Working Draft — NUREG-1860

FRAMEWORK FOR DEVELOPMENT OF A RISK-INFORMED, PERFORMANCE-BASED 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 May 2006.

This version of the framework is a working draft. It does not represent a staff position and is subject to changes and revisions.

ABSTRACT

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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. Table A-1 presents examples of technology specific safety issues which the protective strategies need to address.

(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.

Table A-1 Examples of Technology-Specific Safety Issues Which the Protective Strategies Need to Address

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

Table A-1 Examples of Technology-Specific Safety Issues Which the Protective Strategies Need to Address

Reactor Technology	Protective Strategies				
	Physical Protection	**Stable Operations**	**Protective Systems**	**Barrier Integrity**	**Protective Actions**
• **Water-Cooled:** - ALWR - SCWR		• Materials behavior: - cracking - effect of coolant impurities - fatigue - embrittlement • Fuel performance: - steady state - reactivity transient - decay heat	• Plant response to: - reactivity insertions - loss of coolant - loss of power	• Prevention of RPV rupture: - PTS - other?	• Desire for reduction in EP • Staffing
• **Heavy-Water:** - ACR - APHWR	• On-line refueling implications for theft or diversion	• Pressure tube integrity	• Plant response to: - reactivity insertions - loss of coolant - loss of power • Fuel-coolant / moderator interaction (callandria over-pressure) • Coolant void coeficient	• Capability to accommodate: - fuel-coolant interaction - security-related events	

Table A-1 Examples of Technology-Specific Safety Issues Which the Protective Strategies Need to Address

Reactor Technology	Protective Strategies				
	Physical Protection	**Stable Operations**	**Protective Systems**	**Barrier Integrity**	**Protective Actions**
• **Sodium-Coded**	• Pool versus loop design	• Materials behavior and design codes and standards: - thermal stress - cracking - carbon transfer - nitriding - creep - fatigue - swelling - embrittlement • Fuel performance: - metal fuel - oxide fuel - run beyond clad breach - grid spaces versus wire wrapped fuel pins - reactivity transient - actinide burning • Prevention of loss of coolant • Flow blockage prevention: - sodium freezing - loose material	• Plant response to: - reactivity insertions - loss of power • Sodium/water reaction • Fuel-coolant interaction • Sodium leak detection: - leak before break (i.e., no LB LOCA) • Sodium spills: - fires - reaction with concrete • Prevention of control-rod hydraulic lifting during refueling • Sodium void coeficient • Sodium activation	• Capability to accommodate: - Na spills - Security related events - Fuel-coolant interaction - Recriticality • In-service inspection techniques	• Desire for reduction in EP • Staffing • Source terms

Table A-1 Examples of Technology-Specific Safety Issues Which the Protective Strategies Need to Address

<table>
<tr><th rowspan="2">Reactor Technology</th><th colspan="5">Protective Strategies</th></tr>
<tr><th>Physical Protection</th><th>Stable Operations</th><th>Protective Systems</th><th>Barrier Integrity</th><th>Protective Actions</th></tr>
<tr><td>• Lead Cooled</td><td></td><td>• Materials behavior and design codes and standards:
- thermal stress
- cracking
- effect of coolant impurities
- carbon transfer
- nitriding
- creep
- fatigue
- swelling
- embrittlement
• Fuel performance:
- nitride fuel
- metal fuel
- actinide burning
• Prevention of loss of coolout
• Flow blockage prevention:
- Pb freezing
- loose material</td><td>• Plant response to:
- reactivity insertions
- loss of power
• Pb-water reaction
• Fuel-coolant interaction
• Pb leak detection
• Pb spills:
- reaction with concrete
• Void co-efficient
• Po generation</td><td>• Capability to accommodate:
- Pb spills
- security related events
- fuel-coolant interaction
- recriticality
• In-service inspection techniques</td><td>• Desire for reduction in EP
• Staffing
• Source Term</td></tr>
</table>

A.3 Safety Characteristics of the New Advanced Reactors

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

- Characteristics of inherent properties of core, fuel, moderator, and coolant
- Characteristics of the radioactive material sources (including multiple reactors and non-core related sources)
- Characteristics of radionuclide transport barriers, including:
  - Fuel elements barrier
  - Coolant pressure boundary
  - Reactor building boundary
  - Site selection
- Characteristics of safe stable operating and shutdown states
- Characteristics of the safety functions and success criteria and the design features and SSCs that provide safety functions, including:
  - Inherent safety features
  - Engineered safety feature SSCs
    - Active engineered safety features
    - Passive engineered safety features

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

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

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

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

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

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

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

A.3.5 Lead-Cooled Fast Reactor (LFR)

