FRAMEWORK FOR DEVELOPMENT OF A RISK-INFORMED, PERFORMANCE-BASED, TECHNOLOGY-NEUTRAL ALTERNATIVE TO 10 CFR PART 50

APPENDICES

Working Draft Report
(Does not represent a staff position)

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research

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FOREWORD

The purpose of this draft NUREG is to discuss an approach, scope, and acceptance criteria that could be used to develop risk-informed, performance-based requirements for future plant licensing. The Nuclear Regulatory Commission (NRC) is making the latest working draft framework available to stakeholders. This working draft is to inform stakeholders of the NRC staff's consideration of possible changes to its regulations, and to solicit comments on the staff's direction as described in an advance notice of proposed rulemaking published in the Federal Register in April 2006.

This version of the framework is a working draft. It does not represent a staff position and is subject to changes and revisions. The framework is expected to be updated in June 2006 as a final draft. The NRC will post the final draft of the framework on Ruleforum website when it is complete.

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A. SAFETY CHARACTERISTICS OF THE NEW 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 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 framework include advanced LWR and CANDU reactors, modular HTGRs[1], 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.

Physical Protection

[to be developed]

[1] 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.

Stable Operation

To ensure stable operation, a thorough examination of potential initiating events is conducted as part of the risk analysis of the design. For the future reactor technologies, initiating event considerations may be substantially different from those for current LWRs. Examples are events associated with on-line refueling, recriticality due to more highly enriched fuels, performance of fuels with higher burnup, chemical interactions with some reactor coolants or structures, and events that impact two or more reactors on the same site.

Protective Systems

Plant features are provided to mitigate the consequences of initiating events. A critical part of the determination of these features is a qualitative review of the reactor-specific design philosophy, which includes a review of the design and performance features of the barriers, the reactor-specific safety functions that protect these barriers, and the specific inherent and engineered safety features of the reactor concept in light of their capability to protect the barriers. For the future reactor technologies, some mitigative considerations will be substantially different from those for current LWRs. Examples are performance and monitoring of passive safety systems including passive decay heat removal, the performance and testing as well as the PRA modeling of digital systems, qualification and testing of new materials including fuel, non-traditional emergency core heat removal systems, a greater reliance on automated operation and limited operator intervention, and, for liquid metal reactors, potential energetic interactions of the working fluid when exposed to the environment.

Barrier Integrity

Functional barriers to radionuclide release are provided to maintain isolation of hazardous nuclear material within the system. Barriers can be both physical barriers and barriers to mobilization and transport of radioactive material, e.g., the physical-chemical form that retards the spread and dispersion of the material. All current nuclear power plants include a fuel barrier, a reactor coolant boundary, and a reactor building or containment barrier, as well as additional barriers for sources of radioactive material outside the reactor core. The design features, capabilities, and selection of materials for these barriers and the way they are deployed to effect a defense-in-depth design philosophy varies among the new reactor concepts. For example, the fuel for HTGRs consisting of ceramic coated fuel particles embedded in graphite fuel compacts or pebbles have fundamentally different properties than the zircalloy-clad UO_2 fuel elements used in LWRs. The roles of the fuel elements, pressure boundary, and containment in preventing and mitigating radioactive releases are also fundamentally different across existing and new reactor technologies.

Accident Management

Accident management includes management of all accident scenarios, whether release has occurred or not. Therefore, plant abnormal and emergency procedures are part of accident management, as well as severe accident management guidelines and on-site and off-site emergency plans. If functional barriers fail to adequately limit the radionuclide release, accident management is provided to control the accident progression and ultimately to limit the public health effects of accidents. Because there are differences in the challenges to stable operation, protective systems, and barriers among the different reactor technologies, the accident management strategies will also be different.

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 six new reactor designs to illustrate the variation found in such characteristics. The six designs are: the pebble bed modular reactor (PBMR), the Advanced CANDU Reactor (ACR) 700, and four Generation IV reactors. The Four Gen IV deigns are: Very-High-Temperature Reactor (VHTR), Supercritical Water-Cooled Reactor (SCWR), Gas-Cooled Fast Reactor (GFR), and Lead-Cooled Fast Reactor (LFR). The information on these reactor designs is taken from Ref. A.1.

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 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 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 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:

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

2. 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;

3. 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;

4. 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;

5. 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;

6. 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;

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

8. 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 sub critical. 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.3.5 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.

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.

B. RELATIONSHIP TO 10 CFR

B.1 Introduction

This Appendix contains (a) the relationship of the requirements in 10 CFR Part 50 to requirements in other parts of 10 CFR shown in Table B.1, and (b) the relationship of the requirements of other parts of 10 CFR to the requirements of 10 CFR 50 shown in Table B.2. 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 Requirements in Other Parts of 10 CFR

The data in Table B.1 show 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 the description refers to. 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
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

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
50.30 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
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 (I) 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

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
50.54(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
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 part s55.31and 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

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
50.78 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
50.12	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.2 Relationship of Requirements in Other Parts of 10 CFR to Requirements in 10 CFR 50

(To be written)

C. Protection of the Environment

Protection of the environment during normal operation is required by 10 CFR Part 50.34a, which sets forth design objectives for equipment to control releases of radioactive material in effluents to the environment and by 10 CFR Part 50.36a, which provides technical specifications for effluents during operation. 10 CFR Part 50.34a specifies that the design objectives for keeping releases contained in effluents during normal operation and expected operational occurrences should be ALARA (as low as reasonably achievable considering technology, cost-benefit to society and other related socio-economic considerations). 10 CFR Part 50.36a provides technical specifications for releases of liquid and gaseous effluents to unrestricted areas, that, in addition to meeting the requirements of Part 20, should be as low as reasonably achievable. Numerical guidance on design objectives and limiting conditions of operation for releases to meet the ALARA criterion is provided in Part 50, Appendix I. This guidance states:

(1) "The calculated annual total quantity of all radioactive material above background to be released from each light-water-cooled nuclear power reactor to unrestricted areas will not result in an estimated annual dose or dose commitment from liquid effluents for any individual in an unrestricted area from all pathways of exposure in excess of 3 millirems to the total body or 10 millirems to any organ."

(2) "The calculated annual total quantity of all radioactive material above background to be released from each light-water-cooled-nuclear power reactor to the atmosphere will not result in an estimated annual air dose from gaseous effluents at any location near ground level which could be occupied by individuals in unrestricted areas in excess of 10 millirads for gamma radiation or 20 millirads for beta radiation."

(3) "The Commission may specify, as guidance on design objectives, a lower quantity of radioactive material above background to be released to the atmosphere if it appears that the use of the design objectives in paragraph (2) is likely to result in an estimated annual external dose from gaseous effluents to any individual in an unrestricted area in excess of 5 millirems to the total body; and

(4) "Design objectives based upon a higher quantity of radioactive material above background to be released to the atmosphere than the quantity specified in paragraph (2) will be deemed to meet the requirements for keeping levels of radioactive material in gaseous effluents as low as is reasonably achievable if the applicant provides reasonable assurance that the proposed higher quantity will not result in an estimated annual external dose from gaseous effluents to any individual in unrestricted areas in excess of 5 millirems to the total body or 15 millirems to the skin."

(5) "The calculated annual total quantity of all radioactive iodine and radioactive material in particulate form above background to be released from each light-water-cooled nuclear power reactor in effluents to the atmosphere will not result in an estimated annual dose or dose commitment from such radioactive iodine and radioactive material in particulate form for any individual in an unrestricted area from all pathways of exposure in excess of 15 millirems to any organ."

Protection of the environment is also provided by 10 CFR Part 51 which contains the environmental protection regulations applicable to NRC's domestic licensing and related regulatory functions. Part 50 implements the relevant portions of the provisions of the National Environmental Policy Act (NEPA) of 1969, as amended, in a manner consistent with the NRC's domestic licensing and related regulatory authority under the Atomic Energy Act of 1954, as amended. Section 51.20

specifies the criteria for and identification of licensing and regulatory actions requiring environmental impact statements (EIS); for example, a permit to construct or operate a nuclear power reactor, and Section 51.29 provides the scope of the EIS. Section 51.45 specifies the requirements of the environmental report. Sections 51.50, 51.51, and 51.52 specify the data required to comply with requirements to obtain a construction permit, and Section 51.53 provides requirements for the post construction environmental reports, including reports on the operating license stage, the license renewal stage, and post operating license (i.e., decommissioning) stage.

Currently, there are no requirements for protection of the environment from accidents at NPPs. It has been generally accepted that the current low risk to members of the public also provides for low risk to the environment. Many new plant designs will have long response times under accident conditions, allowing licensees to meet the Commission's safety goals by greater reliance on evacuation of the public, a situation where the public can be protected, even though the land may be contaminated, could be the result.

In consideration of the above, the need for a separate goal related to protection of the environment was evaluated. This evaluation consisted of assessing how well the frequency-consequence curve (discussed in Chapter 6) and the Commission's Safety Goal Quantitative Health Objectives (QHOs) provide protection for the environment. The adequacy of the environmental protection provided by the frequency-consequence curve (Figure 6-1) and the QHOs was assessed using the criteria for an extraordinary nuclear occurrence (ENO) contained in 10 CFR Part 140. The ENO criteria represent levels of individual dose and land contamination or offsite cleanup costs resulting from an accident below which there should be minimal societal impact, since the cost of any remedy would be borne by the licensee. Accordingly, both the ENO dose, land contamination criteria and cleanup cost criteria were used in this assessment as discussed below. In all cases, the objective is to show that the environment is being protected to the same degree as the public and that, accordingly, the societal risk from land contamination is very small.

