8/15/06		1-7	8-12, 35	13-20	21-23	25-30	25-30	31-34	35, 36

Table L-3 Continuation of categorization of stakeholder comments to ANPR. (Note 1)

ANPR Topics / Stakeholders		G.c Defense in Depth	G.d Protective Strategies	G.e Lic. Basis Events	G.f PRA	G.g Licensing Topics	H. Defense in Depth	I. Single Failure Criterion	J. Individual Rule-makings
IEEE 12/23/06		NC	NC	43	NC	53	NC	60-63	NC
NEQ 12/28/06		NC	NC	43	NC	53	NC	60-63	NC
GE 12/29/06 (Note 8)		refer to NEI	refer to NEI	refer to NEI	refer to NEI	refer to NEI	refer to NEI	refer to NEI	refer to NEI
WEST 12/22/06 (Note 7)		NC	NC	NC	NC	NC	NC	NC	NC
PBMR 12/28/06		refer to NEI	refer to NEI	43, refer to NEI	refer to NEI	refer to NEI	refer to NEI	refer to NEI	64, refer to NEI
PBMR 9/11/06		pg 4 of letter, 1st para	NC	NC	NC	pg 3 of letter, last para	NC	NC	NC
ANS 10/05/06 (Note 6)	ANS 28	NC	NC	42-44	48	54	55, 57-59	NC	1, 64, 66, 67
	ANS 22	NC	NC	46	NC	NC	NC	60	1, 64
ANS 9/12/06 (Note 5)		NC	NC	NC	NC	NC	NC	NC	L1
NEI 12/20/06		37-39	40, 41	30, 42-47	48, 49	50-54	55-59	60-63	64-67
NEI 9/11/06		37, 38, refer to NEI (12-20-06)	40, 41	42-44, refer to NEI (12-20-06)	49, refer to NEI (12-20-06)	refer to NEI (12-20-06)	55-57, refer to NEI (12-20-06)	60-62, refer to NEI (12-20-06)	64, refer to NEI (12-20-06)
STARS 8/17/06 (Note 4)		refer to NEI	refer to NEI	refer to NEI	refer to NEI	refer to NEI	refer to NEI	refer to NEI	SC2, refer to NEI
ASME 12/27/06 (Note 3)		L5, refer to NEI (37-39)	refer to NEI (40, 41)	refer to NEI (42, 46)	47-49	L6, 50-54	55-57, 59, refer to NEI (58)	60-63	64-67
ASME 8/25/06 (Note 2)		EG	EG	EG	EC	L8	EH	EI	EJ
AREVA 8/15/06		37-39	40, 41	42-46	47-49	50-54	55-59	60-63	64-67

NOTES:

(1) ASME PBMR, STARS, GE also endorsed the comments submitted by NEI.
(2) In the ASME 8/25/06 column:
- L1 through L8 refer to items 1 through 8 in the cover letter, E.A through E.J refer to items A through J in the enclosure to the letter
- Comments EE and EG indicated that detailed comments would be provided at a later date (see letter dated 12-27-06)

(3) In the ASME 12-27-06 column, L1 through L6 refer to items 1 through 6 in the cover letter.
(4) In the STARS 8-17-06 column, GC refers to the General Comment and SC1 and SC2 refer to the two specific comments.
(5) In the ANS 9-12-06 column, L1 through L5 refers to the 5 items in the submitted letter. ANS indicated that detailed comments would be provided at a later date (see letter dated 10-05-06).
(6) In the ANS letter of 10-05-06, comments were provided by two ANS subcommittees to the Nuclear Facilities Standards Committee, ANS 22 (Nuclear Power System Level Design) and ANS 28 (Gas Cooled Reactor Standards). These two subcommittees did not provide comments to every question, only those indicated in the table.
(7) In the Westinghouse 12-22-06 column, L refers to the submitted letter.
(8) In the GE 12-29-06 column, L refers to the submitted letter.

L.3 Comments from ACRS

The ACRS submitted a letter, dated September 26, 2007, to the Commission on the Framework. The ACRS letter provides five specific conclusions and recommendations (along with a detailed discussion providing their bases), and three sets of additional comments by Drs. Kress, Powers, and Wallis. The authors responses to the ACRS letter are provided below.

In reviewing the ACRS letter, the responses fall into one of the following categories:

- Comment(s) is an observation and there is no need to modify the NUREG.
- Comment(s) involves implementation of the NUREG. The authors do not necessarily disagree with the comment(s), however, any further study will be determined based on future use of the NUREG. These are noted in Appendix C for further study.
- Comment(s) is a suggested clarification which has been included in the NUREG, as appropriate.
- Comment(s) involves issues that the authors disagree with and no modification was made to the NUREG; the basis for the authors disagreement is provided.

L.3.1 ACRS Conclusions and Recommendations

The authors' responses to the five conclusions and recommendation provided by the ACRS are provided below.

ACRS Conclusion and Recommendation #1 –

- We concur with the staff that the safety objective of the Framework should be to ensure that advanced reactors, as a minimum, provide at least the same degree of protection of the public and the environment that is required for current-generation light water reactors (LWRs), and that advanced reactor designs comply with the Commission's safety goal quantitative health objectives (QHOs).

  Response:

  Comments are observations and there is no need to modify the NUREG.

ACRS Conclusion and Recommendation #2 –

- We concur with the staff that a set of licensing-basis events (LBEs) is needed as part of the licensing basis to structure the interactions between the staff and the applicant and to focus the conduct of mechanistic analyses. Identifying the LBEs by using the probabilistic risk assessment (PRA) reduces the risk that licensing-basis requirements will divert attention from events of real safety significance.

  Response:

  Comments are observations and there is no need to modify the NUREG.