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

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

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

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

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

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

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

- Transuranic nitride fuel that is chemically compatible with the lead coolant. The high nitride thermal conductivity together with bonding of the fuel and cladding with molten Pb results in low fuel centerline temperatures and small thermal energy storage in the fuel;
- External natural convection-driven passive air cooling of the guard/containment vessel (surrounding the reactor vessel) that is always in effect and removes decay heat power levels;

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

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

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

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

Following an accident such as a loss-of-heat sink without scram in which the reactor power has passively decreased to a low level of after-heat typical of decay heat levels, it may be enough to simply return to power. Or it may only be required for an operator to ultimately insert the shutdown rod(s) to terminate possible fission power at low after-heat levels and render the core 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.3.6 Advanced CANDU Reactor 700 (ACR-700)

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

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

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

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

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

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

A.3.7 Pebble Bed Modular Reactor (PBMR)

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

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

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

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

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

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

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

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

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 1, and (b) the relationship of the requirements of other parts of 10 CFR to the requirements of 10 CFR 50 shown in Table 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 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 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	Part 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
50.30 Filing of Applications	Part 2.101	*Admin requirement* that requires docketing of application under part 2.101 before releasing copies

Part 50 Subpart	Link to other 10 CFR	Content of link
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	Part 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

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 Decommissionin g planning	Part 30	*Administrative*: Guarantee of funds for decommissioning costs may comply with requirements of part 30 App A, B, and C as alternative to 50.75
50.78 IAEA Safeguards	Part 75	*Administrative*: Each holder of CP shall comply with parts 75.6 and 75.11 through 75.14 to permit verification by IAEA

Part 50 Subpart	Link to other 10 CFR	Content of link
50.82 License Termination	Part 20	*Conditions for termination*: Meet dose criteria of part 20 subpart E
50.83 Partial release of site or facility for unrestricted use	Parts 20, 51, 100	*Dose and siting criteria*: public dose remains within limits of part 20 subpart D; siting criteria of part 100 continue to be met; surveys demonstrate compliance with part 20.1402 for unrestricted use areas; compliance with reporting requirements of parts 20.1402 and 51.53
50.91 License amendment	Part 2	*Administrative*: Exceptions for public comment hearings and state consultations under part 2 subpart L; notice for public comment under part 2.105 and, for emergency situations, under part 2.106
50.92 Issuance of amendment	Part 2	*Administrative*: Notice under part 2.105 for amendments involving significant hazards
50.120	Part 55	*Training* of personnel: Comply with part 55.4
Appendix C Financial qualifications for CP	Parts 2 and 9	*Administrative*: Allows applicants to withhold information from public disclosure per parts 2.790 and 9.5

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

Table B-2: Link of Other portions of 10 CFR to 10 CFR Part 50 ((2)

10 CFR Subpart	Part 50 Subpart	Content of link
10 CFR 1.43(a)(2)	Part 50	*Defines duties of NRR Office*, e.g., procedures for licensing, inspection, etc. of facilities licensed under Part 50
10 CFR 2.4	Part 50.2	*Definition of facility* as that defined in 50.2
10 CFR 2.101(a)(3)(I)	Part 50	*Procedure for issuance, amendment, transfer, or renewal of a license; Filing of applications*; additional copies required by Part 50
10 CFR 2.101(a)(5)	50.21(b)(2) or (3), 50.22, Part 50, 50.30f, 50.34(a), 50.33, 50.34(a)(1), 50.37	*Procedure for issuance, amendment, transfer, or renewal of a license; Filing of application*; completeness of application
10 CFR 2.101(a)(5)(a-1)	50.21(b)(2) or (3), 50.22, Part 50	*Procedure for issuance, amendment, transfer, or renewal of a license; Filing of application;* early site suitability issues for construction permit

(2) Where the second column mentions Part 50, it pertains to the entire part.