Dose/Land Contamination Assessment

This assessment is based upon showing that the frequency-consequence curve discussed in Chapter 6 is sufficient to ensure that the risk to the environment is approximately equal to that expressed by the Commission safety goal QHOs for risk to the public. Using Equation 1, the individual risk to a member of the public can be estimated using the frequency-consequence curve.

$$R_1 = D*F*C \qquad \text{Equation 1}$$

where:

D = Equivalent dose in rem
F = Frequency (per year)
C = Risk Coefficient (likelihood of fatal cancer/rem

Section 140.84 of 10 CFR Part 140, Equivalent Criterion 1, provides two criteria for determining whether there has been a substantial discharge of radioactive material or substantial radiation levels offsite to cause contamination.

The first criterion is stated in terms of actual or projected doses to one or more persons offsite as a result of the release. A whole body dose of 20 rem, a bone marrow dose of 20 rem, a thyroid dose of 30 rem, a skin dose of 60 rem, or another organ dose of 30 rem provide the basis for making the determination there has been contamination offsite to be categorized as an ENO.

The second criterion is stated in terms of surface contamination levels of at least a total of 100 square meters of any offsite property. These levels are presented in two ways: the first is for property that is contiguous to the licensee's site and is owned or leased by a person with whom an indemnity agreement has been executed and the second is for any offsite property. The second set of levels are as follows:

Contamination Source	Contamination Level
Alpha emission from transuranic:	0.35 microcuries per square meter
Alpha emission from non-transuranic:	3.5 microcuries per square meter
Beta/gamma emissions:	4 millirads per hour

These levels will result in an equivalent dose of approximately 20 rem.

To anchor a frequency to these contamination levels, consider that the projected dose and the surface contamination levels of Criterion I in Section 140.84 are essentially equivalent, i.e., contamination levels of 0.35 microcuries per square meter of alpha emitting non-transuranic of and beta gamma emitters of 4 millirads per hour, are both equivalent to a dose level of 20 rem per year.

Using the frequency vs. consequence curve (Figure 6-1) levels of contamination shown above, it can be seen that a dose level of 20 rem is associated with a frequency of approximately 10^{-5}/yr. Accordingly, the levels of contamination stated above in 10 CFR §140.84 are approximately related to this frequency.

The standard latent fatality risk coefficient for members of the public is $5x10^{-4}$/rem, where an individual exposed to 1 rem has a $5*10^{-4}$ likelihood of contracting a fatal cancer over their lifetime.

This results in an individual latent fatality risk to a member of the public of (10^{-5}/yr) (20rem) ($5x10^{-4}$/rem) = 10^{-7} per year which is much less than the latent fatality QHO individual risk of $2x10^{-6}$/yr. Thus, it can be concluded that a plant meeting the frequency-consequence curve shown in Chapter 6 would provide a level of protection to the environment approximately equivalent to that provided to the public.

This same analysis approach and conclusion can also be applied to the dose that corresponds to an abnormal occurrence as defined in NUREG-0090 (i.e., 25 rem). These limits are used to define the desired outcome of the Commission's strategic goal for safety in the FY2004-FY2009 Strategic Plan as it pertains to releases of radioactive materials that cause significant adverse environmental impacts.

<u>**Cleanup Cost Assessment**</u>

This assessment is based upon showing that the criteria in Chapter 6 provide protection of the environment equivalent to protection of the public on a value-impact basis using the ENO criteria related to cleanup costs as the figure of merit. The assessment is summarized below.

First, a release large enough to result in substantial offsite contamination must occur. Events that could cause such a release would have to involve significant core damage and release to the environment. Since10^{-5}/yr is the dividing line between infrequent and rare events, where infrequent events must maintain coolable geometry, events of this type would have a frequency of less than 10^{-5}/yr. In addition, not all core damage events lead to a significant release to the environment;

therefore, a value of 10^{-6}/yr for a large release was chosen as a reasonable frequency estimate, based upon PRA results for advanced LWRs and the industry's goal to have future plant designs incorporate enhanced safety characteristics (e.g., EPRI-ALWR Utility Requirements Document).. Second, it is assumed that the ENO criteria represent the measure of environmental protection desired and, therefore, a goal of future designs could be to ensure that offsite cleanup costs do not exceed the criteria in 10 CFR Section 140.85:

- $2,500,000 to an individual or
- $5,000,000 cumulative
-

Using a frequency of 10^{-6}/ry, the cleanup cost criteria equate to annualized values of:

- $2.50/ry (individual)
- $5.00/ry (cumulative)

These values corresponds to a range of 1-10 dollars/reactor year.

Using the frequencies for early and latent fatalities associated with the reactor safety goal QHOs:

early fatality frequency = $5*10^{-7}$/ry
latent fatality frequency = $2*10^{-6}$/ry

And the values of a life assumed in regulatory analysis (NUREG/CR-6212):

value for early fatality = $$2.1*10^6$ per life saved
value for latent fatality = $2000/person-rem

Early and latent fatality risk, based on dollars, can be estimated:

fatality = (cost per life saved)*(fatality frequency) Equation 2

early fatality = $(2.1*10^6$ dollars$)\ (5*10^{-7}$/ry)
= 1 dollar/ry

latent fatality = [(2000 dollars/person-rem)/($5*10^{-4}$/person-rem)]*($2*10^{-6}$/ry)
= 8 dollars/ry

These comparisons, using dollars, also show a 1-10 dollars/reactor year range of value-impact for the public. Thus, an approach has been taken to show that by meeting the Safety Goal QHO, protection is provided to the environment at least equivalent to that provided to the public. Therefore, no separate goals on environmental protection are proposed.

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[2] in the vicinity of a nuclear power plant of prompt fatalities[3] 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."

- "The risk to the population in the area of nuclear power plant of cancer fatalities[4] 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 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.

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[2]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).

[3]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 in. [Ref. D.1] An early fatality is defined as one that results in death within 1 year of exposure.

[4]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.)

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.

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 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 implies:

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

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"

$TP(1) \quad = \quad$ 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-1150Ref. A.3 for the Surry PRA (Table 4.3-1)Ref. A.4

- 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.

REFERENCES

[D.1] A discussion of the dose conversion factor databases embedded in MACCS and their use for various types and purposes of calculations performed in the code is contained in the MACCS2 code manual [Chanin and Young, “Code Manual for MACCS2:User’s Guide, NUREG/CR-6613, Vol. 1: SAND97-0594, Sandia National Laboratories, May 1998.]

E. EXAMPLE OF LBE SELECTION PROCESS

This appendix will provide an example of the probabilistic selection process for licensing basis events (LBE) as described in Chapter 6. The term LBEs is used in the framework to indicate those accidents that must be considered in the safety analysis of the plant and must meet some deterministic criteria in addition to meeting the frequency-consequence curve.

In the technology-neutral approach used in the framework, the probabilistic LBEs are selected from PRA sequences. The LBEs not only include sequences that involve a radionuclide release and lead to a dose at the site boundary, but may also include sequences that do not involve any release of radionuclides.

In addition to the probabilistically selected LBEs there is at least one deterministic LBE that must be considered for defense in depth purposes, as discussed in Subsection 6.2.2.2. An example of the selection of this deterministic event will not be included in this appendix.

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 requirements necessary to ensure the quality 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 requirements for a high quality PRA that will be utilized fully in the licensing process. The required scope of the PRA and the corresponding requirements for each technical element are addressed. Specifically, high-level requirements 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 requirements, some unique aspects of new reactors that will impact the PRA are identified.

This appendix builds on existing PRA quality requirements delineated in Regulatory Guide 1.200 and the currently available PRA standards. The requirements focus on a PRA of the reactor core that includes both internal and external events during all modes of operation. In addition to addressing the risk resulting from operation of the reactor, PRA techniques can be used to support the licensing effort by evaluating the risk from accidents involving other radioactive materials (e.g., spent fuel and radioactive waste). Thus, the identified high level requirements are such that they include accident analysis of all sources of radioactive material. Requirements for the use of PRA models to support the assessment of security are also provided. Specifically, the PRA models can be used to identify the minimum set of equipment (referred to as target sets) that must be disabled by an adversary in order to cause a release of radioactive material.

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 sources of radioactive material included in the licensing of the reactor and being addressed in the risk-informed licensing framework;
- 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 requirements for defining POSs for future reactor designs are shown in Table F-1.

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

Item	Requirement
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.

Although PRAs are focused on accidents involving the reactor core, other sources of radioactive materials are addressed in the licensing of a reactor. These sources include the spent fuel pool and waste facilities. In the proposed Technology-Neutral Framework, accidents involving these sources can also be modeled in a PRA and the results used in identical fashion as those obtained for the reactor core analysis. In addition, hazardous chemicals can present a hazard to the plant workers, particularly the reactor operators. Consideration of accidents involving hazardous chemicals is typically considered in the design of the control room HVAC. Table F-1 identifies the high-level requirement that the PRA must include a step to identify all radioactive and hazardous material sources in the plant that pose a risk to the public or 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 requirements presented in this section address all potential initiators during all modes of operation.

Finally, the risk metrics used to help make risk-informed licensing decisions will affect the scope of the PRA. Since the technology neutral framework is using a frequency-consequence curve to identify licensing basis events and in classifying 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 requirements 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 requirements 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 requirements. 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.

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 second level, Release Analysis, consists of an analysis of the physical processes of the accident, the corresponding response of confinement barriers (including a containment if it is part of the new reactor design), 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.

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, address the reactor during all operating modes, and can include other sources of radioactive material besides the reactor core. The requirements for the Level 1 portion of a full scope PRA are discussed in this section. Although the requirements focus on the PRA models for the reactor core, risk models for other radioactive material sources are addressed.

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 requirements which are discussed separately in Sections F.3.2 and F.3.3, respectively.