ACRS Conclusion and Recommendation #3 –

- The use of a frequency-consequence (F-C) curve is an appropriate way to establish a range of regulatory requirements to limit radiation exposure to the public. However, a sequence-specific F-C curve, such as that developed in NUREG-1860, may not be a sufficient licensing criterion. A complementary cumulative distribution function (CCDF) F-C curve ("risk curve") that sums the contributions to risk from the entire spectrum of accident sequences establishes limits on risk better than the LBE F-C curve.

  Response:

  Comments involve further study to support implementation of the NUREG. The authors do not necessarily disagree with the comments, however, any further study will be determined based on future use of the NUREG. These are noted in Appendix C for further study. The authors believe that the sequence-specific F-C curve combined with the other requirements specified in the Framework, such as meeting the safety goals and satisfying the protective strategies does provide a sufficient licensing criterion. As described in Appendix C, adoption of a CCDF for evaluating PRA results would be helpful, if a sound basis can be developed for such a curve.

ACRS Conclusion and Recommendation #4 –

- We are concerned that extension of the F-C curves to very low dose levels may unduly increase requirements for the scope and level of detail in the PRA performed to demonstrate compliance with the F-C curve. It may also detract attention from accidents which could have a more significant impact on public health and safety.

  In the discussion, the ACRS further notes that it is premature to select the F-C curve presented and justified in the Framework.

  Response:

  Comment is a suggested clarification which has been included in the NUREG, as appropriate. The F-C curve was always intended to apply to accidents rather than routine operations, including minor spills. The text has been clarified. Future trials may lead to sharpened definition of scenarios to be included in the PRAs for new reactors.

ACRS Conclusion and Recommendation #5 –

- The Framework should recognize accident prevention as a fundamental regulatory goal and should specify a quantitative limit on the frequency of an accident. In technology neutral terms, an accident can be defined as the release of radionuclides within the plant significantly in excess of normal operating limits.

  Response:

  Comments involve further study to support implementation of the NUREG. The authors do not necessarily disagree with the comments, however, any further study will be determined based on future use of the NUREG. These are noted in Appendix C for further study. Appendix C indicates that measures focused on prevention analogous to CDF and LERF may be helpful and should be investigated further. The authors do not believe that meaningful measures for new reactors can be developed before Level 3 PRAs have been completed. Nonetheless, this issue has been identified as one needing further study.

L.3.2 ACRS Additional Comments

The three sets of additional comments provided by Drs. Kress, Powers, and Wallis were attached to the ACRS letter and which do not agree with the ACRS position provided in the letter. The authors' responses are provided below.

L.3.2.1 Additional Comments by Dr. Thomas S. Kress

- The Committee's report does not embrace long-standing ACRS position – the criteria for design safety of new reactor should be consistent with a CDF of 10^{-5} and LRF of 10^{-6}.

  Response:

  While those criteria have been a long-standing ACRS position for LWRs, the authors point out that they have not been long-standing ACRS positions for other technologies. Further, the authors believe that it is difficult to develop a technical basis for such values. Nonetheless, this item has been identified for further study in Appendix C.

- The Committee's report does not embrace long-standing ACRS position – design and siting should be separated as much as much as practical in the regulatory process. In accepting the Framework's LBE F-C figure-of-merit design curve, the Committee has compromised the principle of separation of design and siting. It creates an unnecessary burden on designers to use a surrogate site for PRA calculations, when equivalent curies released will better serve the purpose.

  Response:

  Separation of design and siting *as much as practical* in the regulatory process is not complete separation. The principle of separation of design and siting is not absolute and it is a flawed principle, if applied to new technologies, before measures akin to CDF and LERF can be demonstrated by Level 3 PRA to be useful surrogates for risk. Equivalent curies released are not necessarily directly proportional to risk.

- The report correctly considers that a frequency-consequence complementary cumulative distribution function (CCDF) design acceptance criterion would properly summate the risk. It should be a mandatory part of the Framework.

  Response:

  Dr. Kress' comments involve further study to support implementation of the NUREG. The authors do not necessarily disagree with the comments, however, any further study will be determined based on future use of the NUREG. This issue is noted in Appendix C for further study. The authors believe that the sequence-specific F-C curve combined with the other requirements specified in the Framework, such as meeting the safety goals and satisfying the protective strategies does provide a sufficient licensing criterion. As described in Appendix C, adoption of a CCDF for evaluating PRA results would be helpful, if a sound basis can be developed for such a curve.

- It is possible to construct a CCDF acceptance criterion that would make it consistent with any chosen values of CDF and LRF. The ACRS report should call for the inclusion of such a criterion for CDF of 10^{-5} and LRF of 10^{-6} for several reasons. [The six reasons provided by Dr. Kress are not repeated here.]

  <u>Response</u>:

  Dr. Kress' comments involve further study to support implementation of the NUREG. The authors do not necessarily disagree with the comments, however, any further study will be determined based on future use of the NUREG. These issues are noted in Appendix C for further study. As described in Appendix C, adoption of a CCDF for evaluating PRA results would be helpful, if a sound basis can be developed for such a curve.

- A CCDF limit curve for new reactor design would allow direct comparison of PRA results with the risk requirement.

  <u>Response</u>:

  Dr. Kress' comment involves further study to support implementation of the NUREG. While the authors agree with the comment, any further study will be determined based on future use of the NUREG. These are noted in Appendix C for further study. As described in Appendix C, adoption of a CCDF for evaluating PRA results would be helpful, if a sound basis can be developed for such a curve.

L.3.2.2 Additional Comments by Dr. Dana A. Powers

- A well crafted, technology-neutral regulatory Framework could [be helpful]. The overly complicated regulatory Framework developed by the staff is not a useful first step in the needed evolution of the current regulatory system to become technology neutral.

  <u>Response</u>:

  The authors disagree with this comment. It is to be expected that any feasible Framework developed to support a regulatory structure for licensing of future advanced reactors would be complex. The programmatic, policy and technical issues that need to be considered in developing a risk-informed and performance-based Framework applicable to diverse reactor designs are complex.