10 CFR Subpart	Part 50 Subpart	Content of link
10 CFR 2.101(a)(5)(1)	50.34(a)(1), 50.30(f), 50.33(a) through (e), 50.37, Part 50	*Procedure for issuance, amendment, transfer, or renewal of a license; Filing of application*; early site suitability issues for construction permit; content of application
10 CFR 2.101(a)(5)(2)	50.30(f), 50.33, 50.34(a)(1)	*Procedure for issuance, amendment, transfer, or renewal of a license; Filing of application*; early site suitability issues for construction permit; content of application
10 CFR 2.101(a)(5)(3)	50.34a, 50.34(a)	*Procedure for issuance, amendment, transfer, or renewal of a license; Filing of application*; early site suitability issues for construction permit; content of application
10 CFR 2.101(c)(1)	Part 50	*Procedure for issuance, amendment, transfer, or renewal of a license; Filing of application;* information for antitrust review
10 CFR 2.104(a), (b), (c)	50.21(b), 50.35, 50.22, 50.55b	*Hearing on Application*; Notice of Hearing and contents of Notice; administrative
10 CFR 2.105(a)	50.21(b), 50.22, 50.58, 50.91	*Notice of proposed action on application*; administrative
10 CFR 2.106(a)	50.21(b), 50.22	*Notice of issuance of license or license amendment*; administrative
10 CFR 2.109	50.21(b), 50.22	*Effect of timely renewal application of a license*; administrative
10 CFR 2.202(e)	50.109, Part 50 license	*Procedure for imposing requirements* by order modifying Part 50 license by backfit; administrative
10 CFR 2.310(a)	Part 50	*Selection of hearing procedures*; administrative
10 CFR 2.310(h)	Part 50	*Selection of hearing procedures*; administrative
10 CFR 2.328	50.21(b), 50.22	*Selection of hearing procedures*; Hearings to be public
10 CFR 2.329	50.21(b), 50.22	*Prehearing conference*; notice of timing; administrative
10 CFR 2.401	50.22	*Notice of hearing on applications* pursuant to Appendix N of Part 52 for construction permits for reactors described in 50.22
10 CFR 2.402	50.22	*Separate hearings* on particular issues
10 CFR 2.501	50.22	*Notice of hearing on applications related to Appendix M of Part 52* to manufacture power reactors of type described in 50.22
10 CFR 2.600 Part 2 Subpart F	50.21(b), 50.22	*Additional procedures applicable* to early partial decisions on site suitability
10 CFR 2.602	50.30(e)	*Filing fees for early review* of site suitability issues
10 CFR 2.603	50.21(b), 50.22, 50.33a	*Docketing of applications* for early review of site suitability

10 CFR Subpart	Part 50 Subpart	Content of link
10 CFR 2.605	50.30(f)	*Additional considerations* on site suitability issues
10 CFR 2.606	50.10(e)	*Partial decisions* on site suitability issues
10 CFR 2.1103, Part 2 Subpart K	Part 50	*Hybrid hearing procedures* for expansion of spent fuel storage capacity at nuclear power plants
10 CFR 2.1202	50.92	*Informal hearing procedures* for NRC adjudications; authority/role of NRC staff in licensing actions that involve significant hazards considerations defined in 50.92
10 CFR 2.1301	Part 50	*Public notice* of receipt of a license transfer application
10 CFR 2.1403	50.92	*Expedited proceedings* with oral hearings; authority and role of NRC staff in licensing actions that involve significant hazards considerations defined in 50.92
10 CFR 8.4	Part 50	*AEC jurisdiction* over nuclear facilities and materials under the Atomic Energy Act
10 CFR 11.7	Part 50	*Criteria and Procedures* for determining eligibility for access to or control over SNM; Definitions
10 CFR 19.2	Part 50	*Notices, Instructions and reports* to workers; Scope of worker inspections and investigations
10 CFR 19.3	Part 50	*Notices, Instructions and reports* to workers; inspection and investigations; purpose
10 CFR 19.20	Part 50	*Notices, Instructions and reports* to workers; inspection and investigations; employee protection
10 CFR 20.1002	Part 50	*Standards for Protection Against Radiation*; General Provisions, scope
10 CFR 20.1003	Part 50	*Standards for Protection Against Radiation*; General Provisions, definitions
10 CFR 20.1101	50.34a	*Standards for Protection Against Radiation*; Radiation Protection Programs
10 CFR 20.1401(a)	Part 50, 50.83	*Standards for Protection Against Radiation*; Radiological Criteria for License Termination; General provisions and scope
10 CFR 20.1401(c)	50.83	*Standards for Protection Against Radiation*; Radiological Criteria for License Termination; General provisions and scope
10 CFR 20.1403(d)	50.82(a)&(b)	*Standards for Protection Against Radiation*; Radiological Criteria for License Termination; Criteria for license termination under restricted conditions
10 CFR 20.1404(a)(4)	50.82 (a)&(b)	*Standards for Protection Against Radiation;* Radiological Criteria for License Termination; Alternate criteria for license termination