Initiating event analysis identifies and characterizes the internal events that can upset plant stability and challenge critical safety functions during all plant operating states (i.e., full-power, shutdown, and transitional states). Initiating events must be considered that can affect any source of radioactive material on site in any chemical and physical form. A systematic method for identifying potential initiators must be utilized. Events that have a frequency of occurrence greater than 1E-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, as frequent, infrequent, or rare. 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 requirements for the initiating event analysis are shown in Table F-3.

Table F-3 Initiating event requirements.

Item	Requirement
IE-1	Use a systematic process to identify a complete set of plant-specific internal 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 should be fully defensible and biased towards success such that issues of manufacturer or construction variability, code limitations, and other uncertainties are unlikely to shift a success path to a failure path. 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 uncertainity 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 all sources of radiation (e.g., spent-fuel pool), not just the core. This could include systems required for spent fuel pool cooling as well as for core and containment cooling, inventory makeup, and reactivity control. The high level requirements for the success criteria analysis are shown in Table F-4.

Table F-4 Success criteria requirements.

Item	Requirement

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.

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.

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 internal accident initiators involving the reactor core and other radioactive sources onsite. The high level requirements for the accident sequence analysis are shown in Table F-5.

Table F-5 Accident sequence requirements.

Item	Requirement
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.

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 requirement 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).

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 requirements for the systems analysis are shown in Table F-6.

Table F-6 Systems analysis requirements.

Item	Requirement
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 requirements 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 also 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 modeled.

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. Requirements 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 advance 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 requirements for the human reliability analysis are shown in Table F-7.

Table F-7 Human reliability analysis requirements.

Item	Requirement
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.

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.

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 requirements for parameter estimation are shown in Table F-8.

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 requirements for accident sequence quantification are shown in Table F-9.

Table F-8 Parameter estimation requirements.

Item	Requirement
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.
PE-5	Provide both mean values and a statistical representation of the uncertainty for the parameters.

Table F-9 Accident sequence quantification requirements.

Item	Requirement
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.
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.

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.

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 form the data, the most significant uncertainty is epistemic (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 assessment is different from the analysis of other internal events. It includes 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 requirements for an internal flood PRA must address all of these mediums and include internal floods initiated during all modes of plant operation.

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 requirements for flood source identification are shown in Table F-10.

Table F-10 Flood source identification requirements.

Item	Requirement
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 requirements for flood scenario evaluation are shown in Table F-11.

Table F-11 Flood scenario evaluation requirements.

Item	Requirement
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 (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).

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 requirements for flood scenario evaluation are shown in Table F-12.

Table F-12 Flood sequence quantification requirements.

Item	Requirement
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 requirements 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 requirements 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.

F.3.3 Internal Fire PRA

An internal fire assessment is different from the analysis of other internal events. It includes 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.

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.

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 requirements for faire area screening are shown in Table F-13.

Table F-13 Fire area screening requirements.

Item	Requirement
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 requirements for a fire initiation analysis are shown in Table F-14.

Table F-14 Fire initiation analysis requirements.

Item	Requirement
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 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 requirements for a fire damage analysis are shown in Table F-15.

Table F-15 Fire damage analysis requirements.

Item	Requirement
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 and geometry).
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	Include plant-specific experience and reflect scenario-specific conditions in the analysis of fire suppression. 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 requirements for a fire plant response analysis are shown in Table F-16.

Table F-16 Plant response analysis requirements.

Item	Requirement
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 for the fire area in the response analysis.
PR-4	Identify potential circuit interactions which can interfere with safe shutdown.
PR-5	Human reliability analysis addresses the effect of fire scenario-specific conditions on the operator performance.
PR-6	Estimate the required end-state frequency for each fire-induced scenario
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.

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 requirements for a seismic hazard analysis are shown in Table F-17.

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 requirements for a seismic fragility analysis are shown in Table F-18.

Table F-17 Seismic hazard analysis requirements.

Item	Requirement
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 study is used, confirm that the data and information is still valid.
SH-2	Determine the level of hazard analysis based on the intended application and on site-specific complexity.
SH-3	The hazard analysis uses pertinent site information (e.g., geological, seismological, and geophysical data; site topography) and historical information.
SH-4	The hazard analysis considers all sources of potentially damaging earthquakes that can affect the seismic hazard at the site.
SH-5	The hazard analysis accounts for all credible mechanisms influencing vibratory ground motion that can occur at the site.
SH-6	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.

Table F-18 Seismic fragility analysis requirements.

Item	Requirement
SF-1	Develop realistic fragility estimates for all SSCs identified in the seismic systems analysis.
SF-2	Criteria for screening of high seismic capacity SSCs, if performed, is provided.
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 (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 int 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 requirements for a seismic systems analysis are shown in Table F-19.

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 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.

Screening and bounding analysis identifies external events other than earthquake 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.

Table F-19 Seismic systems analysis and quantification requirements.

Item	Requirement
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	Ensure the system model reflects as-built (or as-designed), as operated plant.
SS-5	Integrate the seismic hazard frequencies and the seismic fragilities into the plant system model.
SS-6	Quantify the seismic scenarios to obtain the desired risk metrics in accordance with the requirements 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-7	Modify human recovery failure events to account for seismic-related impacts and include any seismic-specific recovery action.
SS-8	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-9	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.

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 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 requirements for performing an external event screening and bounding analysis are shown in Table F-20.

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

Item	Requirement
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 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 requirements for an external event hazard analysis are shown in Table F-21.

Table F-21 External event hazard analysis requirements.

Item	Requirement
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 requirements for an external event fragility analysis are shown in Table F-22.

Table F-22 External event fragility analysis requirements.

Item	Requirement
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 requirements for an external event systems analysis are shown in Table F-23.

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

Item	Requirement
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	Ensure the system model reflects as-built (or as-designed), as operated plant.
SQ-5	Integrate the external event hazard frequencies and the SSC fragilities into the plant system model.
SQ-6	Quantify the external event scenarios to obtain the desired risk metrics in accordance with the requirements 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-7	Modify human recovery failure events to account for external event-related impacts and include any recovery actions specific to the external event.
SQ-8	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-9	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.3.6 Vital Area and Target Set Analysis

To be developed

F.4 Release Analysis Technical Elements

The requirements 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 containment if one is included in the new reactor design), 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 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 confinement building or containment depending on what is included in the design) 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 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 affects 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. 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 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 confinement/containment are evaluated against the capacity of the confinement/containment to withstand the identified challenges. A probabilistic framework is used to combine the two pieces to determine the probability of confinement/containment 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 confinement/containment response. The high level requirements for an accident progression analysis are shown in Table F-24.

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 containment 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 requirements for a source term analysis are shown in Table F-25.

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 containment. The computer codes should be verified to cover the range of conditions included in the calculations. Accident sequence 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.

Table F-24 Accident progression analysis requirements.

Item	Requirement
AP-1	For each accident sequence, identify important attributes that can influence the accident progression, confinement/containment response, and subsequent radionuclide release. Include the impact of accident initiators on confinement/containment systems 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 the vessel and confinement/containment 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 and confinement/containment capacity to withstand the challenges introduced by accident phenomena. This requires identification of the vessel and confinement/containment failure criteria.
AP-5	Use a probabilistic framework to assess vessel and confinement/containment 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 confinement/containment 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.

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.

Table F-25 Source term analysis requirements.

Item	Requirement
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.
ST-3	Include accident sequence specific characteristics in the calculations that affect the timing, form and magnitude of radioactive material released from the fuel and coolant.
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.

F.5 Consequence Assessment Technical Elements

The requirements 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 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 impacts on human health, the environment, and economic impacts. The consequence measures of most interest focus on impacts on human health. Specific measure of accident consequences developed in a PRA can include: number of early fatalities, number of early injuries, number of latent cancer fatalities, population dose to 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 Technology-Neutral 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 requirements for a consequence analysis are shown in Table F-26.

Table F-26 Consequence analysis requirements.

Item	Requirement
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, 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 requirements for an external event systems analysis are shown in Table F-27.

Table F-27 Health and economic risk estimation requirements.

Item	Requirement
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.
HE-3	Link portions of the accident analysis.

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 containment 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.

G. COMPLETENESS CHECK AND PRELIMINARY ASSESSMENT OF THE APPLICABILITY OF 10 CFR 50 REQUIREMENTS TO THE TECHNOLOGY-NEUTRAL FRAMEWORK

G.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 technology-neutral and risk-informed 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 technology-neutral 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 3 to each protective strategy. The end result of applying this process is summarized in Table 8-6, which lists the technical topics which the technology-neutral requirements must address.

A similar process was followed for the administrative requirements, as described in Section 8.3 of the framework; 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 shown in Table 8-9.

To help ensure that the list of technical and administrative topics shown in Tables 8-6 and 8-9 is complete, a check was made against other documents containing requirements for reactor safety. Specifically, the following documents were compared against Tables 8-6 and 8-9:

- 10 CFR 50: “Domestic Licensing of Production and Utilization Facilities”
- IAEA Safety Standards Series NS-R-1: “Safety of Nuclear Power Plants: Design”
- IAEA Safety Standards Series NS-R-2: “Safety of Nuclear Power Plants: Operation”
- NEI 02-02: “A Risk-Informed, Performance-Based Regulatory Framework for Power Reactors”

In the comparison against 10 CFR 50, a preliminary assessment for each topic was also made regarding the potential for using current 10 CFR 50 wording for that topic. This assessment was made following the guidelines in Section 8.5 of the framework. This Appendix documents the results of the completeness check and the preliminary assessment of the applicability of 10 CFR 50 requirements to the technology-neutral requirements.

The results of the comparisons are shown in Tables G-1 through G-4. A summary of each comparison is provided below.