- The proposed Framework is not well founded. The staff did not take advantage of the current General Design Criteria (GDCs) and many of which are technology neutral. These criteria would have provided a sound foundation for a technology-neutral regulatory Framework.

  <u>Response</u>:

  The authors disagree with this comment. The GDCs were used in the Framework. However, there are difficulties in using the GDCs. The majority of the GDCs are not technology-neutral; only a small fraction of them (16 out of 55) are technology-neutral which result in a limited (and somewhat random) list of GDCs. As such, they do not provide as strong a foundation for a new process for regulation as the top-down system of criteria developed in the Framework. The foundation needs to support, as well as possible, a complete structure and address the basic safety fundamentals for providing for the public health and safety. A limited, random list of some technology-neutral GDCs does not provide this foundation. They do serve as a good check at the end to ensure the process used does encompass them, which was done in the Framework. The use of the GDCs is discussed in Chapter 8 and in detail in Appendix H.

- Staff has chosen to base it Framework on risk assessment. The proposed Framework demands PRAs well beyond the current state of the art. The staff has gone well beyond plausible future to expand the scope of PRAs mandated for regulation to extremes not even imaginable for today. Risk assessment will need to include events associated with drains in the plant chemistry laboratory to meet the staff expectations communicated through the F-C curves. This expansion will impose burdens on both licensees and regulatory heretofore never imagined. It will detract from a focus on safety issues that really do pose significant threats to the public health and safety.

  <u>Response</u>:

  The authors disagree with this comment. First, the Framework is not based on risk assessment, but integrates risk (probabilistic) and deterministic information throughout the Framework. Second, PRAs needed to support the process developed in the Framework are within the current state-of-the-art. Appendix F elaborates on the state of the art needed for the PRA supporting the Framework approach. ASME is currently developing PRA standard to support an approach akin to the Framework. The PRA demanded by the Framework will not require events, for example, associated with drains in the plant chemistry laboratory.

- Preservation of the design basis accident (DBA) concept under the guise of “licensing-basis events” (LBEs) is remarkable. Dr. Powers’ main points include:

  — The deficiencies of DBAs as a feature of the regulatory system have become apparent to us all since Three Mile Island (TMI).

  — These LBEs will be analyzed using very conservative methods.

  — Staff discounts the likelihood that LBEs will ossify into a legalistic Framework disconnected from physical reality.

  — The Framework is destined to descend into a few stylized accidents with the consequent neglect of more probable events that actually pose risks to the public.

  — Preservation of the DBA concept turns its back on the breadth of attention sought in the drive over the last few years to develop a risk-informed regulatory system.

  <u>Response</u>:

  The authors disagree with Dr. Powers comments and predictions. It is difficult to understand and respond to his concerns since no bases are provided, and they seem to be in contradiction with his earlier comments (e.g., “staff has chosen to base its framework on risk assessment” versus “turns its back on the breadth of attention sought . . . to develop a risk-informed regulatory structure”). Further, his comments appear to ignore the structure and protections defined in Chapter 6 of the framework. The concept of DBA (not the actual defined DBAs) was discussed at length with the ACRS and the stakeholder community. The authors agree that for both the designer and the regulator, the essence of the concept embodied in the DBA should be maintained. The authors further believe that in maintaining the concept, the LBEs should not be defined a priori, that is, there should not be a stylized set of DBAs, analyzed using very conservative methods, and disconnected from reality. In the Framework, the LBEs are risk derived, are both reactor and plant specific, and are continuously refined based on the current risk of the plant. Detailed discussions on this approach are provided in Chapter 6 of this NUREG. Further, Appendix E provides a detailed example of how the PRA is used to identify the LBEs for a specific plant. This example confirmed the feasibility of defining LBEs using a probabilistic approach.

- The authors mandate construction of risk assessments of unbelievable scope and depth but make no use of the results of the results beyond a rather effete comparison to “bottom line” risk results. Use of both risk reduction worth and risk achievement worth could be developed into a rational mechanism for introduction of defense in depth into safety regulation. Yet, importance metrics make no appearance in the proposed regulatory Framework.

  Response:

  The authors disagree. Extensive use of the PRA for defining the LBEs, for evaluating overall design performance, for examining safety significance and establishing when special treatment must apply relies on the use of importance measures, and is described in detail in the Framework (see Chapter 6). In addition, the test case provided in Appendix E shows how importance measures are used in the Framework. However, while importance measures are useful, there are difficulties associated with developing such measures for as yet undefined technologies, which is discussed in the Framework. These are issues deserving further study which are identified in Appendix C.

- Some suggestions that the Framework be tested on a new reactor technology such as a gas-cooled nuclear power plant. This is not a good idea. There is not a good phenomenological basis for assessing gas-cooled reactor safety. Even such a routine analysis as assessing the radionuclide release associated with expected depressurization events at gas-cooled reactors cannot be confidently done today as has been demonstrated in a Phenomena Identification and Ranking exercise recently undertaken by the NRC staff. This will assuredly handicap any application of a proposed regulatory Framework focused as this one is on F-C curves and bottom-line results.

  Response:

  The authors disagree. Meaningful results can be (and have been) derived from PRAs before all the technical issues are resolved. Therefore, there it would be incorrect to completely dismiss the possibility of meaningful results from PRA until all technical issues (e.g., fuel pellet performance) are resolved. PRA is the best tool for addressing risk and uncertainty.

L.3.2.3 Additional Comments by Dr. Graham B. Wallis

While the ACRS letter included additional comments from Dr. Wallis, these comments represent a small subset of the issues he raised in a much longer letter submitted to the NRC Executive Director of Operations (EDO).

- There are many features of some of the recommendations by [ACRS 2007] and by the staff for which the justification and implications have not been adequately evaluated. The Framework requires substantial revision to demonstrate that it responds to the needs of the Agency and that appropriate choices have been made.

  Response:

  Dr. Wallis states that his detailed reasons for this comment are provided in his separate letter that is addressed in Section L.4 below. The authors responses are provided there. However, the authors disagree with his conclusion. Dr. Wallis's conclusion appears to be the result of an expectation that the Framework is to represent a final product ready for use in regulation. However, the Framework is actually the first step in developing a risk-informed and performance-based regulatory licensing approach. The major objective of the Framework is to demonstrate the feasibility of such a concept; that is, the feasibility of developing a risk-informed and performance-based licensing approach. To demonstrate the feasibility, a process and criteria that can support the development of requirements was developed. However, as discussed in Chapter 9 of this NUREG (and as discussed in length in Appendix C), there are many other issues that need to be addressed and resolved before a risk-informed and performance-based licensing approach can be implemented.

- Dr. Wallis remaining comments were provided in the form of questions, and that in the revised document, they should be answered by providing convincing analysis and rationale.

  Response:

  The questions raised by Dr. Wallis are also covered in his detailed letter. Therefore, a response to each question is not provided below since they are covered in the responses to his letter (Section L.4). However, some general comments by the authors are provided below.

  — The authors agree that development of a new regulatory process for new reactors requires substantial work beyond the Framework.

  — While the authors agree that development of a new regulatory process for new reactors requires substantial work beyond the Framework, Dr. Wallis' comments imply that the current Framework should be revised to perform that task. Chapter 1 of the Framework lays out the complete process and defines the role of the current Framework in that process.

  — The F-C requirements for accidents are fully discussed in Chapter 6.

  — Defense-in-depth requirements and how they are implemented are discussed at length in Chapters 4, 5, and 6, and are introduced in Chapter 2. The structure of the Framework has evolved in response to a series of public meetings, reviews by and discussions with NRC staff and management, the ACRS, and public comment.

L.4 Letter from Dr. Graham Wallis

In a letter to Luis Reyes, NRC - EDO, dated July 24, 2007, Dr. Graham Wallis (ACRS member) provided comments on a draft of the Framework. Dr. Wallis states that his letter is based upon a thorough review of the entire Framework document and provides a comprehensive set of comments and his views on its clarity and content

In general, the majority of Dr. Wallis's comments appear to be the result of an expectation that the Framework is to represent a final product ready for use in regulation. However, the Framework is actually the first step in developing a risk-informed and performance-based regulatory licensing approach (as discussed in Chapter 1). The major objective of the Framework is to demonstrate the feasibility of such a concept; that is, the feasibility of developing a risk-informed and performance-based licensing approach. To demonstrate the feasibility, a process and criteria that can support the development of requirements was developed and is included in the Framework. However, as discussed in Chapter 9 of this NUREG (and as discussed in length in Appendix C), there are many other issues that need to be addressed and resolved before a risk-informed and performance-based licensing approach can be implemented.

Dr. Wallis provided three sets of comments: (1) an overview of his major conclusions, (2) detailed comments on the Framework, and (3) a description of the general features of ways to represent and manage risk. The following sections include a response to the three sets of comments.

Dr. Wallis's comments have been reviewed and have been grouped into one of four categories as follows:

- Comment(s) is an observation and there is no need to modify the NUREG. In some cases, the observation involved a suggestion for restructuring of the NUREG. The authors do not believe a restructuring of the NUREG is warranted. The structure of the document has evolved over time, and the authors believe the current structure, relative to the objective of the document and the comments from numerous other stakeholders, is appropriate.

- Comment(s) is a suggested clarification which has been included in the NUREG, as appropriate. Or comment(s) involves implementation of the NUREG. The authors do not necessarily disagree with the comment(s), however, any further study will be determined based on future use of the NUREG. These are noted in Appendix C for further study.

- Comment(s) involve issues that the authors disagree with and no modification was made to the NUREG; the basis for the authors disagreement is provided.

L.4.1 Overview Comments

At the beginning of his letter, Dr. Wallis provided 20 overview comments on the Framework. These are summarized below and a response to each of Dr. Wallis overview comments is provided.

- Comment # 1 -There is no development of a clear set of performance-based objectives, or top level design criteria.

  Response:

  There are top level design criteria in the Framework. They have been developed starting with objectives in Chapter 1 through a logical progression describing the philosophy of the approach taken leading to the top level criteria. They start as high as the Atomic Energy Act for protection of public health and safety, to safety, security and preparedness expectations, to defense-in-depth principles, to defined safety fundamentals, to probabilistic design criteria. They relate to defense-in-depth and the probabilistic licensing basis. They are incorporated into example requirements (to illustrate their implementation) as discussed in Chapter 8 and presented in length in Appendix J.

- Comment # 2 - The provision of containment appears to be a high level design criterion. This should be given prominence in the Framework, justified and criteria established for judging its adequacy.

  Response:

  Comment is a suggested restructuring of the NUREG. Further, the authors believe that this issues is given the necessary prominence in the Framework. For example, it is one of the major principles defined for defense-in-depth.

- Comment # 3 -There is no analysis and evaluation of various ways to describe and determine the impact on public health and safety.

  Response:

  The QHOs, which are specified as top level criteria in the Framework, define the Commission's expectations associated with public health and safety. In the Framework, they are proposed as criteria that future designs would have to meet. The dose limits in the F-C curve also establish per-event and per-year limits on dose, which are directly related to public health and safety, as described in Chapter 6.

- Comment # 4 - Important criteria (e.g., economy, effectiveness) are not articulated or evaluated.

  Response:

  Chapter 1 describes the overall objectives and sub-objectives of the Framework. How they are met is described in Chapter 9.

- Comment # 5 - There is no explanation of how the QHOs are met by the Framework.

  Response:

  The entire Framework has been developed consistent with the QHOs. This approach includes requiring a calculation of the QHOs themselves as well as meeting other criteria which, other than defense-in-depth, have been derived consistent with the QHOs.