G.2 Comparison Against 10 CFR 50

Table G-1 shows the results of the comparison against 10 CFR 50. Table G-1 addresses all requirements in 10 CFR 50 and the results of the preliminary assessment of the applicability of 10 CFR 50 wording to the technology-neutral requirements. Table G-1 (and Table G-2) are organized by major categories to make comparisons among the framework, 10 CFR 50 and NS-R-1 easier. No technical topics were found in 10 CFR 50 that were not included in Table 8-6. For the technical topics, there are several areas where it appears 10 CFR 50 wording is technology-neutral (or can be made technology-neutral) and should be considered for inclusion in the technology-neutral requirements. These are identified in Table G-1. Examples include:

- many of the General Design Criteria found in 10 CFR 50, Appendix A
- 10 CFR 50, Appendix B - Quality Assurance
- 10 CFR 50.65 - Maintenance Rule
- portions of 10 CFR 50.44 - Combustible Gas Control

For the administrative topics, Table 8-9 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 other administrative items (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 are shown in Table 1 and include:

- financial items
- process items
- employee protection items
- legal items

These items need to be included in the technology-neutral requirements.

G.3 Comparison Against IAEA NS-R-1

Table G-2 shows the results of the comparison against 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.

In reviewing Table G-2 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 topics not found in Chapter 8. These are:

- management and organization
- safety culture
- minimizing radioactive waste generation
- ensuring failure of non-safety SSCs will not fail safety SSCs
- passive safety or continuously operating safety systems
- automatic safety actions in initial stage of accidents
- single failure criterion (framework uses probabilistic approach)
- escape routes
- consider decommissioning as part of design
- design fuel assemblies to permit inspection
- coverings and coatings integrity
- design should address transport and packaging of radioactive waste
- design for on-line maintenance

Accordingly, these need to be assessed as to whether or not they should be incorporated into the framework.

G.4 Comparison Against IAEA NS-R-2

Table G-3 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. Most of the topics included in NS-R-2 are also included in Chapter 8 of the framework. In reviewing Table G-3 it can be seen that the framework does not include the following items:

- organizational responsibilities and functions
- qualification of personnel
- commissioning program
- core management and fuel handling
- spare parts procurement, storage and dissemination
- preparation for decommissioning

Similar to the NS-R-1 comparison, these items need to be assessed as to whether or not they should be incorporated into the framework.

G.5 Comparison Against NEI 02-02

NEI 02-02 was written to suggest a risk-informed, performance-based alternative to 10 CFR 50, which NEI called Part 53. NEI 02-02 proposes a structure and content for Part 53. Table G-4 shows a comparison of the Part 53 content against the framework. All items listed in the NEI proposed Part 53 are included in the framework except an item on selective implementation which NEI propose. The framework does not currently address selective implementation. NEI 02-02 also suggests where 10 CFR 50 wording should be retained and where it should be revised. A table is included in NEI-02-02 cross referencing it to 10 CFR 50. The NEI comparison to 10 CFR 50 is very similar to the comparison included in Table G-1, thus providing additional confirmation that there are no significant omissions.

Table G-1 10 CFR 50 comparison and applicability - initial assessment.

US 10 CFR Part 50		Technology Neutral Framework
1. Objectives, Purposes, and Bases		
50.1	Basis, Purpose, and Procedures Legal Authority Applicability and Regulating Authority	• Use 10 CFR 50 words
50.2	Definitions	• Review for applicability
50.3	Interpretation Assigns legal interpretation authority to NRC General Counsel	• Use 10 CFR 50 words
2. Oversight/Enforcement		
50.7	Employment Protection Protects employees of licensees against discrimination and retribution for providing information to NRC, Congress, etc.	• Use 10 CFR 50 words
50.8	Information Collection Requirements Requires NRC to submit information collection requirements to OMB for approval to collect the information	• Use 10 CFR 50 words
50.9	Completeness Requirements	• Use 10 CFR 50 words
50.10	License Requirements (Construction and Operation) Establishes license requirement Identifies facilities which are required to obtain an NRC license and which are not	• Use 10 CFR 50 words
50.11	Exceptions and Exemptions from License Requirements	• Use 10 CFR 50 words
50.12	Specific Exemptions	• Consider risk-informing 10 CFR 50 words
50.35	Issuance of Construction Permits	• Use 10 CFR 50 words
50.39	Public Inspection of License Requirement	• Use 10 CFR 50 words
50.50	Issuance of Licenses and Construction Permits Technical Specifications, Conditions, and Limitations	• Consider use of 10 CFR 50 words
50.51	Continuation of License Set time limits on term of license Holds licensee responsible for site after permanent shutdown	• Use 10 CFR 50 words
50.53	Jurisdictional Limits	• Use 10 CFR 50 words
50.58	Publishing and Hearing Requirements to Issue Construction Permits	• Use 10 CFR 50 words
50.76	Licensee Change of Status, Financial Qualifications Requires licensee to inform NRC 75 days before ceasing to exist	• Use 10 CFR 50 words
50.78	Installation information and verification Requires licensees to submit to IAEA inspection when directed by NRC	• Use 10 CFR 50 words

Table G-1 10 CFR 50 comparison and applicability - initial assessment.

US 10 CFR Part 50		Technology Neutral Framework
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 words
50.90	Application for Amendment of License or Construction Permit	• Consider risk-informing 10 CFR 50 words
50.91	Notice of Public Comment and State Consultation concerning License Changes 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 words
50.92	Issuance of Amendments Identifies issues which are to be considered when evaluating a request for a license change	• Consider risk-informing 10 CFR 50 words
50.100	Revocation, Suspension, and Modification of Licenses and Construction Permits	• Use 10 CFR 50 words
50.101	Retaking Possession of Special Nuclear Fuel The NRC may retake fuel upon revocation of license.	• Use 10 CFR 50 words
50.102	Commission Orders for Operation After Revocation Allows Commission to require a plant to be operated after licenses have been revoked	• Use 10 CFR 50 words
50.103	Suspension and Operation in War or National Emergency	• Use 10 CFR 50 words
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 words
50.111	Criminal Penalties	• Use 10 CFR 50 words
3. Management Requirements/Confidence		
50.30	Filing Procedure, Oath or Affirmation	• Use 10 CFR 50 words
50.33a	Anti Trust Limitation	• Use 10 CFR 50 words
50.40	Common Standards Compliance requirement Requirement for licensee to be technically and financially qualified Operation does not infringe on defense or public health	• Use 10 CFR 50 words
50.81	Creditor Regulations Sets conditions under which a creditor may posses a lien on a utilization and production facility	• Use 10 CFR 50 words

Table G-1 10 CFR 50 comparison and applicability - initial assessment.

US 10 CFR Part 50		Technology Neutral Framework
Appendix C:	A Guide for the Financial Data and Related Information Required to Establish Financial Qualifications for Facility Construction Permits	• Use 10 CFR 50 words
Appendix L:	Information Requested by the Attorney General for Antitrust Review of Facility Construction Permits and Initial Operating Licenses	• Use 10 CFR 50 words
4. Tracking and Records Schema/Requirements		
50.4	Written Communications Communication Delivery Requirements and Procedures Distribution Requirements Communication Requirements Required Submissions	• Use 10 CFR 50 words, if sufficiently technology-neutral
50.20	Two Classes of Licenses	• Not applicable to technology-neutral framework
50.21	Class 104 License Medical facility and device manufacturer licenses	• Not applicable
50.22	Class 103 License Commercial and industrial license	• Use 10 CFR 50 words, if sufficiently technology-neutral
50.23	Construction Permits	• Use 10 CFR 50 words, if sufficiently technology-neutral
50.31	Allowance for Combining Applications	• Use 10 CFR 50 words, if sufficiently technology-neutral
50.32	Elimination of Repetition	• Use 10 CFR 50 words, if sufficiently technology-neutral
50.33	Contents of Application (General Requirements)	• Needs revision to account for technology-neutral and risk-informed
50.41	Additional Standards for Class 104 License	• Not applicable to technology-neutral framework
50.42	Additional Standards for Class 103 License Usefulness Requirement Antitrust Restriction Open Communication Requirement	• Use 10 CFR 50 words, if sufficiently technology-neutral
50.43	Additional Standards for Class 103 License NRC is required to inform the following of applications for licenses: 1. State and Local Authorities 2. Public via Federal Register 3. Other Cognizant Federal Agencies	• Use 10 CFR 50 words, if sufficiently technology-neutral
50.70	Inspections Requires licensees to submit to routine inspection Requires licensee to provide reasonable space accommodation to inspectors	• Use 10 CFR words

Table G-1 10 CFR 50 comparison and applicability - initial assessment.

US 10 CFR Part 50	Technology Neutral Framework
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	• Modify to be consistent with technology-neutral and risk-informed nature of framework
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	• Consider modification to be technology-neutral and risk-informed
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	• Modify to be technology-neutral and risk-informed
50.75 Reporting and Record Keeping for Decommissioning Planning Establishes reasonable assurance that funds will be available for decommissioning process	• Use 10 CFR 50 words
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 words
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 technology-neutral requirements
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 technology-neutral requirements
Appendix Q: Pre-Application Early Review of Site Suitability Issues	• Use 10 CFR 50 words, if sufficiently technology-neutral
5. Safety Objectives	
Appendix A: General Design Criteria for Nuclear Power Plants	• See Appendix to Table G-1
6. Owner/Management Competency and Fitness Requirements	

Table G-1 10 CFR 50 comparison and applicability - initial assessment.

US 10 CFR Part 50		Technology Neutral Framework
50.55	Conditions of Construction Permits Construction time requirements 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 CFR50 words, if sufficiently technology-neutral
7. Confidence in Personnel		
50.5	Deliberate Misconduct	• Use 10 CFR 50 words
50.74	Notification of Change in Operator or Senior Operator Status Reporting Requirement	• Use 10 CFR 50 words, if sufficiently technology-neutral
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	• Consider use of 10 CFR 50 words, if sufficiently technology-neutral
8. Confidence in Engineering		
50.34	Contents of Application (Technical Requirements)	• Need to modify to be technology-neutral and risk-informed
50.36	Technical Specifications	• Need to modify to be technology-neutral and risk-informed
50.45	Standards for Construction Permits	• Consider use of 10 CFR 50 words, if sufficiently technology-neutral

Table G-1 10 CFR 50 comparison and applicability - initial assessment.