- Comment # 6 - There is no comparison between the Framework and the existing system of regulation to show what improvements are being made.

  Response:

  Chapter 1 provides a general discussion on this point.

- Comment # 7 - No method is provided for adding up the risk represented by the individual PRA sequences.

  Response:

  In addition to the QHOs (which account for integrated risk), the usefulness of adding a CCDF curve (which is based on cumulative risk) has been identified as an issue to be addressed as part of implementation of the Framework (see Appendix C, Section C.4.1).

- Comment # 8 - The staff's F-C curve does not address cumulative risk.

  Response:

  See response to Comment # 7 above.

- Comment # 9 - The use and desirability of the F-C curve needs to be justified.

  Response:

  Chapter 6 describes, in detail, the basis for and the use of the F-C curve. Its use is intended to ensure that high frequency events have low consequences (i.e., dose), which has been NRC's policy and practice. It's consequence limits are derived from ICRP-64, as explained in Chapter 6, and the frequencies associated with the dose values are based upon previous NRC practice or judgements that take into consideration the QHOs. See Chapter 6 for the detailed discussion.

- Comment # 10 - The use of metrics resembling CDF and LERF is dismissed summarily, without explanation.

  Response:

  The authors recognize the value of subsidiary risk objectives in eliminating the need for a Level 3 PRA and establishing accident prevention and mitigation goals. This is discussed in Chapter 3. The development of such subsidiary risk objectives has been identified as an issue which should be resolved as part of implementation of the Framework (see Appendix C, Section C.4.9).

- Comment # 11 - Regulation of individual PRA sequences may not succeed because of the potential to manipulate the sequences to meet the criteria.

  Response:

  The guidance in Chapter 6 regarding how to select LBEs addresses the definition and selection of PRA sequences and is intended to prevent manipulation of the results.

- Comment # 12 - The F-C curve contains dose cumulative criteria (i.e., dose per year) and criteria for individual PRA sequences. This is not the usual F-C curve.

  Response:

  It is recognized in the Framework that the F-C curve contains some dose limits specified on a per-year basis and some on a per-event basis; the basis for this difference is provided in Chapter 6. However, the authors recommend this issue be explored further as part of implementation of the Framework (See Appendix C, Section C.4.2).

- Comment # 13 - Significant complicating features (e.g., complexity) of having the PRA output expressed as dose need to be explained and analyzed.

  Response:

  Historically, the consequences of a Level 3 PRA are plotted on a series of risk curves that show the results of calculations of early fatalities, early injuries, latent cancers, latent fatalities (long term cancer-related fatalities), and ground and water contamination. An intermediate result (seldom presented in more recent studies) is, of course[13], a detailed calculation of dose, something the comment implies is beyond the current state-of-the art. This practice has been standard for roughly 30 years. The comment correctly worries that this is a messy and difficult problem. However, this problem was solved years ago and the solution is imbedded in the PRA codes of today.

  Specifically, the results of a Level 2 PRA are a set of probability of frequency curves for the frequency of occurrence of a group of plant damage states, defined in a way that simplifies input to the consequence (Level 3) calculation. That is, each plant damage state represents a core damage scenario with specific release characteristics defined by the Level 2 model: release height and energy, isotopic quantity as a function of time, and other characteristics that affect how post release dispersion proceeds. These Level 2 results have been grouped into as few as three categories (LERF, releases affecting latent effects, and contained cases). These cases were defined after experience with full Level 3 PRAs showed which plant states were capable of causing early health effects for a wide range of plant types and local conditions. LERF results are plotted as a P(LERF) curve, which is not a CCDF and is not plotted in F-C space. It is a probability curve in F space.

[13]How else could the health effect consequences have been calculated, without a calculation of dose to the nearby, and possibly evacuating or sheltered population. One of the intermediate results is the series of dose isopleths that map the dose onto the local area. Real-time plume tracking codes project dose isopleths to allow emergency response managers to most effectively protect the community during an accident.

Misinterpretations of this capability and practice of PRA results in numerous comments and questions about the Framework. In common current practice, the terms CCDF and risk curve refer to the results of PRA presented in F-C space. The term "F-C curve" has entered common usage in recent years to describe the kind of individual sequence-by-sequence acceptance curve presented in Chapter 6 of the Framework as the criteria individual sequences must meet during the design process. This kind of limit makes it likely that the aggregated results of the PRA will meet CCDF expectations that ensure QHOs are met.

It has not been the purpose of the Framework to fully explain and define PRA, its calculation techniques, and its presentation formats. This information is contained in the current ASME and ANS PRA standards. A Level 2 and a Level 3 PRA standard is being developed by ANS/ASME, and the societies are developing a PRA standard for supporting a licensing approach akin to the one conceptualized in the Framework. The Framework recognizes the importance of the technical acceptability of the PRA which is discussed in detail in Chapter 7 and Appendix F.

- Comment # 14 - The need for LBEs, their definition, function and use should be better explained.

  Response:

  Comment is a suggested clarification which will be addressed.

- Comment # 15 - The proposed LBEs appear to be quite different from the traditional DBAs. No more detailed technical analysis of them is performed, unlike the practice in Chapter 15 of SARs.

  Response:

  In general, the traditionally defined DBAs, which were developed for LWRs, are difficult to apply to reactor designs that may bear little or no resemblance to LWRs. Second, such technical analyses cannot be carried out in a technology-neutral Framework as the details of accident sequences will differ greatly, depending on design.

- Comment # 16 - The traditional deterministic analysis of the type found in Chapter 15 of current SARs appears to have been abandoned completely.

  Response:

  Such analysis has not been abandoned. It is embedded in the LBEs.

- Comment # 17 - the criterion for ignoring all sequences with a probability less than 10^{-7}/yr appears to cut out major contributors that could cause the early fatality QHO to be violated.

  Response:

  The PRA does not ignore accident sequences with a frequency less than 10^{-7}/year. Accident sequences with a frequency below this value are included in the QHO calculations.

However, the 10^{-7}/year cut-off is for defining LBEs (i.e., only accident sequences with a frequency greater than 10^{-7}/year are used in LBE selection). This distinction is intended to provide a reasonable set of accident sequences for plant design and emergency planning, including realistic consideration of core damage scenarios in the design.

- Comment # 18 - It appears from Appendix F that there is an expectation that computer codes currently used to analyze DBAs are to be incorporated into the PRA. This is presently unfeasible.

  Response:

  There is no intent to incorporate detailed systems analysis codes into the PRA. They may, however, be used for some LBE analysis to confirm the acceptability of some of the PRA analysis.

- Comment # 19 - The approach of evaluating defense-in-depth by analyzing the uncertainties in the PRA is mentioned, but not developed as an element in the Framework.

  Response:

  Defense-in-depth is one of the major elements of the Framework. As an element, the intent is not to evaluate defense-in-depth, but to ensure that there is adequate defense-in-dpeth to address uncertainties. Development of what is meant by defense-in-depth, and how it is a part of the regulatory structure, is an element of the Framework (See Chapter 4). For example, the defense-in-depth principles are written as guidance for the NRC staff to use in developing requirements and acceptance criteria. As such, the principle that addresses uncertainty is intended to ensure that the requirements, when developed, require applicants and licensees to quantify uncertainties in their analysis and that in setting regulatory acceptance criteria, the NRC staff include margin to account for uncertainties.

- Comment # 20 - Safety margin is not defined in a useful form for performance-based regulation.

  Response:

  The definition of safety margin in Chapter 4 is useful in defining the terminology and concept used in the Framework. Use of this concept would be done primarily by the NRC staff as part of developing guidance for the requirements (e.g., setting regulatory limits on specific parameters, ROP performance indicators).

L.4.2 Detailed Comments

Following the overview comments, Dr. Wallis provided a number of detailed comments essentially organized to parallel the draft Framework. These are summarized below, by section, as presented in Dr. Wallis's letter. A response is provided for each.

- Objectives - The Framework objectives should be a clear statement of what the authors intend to achieve and it should be described how they are met. Suggestions for additional objectives were provided.

Response:

The comments suggests clarification and expansion of the Framework. The authors have made changes to the NUREG as a result of the comments.

- Relationship to Current Licensing Process - There is no direct link between DBAs and the PRA. LBEs are defined but they are not DBAs and fulfill none of the DBA functions.

  Response:

  The LBEs are intended to replace and fulfill the role of DBAs. They are derived from a design-specific PRA, as described in Chapter 6, and are different for each design, unlike the currently defined DBAs.

- Structuring a Framework to Meet the Objectives - The Framework should be justified in a top down fashion with alternatives defined, evaluated and the optimum way of proceeding described.

  Response:

  Alternative approaches/criteria to what is in the document have been examined and have been discussed in numerous earlier drafts, public meetings and workshops. Stakeholder input on these alternatives has helped the authors to converge on the present structure and approach.

- Risk - The Framework should include quantitative measures to address core damage. The QHOs do not address this. The PRA could be used to calculate other measures of risk (e.g., total deaths). Alternatives to the use of dose as the consequence measure are not provided.

  Response:

  The development of quantitative measures to address core damage is identified as an open item for resolution during implementation of the Framework (see Appendix C, Section C.4.9). An explanation of why dose was chosen as the consequence measure has been added to Chapter 6. Other measures of risk (e.g., total deaths) are not calculated because there are no licensing criteria to compare them to. These additional measures are addressed in site specific environmental impact statements (required by 10 CFR 51), which each applicant to build an NPP must submit as part of the application.

- Framework Overview - The safety, security and preparedness expectations are discussed, but only the safety expectations define what is expected (i.e., the QHO). There is no discussion of subsidiary objectives. The protective strategies are not design objectives, but a means to an end.

  Response:

  Clarifications have been added to the Framework to better describe the role of the protective strategies and the safety, security and preparedness expectations. The comment on subsidiary objectives was discussed above,

- Safety, Security and Preparedness Expectations - There are insufficient linkages between the agency objectives and the Framework structure.

  Response:

  Additional explanation has been included in the Framework.

- Defense-in-Depth and Safety Fundamentals - The chapters on defense-in-depth and safety fundamentals provide little substance by way of risk-informed or performance-based criteria. They are too qualitative for inclusion in the Framework.

  Response:

  The defense-in-depth principles and safety fundamentals in the Framework represent guidance for the NRC staff to use in developing the detailed licensing requirements. They are not intended to be used by the designer. As such, they are intended to lead to specific requirements whose purpose is to provide defense-in-depth in reactor designs and to ensure that designs are developed with safety as an integral part of the design process.

- Design Criteria and Guidelines - An F-C curve that uses cumulative risk values should be used. The purpose and selection process for the LBEs is not clear. Suggested areas for clarification are identified. The dose values and frequencies used to construct the F-C curve and their bases are not clear. It appears that the QHOs were not used to create any part of the F-C curve, although it is stated that PRA sequences meeting the F-C curve will meet the QHOs. Using the latent fatality QHO ($2x10^{-6}$/rem), and the individual risk co-efficient ($5x10^{-4}$/rem) a value of 4 rem/yr should be the dose limit. This even exceeds the ALARA dose of 5 rem/yr.

  Response:

  The F-C curve was constructed using existing dose criteria wherever possible. The dose criteria not in existing regulations were derived from ICRP-64 as described in Chapter 6. The frequency values assigned to the dose criteria are also based upon guidance in ICRP-64. The doses specified in the lower frequency range of the F-C curve are based upon the early fatality QHO, such that individual event scenarios would meet the QHO. The cumulative effect of all event scenarios in this range must also meet the QHO.