US 10 CFR Part 50		Technology Neutral Framework
50.54	Conditions of Licenses Fuel Reprocessing Quality assurance Safety Analysis Report Quality Assurance Requirement Safety Analysis Report Quality Assurance Change Allowances 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 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 Plan Requirement Bankruptcy Notification Requirements National Security Technical Spec Allowance Earthquake Damage Identification and Elimination Requirement	• Use 10 CFR 50 words, if sufficiently technology-neutral and risk-informed. • Drop non-power reactor requirement.
50.55a	Codes and Standards Sets minimum standards commensurate with safety Identifies ASME Standards as minimums Sets Minimum Requirements for Specific Structural Materials	• Needs modification to be technology-neutral and risk-informed
50.65	Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants Requirements to Set Maintenance Effectiveness Goals Commensurate with Industry Goals Sets Monitoring Requirements and Frequency Requirements Requires Risked-Informed Management of Maintenance	• Consider use of 10 CFR 50 words, if sufficiently technology-neutral and risk-informed
50.69	Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants Defines Safety Classes Defines Applicability and Scope of Risk-Informed Treatment of SSCs Sets Evaluation Level of Risk-Informed Analysis	• Needs modification to be technology-neutral
50.109	Backfitting Definition of Backfitting Conditions to Require Backfitting	• Consider use of 10 CFR 50 words
Appendix B:	Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants	• Consider use of 10 CFR 50 words

Table G-1 10 CFR 50 comparison and applicability - initial assessment.

US 10 CFR Part 50	Technology Neutral Framework
Appendix O: Standardization of Design; Staff Review of Standard Designs	• Not needed in technology-neutral requirements
9. Contingency Planning	
50.47 Emergency Plans Requires NRC to consult FEMA findings when approving emergency plans Responsibility Assignments State and Local Authorities On Shift Personnel Responsibility Near Site Emergency Authorities Information Dissemination Requirements Assay and Monitoring Requirements Public Exposure Assessment Requirement Exposure Protection for Emergency Workers Requirement Drill Requirements Plan Review Requirements Failure to Comply Sanctions Participation Requirements Public Area Exposure Analysis Requirements Less then 5% Fuel Loading Exception	• Modify to be technology-neutral and risk-informed
50.48 Fire Protection General Description Specific Hazard Detection and Suppression Systems Administrative Controls Risk-informed Analysis Requirement	• Modify to be technology-neutral and risk-informed
50.49 Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants	• Needs to be risk-informed and technology-neutral
50.59 Changes, Tests, and Experiments Definitions of Changes, Tests, and Experiments Definition of Scope Reporting Requirements of Changes, Tests, and Experiments	• Needs to be risk-informed and technology-neutral
Appendix E: Emergency Planning and Preparedness for Production and Utilization Facilities	• Needs to be risk-informed and technology-neutral
Appendix F: Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities	• Not applicable to technology-neutral framework
10. Engineering Prescriptives	
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 (consider use of 10 CFR 50.44(a) and (d) words)
50.46 Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Reactors	• Not applicable - LWR specific

Table G-1 10 CFR 50 comparison and applicability - initial assessment.

US 10 CFR Part 50		Technology Neutral Framework
50.46a	Acceptance Criteria for Reactor Coolant System Venting System	• Make technology-neutral and risk-informed
50.60	Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation	• Make technology-neutral
50.61	Fracture toughness requirements for protection against pressurized thermal shock events	• Make technology-neutral
50.62	Requirements for reduction of risk from ATWS events for light water cooled nuclear power plants	• Not applicable - LWR specific
50.63	L:oss of all alternating current power	• Not applicable - LWR specific
50.66	Requirements for Thermal Annealing of the Reactor Pressure Vessel	• Not applicable - LWR specific
50.68	Criticality Accident Requirements Limits Concentrations of Storage Fuel Rods Limits Credit Taken for Moderation Limits Fuel Rod U-235 Purity	• Make technology-neutral and risk-informed
Appendix G:	Fracture Toughness Requirements	• Make technology-neutral
Appendix H:	Reactor Vessel Material Surveillance Program Requirements	• Make technology-neutral
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 R:	Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979	• Not applicable - LWR specific
Appendix S:	Earthquake Engineering Criteria for Nuclear Power Plants	• Use 10 CFR 50 words, if sufficiently technology-neutral
11. Security of Material and Facilities Requirements		
50.13	Requirement for Security Requires licensees to maintain security against foreign enemies and domestic criminals	• Expand 10 CFR 50 words to include vulnerability assessment
50.37	Agreement Limiting Access to Classified Information	• Use 10 CFR 50 words
50.38	Foreign Corporation or Individual Restriction	• Use 10 CFR 50 words
50.64	Limitation on the use of Highly Enriched Uranium (HEU) in Domestic Non-power Reactors	• Not applicable
12. Containment and Exposure Requirements		
50.34a	Design Objective Requirements for Equipment to Control the Release of Radioactive Active Material	• Use 10 CFR 50 words, if sufficiently technology-neutral
50.36a	Technical Specifications on Effluent from Nuclear Power Plants	• Use 10 CFR 50 words, if sufficiently technology-neutral

Table G-1 10 CFR 50 comparison and applicability - initial assessment.

US 10 CFR Part 50	Technology Neutral Framework
50.36b Environmental Conditions	• Use 10 CFR 50 words, if sufficiently technology-neutral
50.67 Accident Source Term Defines applicability and requirements Sets radiation exposure limits within defined areas around the plant	• Revise to be consistent with framework guidance on source term and radiation exposure limits
13. Regulation Burden Mitigation	
50.52 Combining Licenses	• Use 10 CFR 50 words
50.56 License Conversion	• Use 10 CFR 50 words
50.57 Issuance of Operating License Requirements to issue an operating license	• Use 10 CFR 50 words
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 words
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 technology-neutral

Appendix to Table G-1

10 CFR 50, Appendix A - General Design Criteria (GDC)

A.) Those that are currently technology-neutral; but may need to be risk-informed

GDC 2: Design Bases for Protection Against Natural Phenomena
GDC 3: Fire Protection
GDC 5: Sharing of Structures, Systems, and Components
GDC 10: Reactor Design
GDC 11: Reactor Inherent Protection
GDC 12: Suppression of Reactor Power Oscillations
GDC 18: Inspection and Testing of Electric Power Systems
GDC 20: Protection System Functions
GDC 21: Protection System Reliability and Testability
GDC 22: Protection System Independence
GDC 23: Protection System Failure Modes
GDC 24: Separation of Protection and Control Systems
GDC 60: Control of Releases of Radioactive Materials to the Environment
GDC 61: Fuel Storage and Handling and Radioactivity Control
GDC 62: Prevention of Criticality in Fuel Storage and Handling
GDC 63: Monitoring Fuel and Waste Storage

B.) Those that are LWR specific

GDC 35: Emergency Core Cooling
GDC 36: Inspection of ECCS
GDC 37: Testing of ECCS
GDC 38: Containment Heat Removal
GDC 39: Inspection of Containment Heat Removal
GDC 40: Testing of Containment Heat Removal

C.) All other GDCs can, with appropriate modifications, be made technology-neutral and risk-informed.

Table G-2 NS-R-1 comparison.

IAEA Safety Standards	Technology-Neutral Framework
1. 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	• Covered in principle
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.	• Covered 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 practible Level 5: mitigation of the radiological consequences of potential releases of radioactive materials that may result from accident conditions	• DID covered, but objectives, scope and approach differ from IAEA
Safety functions The objective of the safety approach shall be to provide adequate means to maintain the plant in a normal operational state. At all levels of operation and accidents design shall Control Radioactivity Remove heat from the core Confine radioactive materials and control operational discharges A systematic approach shall be followed to identify structures, systems, and components that are necessary to fulfill the safety function.	• Covered in principle through protective strategies
2. Oversight/Enforcement	

Table G-2 NS-R-1 comparison.

IAEA Safety Standards	Technology-Neutral Framework
3. 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. Develop and strictly adhere to sound procedures. Review, monitor and audit all safety related design matters on a regular basis. Ensure that a safety culture is maintained.	• Organization and management not addressed • Procedures are addressed • Safety culture is not addressed
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 complimentary probabilistic safety analyses. Design shall ensure that the generation of radioactive waste is kept to the minimum practicable.	• Covered in principle • Covered in principle • Covered in principle • Not addressed
4. Tracking and Records Schema/Requirements	
Safety Classification All structures, systems and components including software that are important to safety shall be identified and classified according to their safety function. The method for classifying safety significant equipment shall be based primarily on deterministic analysis with complementary probabilistic analysis. System interfaces shall be designed such that systems with lower safety significance shall never propagate failure to systems of greater safety significance.	• Covered in principle • Covered in principle • Not addressed
5. Safety Objectives	
Independent Verification of the Safety Assessment	

Table G-2 NS-R-1 comparison.

IAEA Safety Standards	Technology-Neutral Framework
Accident Prevention and Plant safety Characteristics Plants shall be designed such that sensitivity to accidents is minimized. PIE produces no significant safety related effect or produces 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.	• Covered in principle • ?? • Not addressed • Covered in principle • Covered in principle
General Design Basis The design basis shall specify the necessary capabilities of the plant to cope with a specified range of operational states and design basis accidents. Conservative design measures shall be applied and sound engineering practices shall be adhered to in the design basis for normal, abnormal, and accident operation. Performance of the plant in situations beyond design basis shall be addressed in the design.	• Covered in principle
General Requirements for Instrumentation and Control Systems Important to Safety Instrumentation shall 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 shall 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 shall be provided to maintain the plant parameters within specified operational ranges.	• Covered in principle
6. Owner/Management Competency and Fitness Requirements	

Table G-2 NS-R-1 comparison.