  With respect to the F-C curve being compatible with a 4 mrem/year dose in the frequent event range, which Dr. Wallis derives from the latent fatality (LF) QHO, the derivation of the 4 mrem/year dose is not correct and cannot be used as a dose limit. To be correct, the calculation of the QHOs must take into account several key factors, including the range of accidents associated with the specific design, their source term, the site specific meteorology and emergency evacuation. The LF QHO calculation must be done in a probabilistic fashion considering the population dose out to10 miles and the design and site specific factors (e.g., source term) mentioned above. It is not correct to relate the 4 mrem/year, as derived in Dr. Wallis's letter, to the latent fatality QHO since the derivation in Dr. Wallis's letter does not take into account any of the factors listed above. In addition, as stated in the Commission's Safety Goal Policy, the calculations relating to the QHOs are for accidents only, not normal operation, which is the bulk of the frequent event category shown on the F-C curve and the area suggested by Dr. Wallis for application of the

4mrem/yr dose value. Therefore, any dose value derived from the QHOs should not be applied in the frequent event category and, therefore, the 4 mrem/yr dose value has no valid basis, meaning or regulatory use.

The comments related to an F-C curve that is based upon cumulative risk and the purpose of the LBEs, were addressed earlier in this section. Dr. Wallis's other comments on the clarity of the Framework in describing the LBE selection process, have been used to make changes to the NUREG in this area.

- PRA Technical Acceptability - Comments related to the clarity of Chapter 7 of this NUREG.

  Response:

  The comments have been considered and clarifications added to the NUREG.

- Requirements Development Process - There is no explanation of what role LBEs play in the licensing basis. The roles played by significant features in the Framework need to be made more specific.

  Response:

  Table 8-4 in Chapter 8 of this NUREG (which identifies the topics for which requirements are needed) provides a cross reference to each significant feature (Section) in the Framework so as to ensure that the requirements incorporate the significant features. The role of LBEs was discussed earlier in this section.

- Appendix A - The material in this appendix does not lead to identification of any top level regulatory criteria.

  Response:

  The material in this appendix is not intended to identify any top level regulatory material, only to illustrate the range of issues that a technology-neutral set of requirements need to address.

- Appendix B - No assessment is made of how the items in this appendix influence the design of the Framework.

  Response:

  The Framework has been developed as an alternative to 10 CFR 50, and thus must interface with other parts of 10 CFR as does 10 CFR 50. This appendix identifies those interfaces and is essential to ensuring that when the Framework is implemented, the requirements contain the proper interfaces.

- Appendix C - The environmental assessment does not address all environmental issues. It only addresses individual risk. A 4 mrem/yr dose (derived by Dr. Wallis from the latent fatality QHO) was suggested as the proper measure for an environmental assessment.

Response:

See response above to Dr. Wallis's comment on Design Criteria and Guidelines.

- Appendix D - It needs to be clarified what is an acceptable way to calculate early fatalities to ensure a satisfactory basis for LERF.

  Response:

  Appendix D is intended to explain how the current LWR, CDF and LERF values are acceptable surrogates for the QHOs. The method for the derivation of CDF and LERF (including the method and use of early fatality calculations) have been reviewed and accepted, including by the ACRS.

- Appendix E - It appears that if the method for selecting LBEs, that was tested in Appendix E is used, Chapter 15 analysis would not be required. How then would the regulatory functions performed by Chapter 15 be accomplished?

  Response:

  The LBEs are intended to play the same role as DBAs, but will be design specific consistent with the design-specific PRA.

- Appendix F - How will it be possible to incorporate the level of technical analysis represented by accident analysis computer codes into PRA analysis?

  Response:

  This comment was addressed earlier in this section.

- Appendix G - Provides additional discussion on PRA, F-C curves, DBAs, deterministic and probabilistic analysis and defense-in-depth.

  Response:

  The comments in the appendix represent the personal preferences of Dr. Wallis. No changes to the NUREG were made.

L.5 Comments from Electric Power Research Institute

In a December 2006 report (# 1013582), EPRI identified and assessed specific elements of the draft NRC Framework and provided recommendations on where additional development and testing of the Framework would be useful. The EPRI work was part of a larger effort to assist the nuclear industry in responding to NRC's request for comments on its draft Framework. The EPRI report provides useful comments and insights into Framework issues and development needs.

It should be noted that the EPRI report agrees with the Framework approach and criteria in many more areas than where there is disagreement (e.g., use of the QHOs as the level of safety to be achieved). Summarized below are the key areas where EPRI suggests changes to the Framework, with which the authors agree.

- A CCDF curve is needed for assessing risk. Dose should be considered as the consequence measure.

  Response:

  The use of a CCDF curve is an item to be addressed in implementation of the Framework (as discussed in Appendix C), including appropriate consequence measures.

- The Framework should allow for the use of other risk evaluation methods in lieu of a full PRA.

  Response:

  This issue has been added to Appendix C as an item to be addressed during implementation.

- Conservative analysis should be allowed (in lieu of mean values) in calculating the QHOs and F-C curve compliance, provided it does not change the conclusion.

  Response:

  This clarification has been made in the Framework.

- Alternate design and defense-in-depth principles were suggested.

  Response:

  The suggested alternatives should be reviewed as part of implementing the Framework.

There are a few areas where the authors disagree with the EPRI report which are summarized below, along with the authors' basis for the disagreement.

- Defense-in-Depth should include requirements that:

  (a) inherent / passive design features should be used to provide a balance between prevention and mitigation

  (b) the integrity of each barrier should be sufficient to meet the QHOs and F-C curve

  Response:

  To require a type of design feature be incorporated into the design is viewed as being too prescriptive and not necessarily needed to address uncertainty (which is the main purpose of DID in the Framework). Also it is not clear what SSCs the inherent / passive requirement would apply to.

  With respect to requiring each barrier to meet the QHOs / F-C curve, this is viewed as not practical (since an initiating event could be a barrier breach and it should be the design, assessed in an integrated fashion, that must meet the requirements).

- The F-C curve should not be used as a hard limit when assessing individual accident sequences (to allow more design flexibility), but rather the CCDF curve should be the limit.

  Response:

  The use of the F-C curve ensures that more frequent event scenarios only result in small releases of radioactive material. A hard limit is considered appropriate for such a fundamental safety practice. The use of a CCDF curve cannot ensure such a practice is maintained.

- The safety classification scheme is not fully developed.

  Response:

  The EPRI report does not elaborate or specify on what is meant by “not fully developed.” That is, what is missing and needs to be considered is not discussed in the EPRI report. Consequently, it is not possible to respond to this comment. However, the authors believe the safety classification scheme is sufficiently developed to demonstrate feasibility, and also recognize that there are issues to be developed if it is to be implemented (see Appendix C).

L.6 References

[ACRS 2007] Letter dated September 26, 2007 from W.J. Shack, ACRS, to D.E. Klein, Chairman USNRC, Subject: Development of a Technology-Neutral Regulatory Framework.

[ANS 2006a] Letter dated September 12, 2006 from C. Mazzola, ANS NFSC, to Secretary USNRC, Subject: Response by ANS to ANPR on Proposed 10 CFR Part 53: Approaches to Risk-Informed and Performance-Based Requirements for Nuclear Power Reactors.

[ANS 2006b] Letter dated October 5, 2006 from C. Mazzola, ANS NFSC, to Secretary USNRC, Subject: ANS Responses to Published Questions on the ANPR on Proposed 10 CFR Part 53: Approaches to Risk-Informed and Performance-Based Requirements for Nuclear Power Reactors.

[Areva 2006] Letter dated August 17, 2006 from R.L. Gardner, Areva NP, to Secretary USNRC, Subject: Comments on 10 CFR 50 and 53 - Approaches to Risk-Informed and Performance-Based Requirements for Nuclear Power Reactors.

[ASME 2006a] Letter dated August 25, 2006 from K.R. Balkey, ASME, to Secretary USNRC, Subject: Comments on ANPR to Make 10 CFR 50 Requirements Risk-Informed and Performance-Based.

[ASME 2006b] Letter dated December 27, 2006 from K.R. Balkey, ASME, to Secretary USNRC, Subject: Comments on ANPR to Make 10 CFR 50 Requirements Risk-Informed and Performance-Based.

[EPRI 2006] Electric Power Research Institute, Report #1013582, “Technical Elements of Risk-Informed, Technology-Neutral Licensing Framework for Nuclear Power Plants,” December 2006.

[GE 2006] Letter dated December 29, 2006 from R.E. Brown, GE Nuclear Energy, to Secretary USNRC, Subject: Comments on NRC ANPR, “Approaches to Risk-

Informed and Performance-Based Requirements for Nuclear Power Reactors," 71 Fed. Reg. 26.267.

[IEEE 2006] Letter dated December 23, 2006 from J. J. Disosway, IEEE NPEC, to Rules and Directive Branch USNRC, Subject: IEEE Responses to Published Questions on the ANPR on Proposed 10 CFR Part 53: Approaches to Risk-Informed and Performance-Based Requirements for Nuclear Power Reactors.

[NEI 2006a] Letter dated September 11, 2006 from A. P. Heymer NEI to Secretary USNRC, Subject: Comments on ANPR for 10 CFR Parts 50 and 53 - Approaches to Risk-Informed and Performance-Based Requirements for Nuclear Power Reactors," (71 Fed. Reg. 26267).

[NEI 2006b] Letter dated December 28, 2006 from A. P. Heymer NEI to Secretary USNRC, Subject: Comments on ANPR for 10 CFR Parts 50 and 53 - Approaches to Risk-Informed and Performance-Based Requirements for Nuclear Power Reactors," (71 Fed. Reg. 26267).

[NEQ 2006] Letter dated December 28, 2006 from J. F. Gleason to USNRC, Subject: James F. Gleason Responses to Published Questions on the ANPR on Proposed 10 CFR Parts 53: Approaches to Risk-Informed and Performance-Based Requirements for Nuclear Power Reactors.

[NRC 2006] U.S. Nuclear Regulatory Commission (NRC). Federal Register Notice, Advanced Notice of Proposed Rulemaking, "Approaches to Risk-Informed and Performance-Based Requirements for Nuclear Power Reactors." Volume 71, Number 86, Page 26267. May 4, 2006.

[PBMR 2006a] Letter dated September 11, 2006 from E. G. Wallace, PBMR, to Secretary USNRC, Subject: Initial Comments on ANPR on Approaches to Risk-Informed and Performance-Based Requirements for Nuclear Power Reactors, (71 Fed. Reg. 26267).

[PBMR 2006b] Letter dated December 28, 2006 from E. G. Wallace, PBMR, to Secretary USNRC, Subject: Comments on ANPR on Approaches to Risk-Informed and Performance-Based Requirements for Nuclear Power Reactors, (71 Fed. Reg. 26267).

[STARS 2006] Letter dated August 17, 2006 from D. R. Woodlan, STARS, to Secretary USNRC, Subject: STARS Comments on ANPR Approaches to Risk-Informed and Performance-Based Requirements for Nuclear Power Reactors.

[Wallis 2007] Letter dated July 24, 2007, from G.B. Wallis, ACRS, to L.A. Reyes, Executive Director for Operations, USNRC, Subject: Comments on Draft NUREG-1860, "Framework for Development of a Risk-Informed, Performance-Based Alternative to 10 CFR Part 50."

[West 2006] Letter dated December 22, 2006 from J. A. Gresham, Westinghouse, to USNRC, Subject: RIN 3150-AH81, ANPR "Approaches to Risk-Informed and Performance-Based Requirements for Nuclear Power Reactors" Westinghouse Comments on Draft Rule Change.