IAEA Safety Standards	Technology-Neutral Framework
7. Confidence in Personnel	
Proven Engineering Practices Wherever possible, structures, systems and components important to safety shall 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 shall be demonstrated to be adequate by appropriate research and testing. In the selection of equipment, consideration shall be given to both spurious operation and unsafe failure modes.	• Covered in principle
Operational Experience and Safety Research Design shall take into account relevant operational experience.	• Covered in principle
Safety Assessment A comprehensive safety assessment shall be carried out to confirm that the design as delivered meets the safety requirements. Safety Assessment shall be part of the design process. The basis for safety assessment shall have data derived from safety analysis, operational experience, research and proven engineering practice.	• Covered in principle
Human Factors The design shall be operator friendly and shall be designed to minimize the potential for operational error. The working areas and working environment of the site personnel shall be designed according to ergonomic principles. Systematic consideration of human factors and human machine interface shall be included throughout the design process. The human-machine interface shall be designed in order to provide operators comprehensive but easily manageable information. Verification and Validation of aspects of human factors shall be included at appropriate stages to confirm that the design adequately accommodates all necessary operator actions. Operators shall be considered to have dual roles, that of equipment operators and systems managers. Operators shall 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 shall 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. The design should be aimed at promoting the success of operator actions with due regard for time, physical environment, and physiological demands.	• Covered in principle

Table G-2 NS-R-1 comparison.

IAEA Safety Standards	Technology-Neutral Framework
Control Room A control room shall 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 shall 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 shall be such that personnel can have an overall picture of the status and performance of the plant. Devices shall 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.	• Covered
Emergency Control Center An on-site emergency control center separated from the plant control room shall be provided for use by emergency staff.	• Covered
8. Confidence in Engineering	
Quality Assurance A quality assurance program that describes the overall arrangements for the management, performance and assessment of the plant design shall be prepared and implements. Design, including subsequent changes or safety improvements shall be carried out in accordance with established procedures that call on appropriate engineering. Adequacy of design shall be verified or validated by individuals or groups separate from those originating the design.	• Covered in principle
Operational States Plants shall 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 shall be addressed in the design. The design process shall establish a set of requirements and limitations for safe operation. These requirements and limitations shall be a basis for the establishing of operational limits and conditions.	• Covered in principle
Common Cause Failures The potential for common cause failures of items important to safety shall be considered to determine where the principle of diversity, redundancy, and independence should be applied to achieve the necessary reliability.	• Covered in principle

Table G-2 NS-R-1 comparison.

IAEA Safety Standards	Technology-Neutral Framework
Fail-Safe Design Fail-safe design shall be considered and incorporated into the design of systems and components.	• Covered in principle
Auxiliary Services Auxiliary services supporting safety systems shall be considered part of the safety systems and shall be classified accordingly.	• Covered in principle
Provision for In-Service Testing, Maintenance, Repair, Inspection and Monitoring SSCs shall 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 shall be utilized and conservative safety margins shall be applied.	• Covered in principle
Equipment Qualification A qualification procedure shall 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 shall be included in the qualification program.	• Covered
Ageing Appropriate margin shall be provided to incorporate ageing into SSCs designs throughout the design life.	• Covered
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 shall be evaluated.	• Covered in principle
Interactions between the electrical power grid and the plant Account shall 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.	• Covered in principle
Safety Analysis A safety analysis of the plant design shall be conducted in which methods of both deterministic and probabilistic analysis shall be applied.	• Covered

Table G-2 NS-R-1 comparison.

IAEA Safety Standards	Technology-Neutral Framework
Deterministic Approach Deterministic safety analysis shall 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 shall be verified.	• Covered
Probabilistic Approach A probabilistic safety analysis of the plant shall 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 in framework
In-service Inspection of the Reactor Coolant Pressure Boundary The reactor coolant system pressure boundary shall be designed, manufactured and arranged in a manner that adequate inspections and tests can be made at appropriate intervals. It shall 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 shall 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 shall be ensured that it is possible to inspect relevant pars of the secondary cooling systems.	• Covered in principle

Table G-2 NS-R-1 comparison.

IAEA Safety Standards	Technology-Neutral Framework
Use of Computer Based Systems in Systems Important to Safety Computer systems required by safety systems shall be subject to standards and practices for the development and testing of the hardware and software. The level of reliability shall be commensurate with the safety importance of the system. The level of reliability assumed in the safety analysis for a computer based system shall include a specified conservatism to compensate for the inherent complexity of the technology.	• Covered in principle
Automatic Control Various safety actions shall 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 addressed
Functions of the Protection System The protection system shall 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.	• Covered in principle
Reliability and Testing of the Protection System The protection system shall be designed for high functional reliability and periodic testability commensurate with the safety function of the system. Design shall 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 shall 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 shall be designed to permit periodic testing of its function when the reactor is in operation. Protection systems shall be designed to minimize the likelihood that operator actions could defeat the effectiveness of the protection system.	• Covered in principle

Table G-2 NS-R-1 comparison.

IAEA Safety Standards	Technology-Neutral Framework
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 shall be used; The whole development process shall be systematically documented and reviewable; An assessment of the computer based system shall 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 shall be provided.	• Covered in principle
9. Contingency Planning	
Requirements for Defense-in-Depth Multiple physical barriers to uncontrolled release of RAM. Shall be conservative, and construction shall be of high quality. Shall provide for control of the plant behavior during and following an PIE using inherent and engineered features. Shall provide for supplementing control of the plant, by the use of automatic activation of safety systems and operator actions. Shall provide for equipment and procedures to control the course and limit the consequences of accidents. Shall provide multiple means for ensuring that each of the fundamental safety functions is performed. Design shall 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 shall prevent all but the most improbable events. Design shall 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.	• Framework DID has different objectives, scope and approach
Categories of Plant States Plant states shall be identified and grouped into a limited number of categories according to their probability of occurrence.	• Covered
Postulated Initiating Events Plant design shall acknowledge that plant challenges can occur at all levels of defense-in-depth and design measures shall be provided to ensure that the necessary safety functions are maintained.	• Covered
Internal Events All those internal events which could affect plant safety shall be identified including: Fires and explosion, and Other internal hazards.	• Covered

Table G-2 NS-R-1 comparison.

IAEA Safety Standards	Technology-Neutral Framework
External Events A combination of deterministic and probabilistic methods shall be used to select a subset of external events which the plant is designed to withstand. Human caused and nature caused external events shall be considered in the design.	• Covered
Site Related Characteristics Where combinations of randomly occurring events could credibly lead to abnormal or accident conditions, they shall be taken into account in the design.	• Covered
Design Rules The engineering design rules for structures, systems, and components shall be specified and shall comply with the appropriate accepted national, or international or foreign engineering standards. Designs shall maintain sufficient margin to safety against seismic events.	• Covered in principle
Design Basis Accidents A set of design basis accidents shall be derived from potential accidents for the purpose of setting the boundary conditions for SSCs. Where prompt and reliable action is required, automatic systems shall be incorporated into the design. Provision for adequate instrumentation shall 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 shall be placed in the most suitable location to ensure its ready availability.	• Covered 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 al barriers are referred to as severe accidents. Assessment and mitigation of these events shall be performed using best estimate techniques. Combinations of safety and non-safety systems may be considered in the mitigation of severe accidents.	• Covered in principle
Single Failure Criterion The single failure criterion shall be applied to each safety group incorporated in the plant design. Spurious action shall 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 shall not be required for high quality passive components.	• Not addressed • Framework uses PRA
Systems containing fissile and radioactive materials shall be designed to be adequate in operational and design basis accidents.	• (?)

Table G-2 NS-R-1 comparison.

IAEA Safety Standards	Technology-Neutral Framework
Escape Routes and Means of Communication Nuclear power plants shall 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 shall be provided so that all personnel on site can ve warned and instructed. Availability of communications necessary for safety within the immediate vicinity of the site and to off site agencies shall be ensured at all times.	• Not addressed
Decommissioning Consideration shall be given to incorporating features that will facilitate the decommissioning and dismantling of the plant. In particular: Choice of materials such that radioactive waste shall be minimized; Access capabilities that may be necessary; and Facilities necessary for storing radioactive waste generated in both operation and decommissioning of the plant.	• Not addressed
Internal Structures of the Containment The design shall provide for ample flow routes between separate compartments inside the containment. Consideration shall be given to the internal structures during severe accidents.	• Not addressed - 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 shall have suitable redundancy in components and features. Consideration shall be given to the clean up of containment atmosphere during severe accidents.	• Not addressed - LWR specific
10. Engineering Prescriptives	
Sharing of Safety Related Reactor Systems shall be Avoided. When systems are shared, systems shall be demonstrated that safety requirements are met of all reactors under all conditions.	• Covered in principle
Power Plants used for Cogeneration Power plants used for cogeneration, heat generation or desalination shall be designed to prevent radioactive material from the nuclear plant to the desalination or district heating unit under all conditions.	• Not addressed

Table G-2 NS-R-1 comparison.

IAEA Safety Standards	Technology-Neutral Framework
General Design	
Reactor core and associated coolant, control and protection systems shall 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 shall 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 shall 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 shall be minimized. The core and coolant and control and protection systems shall be designed to enable adequate inspection and testing.	• Covered in principle
Fuel Elements and Assemblies	
Fuel elements and assemblies shall be designed to withstand satisfactorily the anticipated irradiation and environment conditions in the reactor core.	• Covered in principle
The deterioration considered shall include that arising from differential expansion and deformation, irradiation, internal and external pressure, static and dynamic loading including vibration, and chemical effects.	• Covered in principle
Specified fuel design limits shall not be exceeded in normal operation and significant occurrences shall not cause further deterioration.	• Covered
Fuel assemblies shall be designed to permit adequate inspection of their structure and component parts after irradiation.	• Not addressed
Requirements shall be maintained in the event fuel management strategy is changed.	• Covered in principle
Control of Reactor Core	
Reactivity, criticality and fuel assembly integrity shall be maintained for all levels and distributions of neutron flux in all modes of operation. Provision shall be made for the removal of non-radioactive substances including corrosion products which may compromise safety systems.	• Covered in principle

Table G-2 NS-R-1 comparison.

IAEA Safety Standards	Technology-Neutral Framework
Reactor Shutdown	
Means shall 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.	• Covered
There shall be at least two different systems available to shutdown reactor.	• Covered
At least one of the systems shall 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.	• Covered in principle
In judging the adequacy of the means of shutdown, considerations shall be given to failures arising anywhere in the plant which could prevent shutdown systems from operating.	• Covered in principle
The means of shutdown shall be adequate to prevent or withstand inadvertent increases in reactivity by insertion during the shutdown including during refueling.	• Covered in principle
Instrumentation shall be provided and tests shall be specified to ensure that the shutdown means are always in the state stipulated for the given plant conditions.	• Covered in principle
In the design of reactivity control devices, account shall be taken of wear-out, and the effects of radiation.	• Covered in principle
Reactor Coolant System	
Reactor coolant systems and associated auxiliary systems, controls and protection systems shall 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 shall 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 shall be designed and constructed to be of the highest quality. The pressure retaining boundary for reactor coolant shall 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 shall 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 shall be such as to minimize the likelihood of failure.	• Covered in principle
Inventory Control	
Provisions shall be made for controlling the inventory and pressure of coolant to prevent exceeding specified design limits.	• Covered in principle

Table G-2 NS-R-1 comparison.

IAEA Safety Standards	Technology-Neutral Framework
Removal of Residual Heat from the Core Means for removing residual heat shall be provided. Interconnection and isolation capabilities shall be provided to ensure reliability of residual heat removal systems.	• Covered in principle
Emergency Core Cooling Core cooling shall 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 shall be provided. Adequate consideration shall be given to extending the capability to remove heat from the core following a severe accident.	• Covered in principle
Inspection and Testing of Emergency Core Cooling Systems The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components and to permit periodic testing.	• Covered in principle
Heat Transfer to an Ultimate Heat Sink Systems shall be provided to transfer residual heat from structures, systems, and components important to safety to an ultimate heat sink. Reliability of the systems shall be achieved by an appropriate choice of measures. Natural phenomena and human induced events shall be taken in account in the design of the systems in the consideration of diversity of an ultimate heat sink. Adequate consideration shall be given to extending the capability to transfer residual heat from the core to an ultimate heat sink in consideration of severe accident.	• Covered in principle
Design of the Containment System A containment system shall 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 shall be taken into account in the design of the containment system.	• LWR specific • No specific requirement for containment in framework

Table G-2 NS-R-1 comparison.

IAEA Safety Standards	Technology-Neutral Framework
Strength of the Containment Structure The strength of the containment structure, including access openings and penetrations and isolation valves shall be designed with sufficient safety margins on the basis of: Internal overpressure Internal underpressure Temperatures Dynamic effects Reaction forces Chemical actions Radiolytic actions Provision shall be made to maintain the integrity of containment in a severe accident.	• LWR specific • No specific requirement for containment in framework
Capability for Containment Pressure Tests Containment shall be designed to allow for pressure testing.	• LWR specific • No specific requirement for containment in framework
Containment Leakage Containment shall be designed so that maximum leakage is not exceeded in design basis accidents. Containment shall be designed and constructed so that leak rate can be tested at the design pressure. Consideration shall be given to controlling leakage in the event of a severe accident.	• LWR specific • No specific requirement for containment in framework
Containment Penetrations The number of penetrations through the containment shall be kept to a minimum. Penetrations shall meet the same design requirements as the containment structure. Resilient seals or expansion bellows shall be designed to have the capability for leak testing at design pressure. Consideration shall be given to penetrations remaining functional in the event of severe accidents.	• LWR specific • No specific requirement for containment in framework
Containment Isolation Each line that penetrates the containment as part of the reactor coolant pressure boundary of that which is connected directly to the containment atmosphere shall be automatically and reliably 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 shall have at least one adequate containment isolation valve. Consideration shall be given to isolation devices remaining functional during sever accident.	• LWR specific • No specific requirement for containment in framework
Containment Air Locks Access to the containment shall be through airlocks equipped with doors that ere interlocked to ensure isolation during operations and accidents. Consideration shall be given to severe accidents.	• LWR specific • No specific requirement for containment in framework

Table G-2 NS-R-1 comparison.

IAEA Safety Standards	Technology-Neutral Framework
Removal of Heat from the Containment The capability to remove heat from the reactor containment shall be ensured. Consideration shall be given to removing heat from the containment during severe accidents.	• LWR specific • No specific requirement for containment in framework
Coverings and Coatings Coverings and coatings shall be selected in order to minimize interference with other safety functions and fulfill their own safety functions even with deterioration.	• Not addressed
Supplementary Control Room Sufficient instrumentation and control equipment shall 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.	• Covered
Separation of Protection and Control Systems Interface between the protected system and the control systems shall be prevented.	• Covered in principle
Emergency Power Supplies It shall 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 shall have a reliability and form that are consistent with all the requirements of the safety systems to be supplied. It shall be possible to test the functional capability of the emergency power supply.	• Covered in principle
11. Security of Material and Facilities Requirements	
Control of Access Plans shall 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 shall be prevented.	• Covered in principle
12. Containment and Exposure Requirements	
Radiation Protection and Acceptance Criteria In the design of plants, all actual and potential sources of radiation shall be identified, properly considered, and strictly controlled. Measures shall be taken in design to ensure that radiation protection and doses to the public and site personnel do not exceed prescibed limits and are kept as low as reasonably achievable. Designs shall 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 shall be restricted to a very low likelihood of occurrence.	• Covered in principle

Table G-2 NS-R-1 comparison.

IAEA Safety Standards	Technology-Neutral Framework
Transport and Packaging Transport and packaging for fuel and radioactive waste shall be incorporated into plant designs.	• Not addressed
Removal of Radioactive Substance Adequate facilities shall be provided for the removal of radioactive substances from the reactor coolant, including corrosion and fission products.	• Covered in principle
Waste Treatment and Control Systems Adequate systems shall be provided to treat radioactive liquid and gaseous effluents in order to keep the quantities radioactive discharges as low as reasonably achievable. Adequate systems shall be provided for the handling of radioactive wastes and for storing waste on site for extended periods of time until disposal.	• Covered in principle
Control of Release of Radioactive Liquids to the Environment Design shall include suitable means to control the release of radioactive liquids to the environment.	• Covered in principle
Control of Airborne Radioactive Material Ventilation systems with appropriate filtration shall: 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.	• Covered in principle
Control of Release of Gaseous Radioactive Material to the Environment Ventilation shall contain appropriate filtration to control the release of airborne radioactive substances to the environment. Filter systems shall be sufficiently reliable and achieve necessary retention factors.	• Covered in principle
Handling and Storage of Non-Irradiated Fuel Handling and storage systems for non-irradiated fuel shall 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.	• Covered in principle

Table G-2 NS-R-1 comparison.

IAEA Safety Standards	Technology-Neutral Framework
Handling and Storage of Irradiated Fuel Handling and storage for irradiated fuel shall 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 shall provide: 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).	• Covered in principle
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 shall be taken of the potential buildup of radiation levels with time in areas of personnel occupancy.	• Covered in principle

Table G-2 NS-R-1 comparison.

IAEA Safety Standards	Technology-Neutral Framework
Design for Radiation Protection Suitable provision shall be made in the design and layout of the plant to minimize exposure and contamination from all sources. The shielding design shall be such that radiation levels in operating areas do not exceed the prescribed limits, and shall facilitate maintenance and inspection so as to minimize exposure of maintenance personnel. Plant layout and procedures shall provide for the control of access to radiation areas and areas of potential contamination. Provision shall be made for appropriate decontamination facilities for both personnel and equipment and for handling any radioactive waste.	• Covered in principle
Means of Radiation Monitoring Equipment shall be provided to ensure that there is adequate radiation monitoring in operational and accident states. Stationary dose rate meters shall be provided for monitoring the local radiation dose rate at places routinely occupied by operating personnel. Monitors shall be provided for measuring the activity of radioactive substances in the atmosphere in those areas routinely occupied by personnel. Stationary equipment and laboratory facilities shall 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 shall be provided for monitoring the effluents prior to or during discharge to the environment. Instruments shall be provided for measuring radioactive surface contamination. Facilities shall be provided for the monitoring of individual doses to and contamination of personnel. In addition to monitoring within the plant, arrangements shall 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.	• Covered in principle
13. Regulation Burden Mitigation	
Equipment Outages Plants shall be designed such that reasonable on-line maintenance and testing of systems important to safety can be conducted without the necessity to shut down.	• Not addressed

Table G-3 NS-R-2 comparison

IAEA Safety Standards	Technology-Neutral Framework
Operating Organization	
- functions	• not included
- responsibilities	• not included
- 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	• not included
- training program	• included
- use of simulators	• included
- AM training	• included
- Operator experience feedback	• included
Commissioning Program	
- testing	• not included
- baseline data collection	• not included
Plant Operations	
- operational limits (tech spec)	• included
- procedures	• included
- core management and fuel handling	• not included
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	• not included
- restart after abnormal occurrences	• included

Table G-3 NS-R-2 comparison

IAEA Safety Standards	Technology-Neutral Framework
Plant Modifications - regulatory approval - work control - update documentation	 • included • included • included
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 (living PRA) • included • included
Decommissioning - funding arrangements - preparation for decommissioning	 • included • not included

Table G-4 **Comparison of NEI 02-02 proposed Part 53 to the framework topics.**

NEI Proposed Content of Part 53	Framework
SUBPART A - General Provisions 53.1 Scope 53.2 Definitions 53.3 Interpretations 53.4 Written communications 53.5 Deliberate misconduct 53.7 Employee protection 53.8 Information collection requirements: OMB approval 53.9 Completeness and accuracy of information 53.10 Risk assessments and classification of structures, systems and components 53.15 Quality Assurance	53.1 through 53.9 are included in the framework and would be very similar to the corresponding 10 CFR 50 sections. Framework includes a separate chapter (Chapter7) on risk assessment and a section on classification. Framework has QA as a topic.
Subpart B - Reactor Safety **Design and Construction** 53.20 Initiating events and prevention 53.21 Mitigation 53.22 Functional barriers to radionuclide release	Framework has guidance on selection of initiating events, accident prevention and mitigation and barriers.
Subpart C - Operational Provisions 53.30 Operational requirements 53.31 Changes, tests, and experiments **Radiation Protection** 53.33 Public radiation safety 53.34 Occupational radiation safety 53.35 Source term **Emergency Preparedness** 53.40 Emergency preparedness **Physical Security** 53.45 Security	Covered in framework. Covered in framework. Covered in framework. Covered in framework.

Table G-4 Comparison of NEI 02-02 proposed Part 53 to the framework topics.

NEI Proposed Content of Part 53	Framework
Subpart D - Administrative Provisions	
53.50 Agreement limiting access to restricted data 53.51 Ineligibility of certain applicants 53.52 Public inspection of applications 53.53 Hearings and report of the Advisory Committee on Reactor Safeguards 53.54 Inspections 53.55 Jurisdictional limitations **Requirement of License, Exceptions** 53.60 License required 53.61 Exceptions and exemptions from licensing requirements 53.62 Specific exemptions 53.63 Attacks and destructive acts by enemies of the United States; and defense activities **Classification and Description of Licenses** 53.65 Power reactor license	53.50 through 53.65 are included in the framework and would be very similar to the corresponding 10 CFR 50 section.
License Applications, Transfers, Suspensions and Amendments: Form, Contents 53.70 Filing of applications for licenses, oath or affirmation 53.71 Combining applications 53.72 Elimination of repetition 53.73 Contents of applications; general information 53.74 Contents of applications; technical information 53.75 Transfer of licenses 53.76 Termination of power reactor licenses 53.77 Amendment to a license 53.78 Public notice and state consultations on license amendments 53.79 Evocation, suspension, modification of license for cause 53.80 Retaking possession of special nuclear material 53.81 Commission order for operation after revocation 53.82 Suspension and operation in war or national emergency 53.83 License conditions 53.85 Combining licenses 53.86 Common standards	53.70 through 53.86 are included in the framework, and would be modified from what is currently in 10 CFR 50, where necessary, to be risk-informed.

Table G-4 Comparison of NEI 02-02 proposed Part 53 to the framework topics.

NEI Proposed Content of Part 53	Framework
53.87 Issuance of combined licenses 53.88 Selective implementation	53.87 and 53.88 are not in the framework. Combined licenses are covered in 10 CFR 52 and selective information is not addressed.
Reporting and Notification 53.90 Documentation update requirements 53.91 Notifications 53.92 Reporting requirements 53.93 Notification of change in operator or senior operator status **Financial Considerations** 53.95 Financial assurance for decommissioning 53.96 Creditor regulations **Backfitting** 53.100 Backfitting **Enforcement** 53.105 Violations 53.106 Criminal penalties **US / IAEA Safeguards Agreement** 53.110 Installation information and verification	53.90 through 53.110 are included in the framework topics.

H. Guidance for the Formulation of 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 3). 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.

H.1 Step 1 – Identifying the Performance Objective and its Context

Purpose – To define a performance objective for the SSC 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 4, 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 framework 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.

H.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 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 4 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.

H.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 4.

“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?"

H.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 analyst'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 (as considered in Chapter 4) 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.

H.5 Step 5 – Formulating a Performance-Based Requirement

Purpose – To determine the appropriate implementation of a performance-based approach within the regulatory framework.

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 3, 4, 5, and 6, 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 framework 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 framework 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 framework 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 framework itself should be subjected to critical scrutiny for inappropriate incentives.

Step 5e: Is it worth modifying the regulatory framework 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 framework.

I. Technology-Neutral Requirements and the Need for Technology Specific Guidance

The purpose of this appendix is to document the technology-neutral requirements that result from the application of the guidance in Section 8.5 of the framework to the topics listed in Table 8-6 and 8-9. Table I-1 contains the topics from Table 8-6 and 8-9 and (when complete) will contain the technology-neutral requirements that correspond to each topic. Table I-1 will also indicate whether or not technology-specific guidance will be necessary for their implementation. Currently, Table I-1 only contains some draft example technology-neutral requirements to illustrate the approach and level of detail envisioned in the requirements. These example requirements may change as comments are received and additional work related to requirements development proceeds. Also, in a future update, additional draft requirements will be added to Table I-1.

Table I-1 Technology-neutral requirements and the need for technology-specific guidance.

Topic	TN Requirement	Technology-Specific Guidance Required?
(A) Overall Requirements		
1) F-C curve		No
2) QHOs (including integrated risk)		No
3) Criteria for selection of LBEs		No
4) Keep initiating events with potential to defeat two or more protective strategies <10^{-7}/plant year		Yes
5) Criteria for safety classification and special treatment		Yes
6) LBE deterministic acceptance criteria: • frequent events • infrequent events • rare events • link to siting		Yes
7) Analysis guidelines • realistic analysis, including failure assumptions • source term		No
8) Siting		No

Table I-1 Technology-neutral requirements and the need for technology-specific guidance.

Topic	TN Requirement	Technology-Specific Guidance Required?
9) Defense-in-Depth principle and process		No
10) QA/QC		No
11) PRA scope and quality		No
(B) Physical Protection		
1) General		No
2) Perform security assessment integral with design		No
3) Security performance standards		No
(C) Good Design Practices		
1) Use consensus design codes and standards	1) The design of safety significant systems structures and components (SSCs) shall be based 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.	Yes - will need to identify acceptable codes and standards
2) Materials qualification		Yes
3) Provide 2 redundant, diverse, independent means for reactor shutdown and decay heat removal		No
4) Minimum - 2 barriers to FP release		Yes
5) Containment functional capability		Yes
6) Need to consider degradation and aging mechanisms in design		No

Table I-1 Technology-neutral requirements and the need for technology-specific guidance.

Topic	TN Requirement	Technology-Specific Guidance Required?
7) Reactor inherent protection (i.e., no positive power coefficient, limit control rod worth, stability, etc.)	7) The reactor shall 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 reactor core power and flow rate. Control rod worth shall be limited such that the inadvertent removal of one control rod shall not cause the reactor to go critical. Control rods shall 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).	No
8) Human factors considerations		No
9) Fire protection		Yes
10) Control room design		No
11) Alternate shutdown location		No
12) Flow blockage prevention		No
13) Reliability Assurance Program		No
14) Research and Development		No
15) Combustible gas control		No
16) Coolant/water reaction control		Yes
17) Prevention of brittle fracture		Yes
18) Leak before break		Yes
19) Spent fuel storage		No
20) I and C System • analog • digital • HMI		No

Table I-1 Technology-neutral requirements and the need for technology-specific guidance.

Topic	TN Requirement	Technology-Specific Guidance Required?
21) Criticality prevention		No
22) Protection of operating staff during accidents		No
(D) Good Construction Practices		
1) Use accepted codes, standards, practices		Yes
2) Security		No
3) NDE		Yes
4) Inspection		Yes
5) Testing		Yes
(E) Good Operating Practices		
1) Radiation protection during routine operation		No
2) Comprehensive maintenance program		No
3) Personnel qualification		No
4) Training		No
5) Procedures	5) Procedures shall be developed and used for the conduct of operations, maintenance and responding to off-normal events. The procedures shall be verified by testing in the plant, on simulators or on mock-ups. Procedures shall be controlled and maintained up to date.	No
6) Use of simulators		No
7) Staffing		Yes
8) Aging management program		No
9) Surveillance		No

Table I-1 Technology-neutral requirements and the need for technology-specific guidance.

Topic	TN Requirement	Technology-Specific Guidance Required?
10) ISI	10) An in-service inspection (ISI) program shall be developed and implemented to inspect safety significant SSCs to ensure their availability and reliability. ISI techniques used shall be qualified for materials, configurations and service conditions expected.	Yes - will need to identify acceptable ISI methods or standards
11) Testing		Yes
12) Technical specifications		Yes
13) Develop EOP and AM procedures integral with design		No
14) EP		No
15) Monitoring and feedback		No
16) Corrective action program		No
17) Work control		No
18) Living PRA		No
19) Security		No
(F) Administrative		
1) Standard format and content of applications		No
2) Change control process		No
3) Record keeping		No
4) Documentation control		No
5) Reporting		No
6) Monitoring		No
7) Corrective action program		No
8) Backfitting		No
9) License amendments		No

Table I-1 Technology-neutral requirements and the need for technology-specific guidance.

Topic	TN Requirement	Technology-Specific Guidance Required?
10) Exemptions		No
11) Other legal and process items from 10CR50		No

A.1. "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.

A.3. USNRC, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, December 1990.

A.4. USNRC, "Evaluation of Severe Accident Risks: Surry, Unit 1," NUERG/CR-4551, Vol. 3, October 1990.