10 CFR Subpart	Part 50 Subpart	Content of link
10 CFR 20.2004	Part 50 App I, 50.34, 50.34(a), 50.71, 50.59	*Treatment or disposal* of radioactively contaminated waste oils by incineration
10 CFR 20.2201	50.73, 50.72	*Reports of thefts or loss* of nuclear material at a nuclear power plant
10 CFR 20.2202	50.72	*Notification of incidents* that exceed specified dose guidelines to individuals
10 CFR 20.2203	50.73	*Reports of exposures, radiation levels, and concentrations* of radioactive materials at operating power plants exceeding constraints or limits
10 CFR 20.2206	50.21(b), 50.22, 50.2	*Reports of individual monitoring* of power plant operators
10 CFR 21.2	50.23, 50.55(e), 50.72, 50.73, Part 50	*Scope of reporting of defects and noncompliance* by persons licensed to construct or operate a power plant
10 CFR 21.3	Part 50, 50.34(a), 50.67, App B,	*Reporting of Defects and Noncompliance*: Definitions
10 CFR 21.21	Part 50	*Notification of failure to comply* or existence of a defect and its evaluation
10 CFR 25.5	Part 50	*Access Authorization for Licensee Personnel*: Definitions
10 CFR 25.17	Part 50	*Approval for processing applicants* for license authorization
10 CFR 30.4	Part 50	*Domestic Licensing* of Byproduct Material: Definitions of Production and Utilization Facility
10 CFR 30.50	50.72	*Reporting Requirements*
10 CFR 40.60	50.72	*Domestic Licensing* of Source Material: Reporting Requirements
10 CFR 51.20	Part 50	*Licensing and Regulatory actions* requiring environmental impact statements
10 CFR 51.22	Part 50	*Licensing and regulatory actions* eligible for categorical exclusion or not requiring environmental review
10 CFR 51.50	50.36b	*Environmental Protection Regulations* for Domestic Licensing and related regulatory functions; Environmental report–construction permit stage
10 CFR 51.53	50.82	*Post-operating license* stage environmental review
10 CFR 51.54	50.4	*Manufacturing license* environmental report
10 CFR 51.101	50.10(c)	*NEPA Procedure* - Limitations on Actions
10 CFR 51.106	50.57(c)	*Public hearings* in proceedings for issuance of operating licenses

10 CFR Subpart	Part 50 Subpart	Content of link
10 CFR 52.3	50.2	*Early site permits*; Definitions
10 CFR 52.13, Part 52 Subpart A	Part 50	*Relationship of application of construction permit* under Part 50 to application for early site permit under Part 52, Subpart A
10 CFR 52.15	50.30, 50.4	*Filing of applications for an early site permit* under Part 52, Subpart A
10 CFR 52.17	50.33, 50.34, 50.47, 50.10	*Contents of applications* for early site permit
10 CFR 52.18	Part 50	*Standards for review* of applications
10 CFR 52.25	50.10	*Extent of activities* permitted under early site permit
10 CFR 52.37	50.100	*Early site permit* is a construction permit for purposes of compliance with 50.100
10 CFR 52.39	50.109	*Finality* of early site permit determinations
10 CFR 52.45, Subpart B	50.4, 50.30(a), 50.30(b)	*Standard Design Certifications*: Filing of applications and filing requirements
10 CFR 52.47	Part 50 and Appendices, 50.34	*Standard Design Certifications*; Contents of applications
10 CFR 52.48	Part 50 and Appendices	*Standards* for review of applications
10 CFR 52.51	Part 50	*Administrative review* of applications
10 CFR 52.63	50.109, 50.12, 50.59	*Finality* of standard design certifications
10 CFR 52.75, Subpart C	50.4, 50.30, 50.38	*Combined Licenses*; Filing of applications
10 CFR 52.77	50.33	*Contents of applications*; general information
10 CFR 52.78	50.120	*Contents of applications*; training and qualification of power plant personnel
10 CFR 52.79	50.10, 50.30, 50.34	*Contents of applications*; technical information
10 CFR 52.81	Part 50	*Standards* for review of applications
10 CFR 52.83	Part 50, 50.51, 50.55 (a), (b), (d), 50.58	*Applicability* of Part 50 provisions
10 CFR 52.91	50.10	*Authorization* to conduct site activities
10 CFR 52.93	50.12	*Exemptions and variances*
10 CFR 52.97	50.40, 50.42, 50.43, 50.47, 50.50, 50.91	*Issuance* of combined licenses
10 CFR 52.99	50.70, 50.71	*Inspection* during construction

10 CFR Subpart	Part 50 Subpart	Content of link
10 CFR 52, Appendix A, *II*	50.2, 50.34, 50.36, 50.36a	*ABWR design certification*; Definitions
10 CFR 52, Appendix A, *IV*	50.36, 50.36a, Part 50	*ABWR design certification; additional requirements* and restrictions
10 CFR 52, Appendix A, *V*	Part 50, 50.34	*ABWR design certification; applicable regulations* (identifies exemptions from specific portions of 50.34)
10 CFR 52, Appendix A, *VIII*	50.12, 50.90, 50.109	*ABWR design certification; processes* for changes and departures
10 CFR 52, Appendix A, *X*	50.4, 50.71(e)	*ABWR design certification; records* and reporting
10 CFR 52, Appendix B, *II*	50.2, 50.34, 50.36, 50.36a	*System 80*+ design certification; Definitions
10 CFR 52, Appendix B, *IV*	50.36, 50.36a, Part 50	*System 80*+ design certification; additional requirements and restrictions
10 CFR 52, Appendix B, *V*	Part 50, 50.34, Appendix J	*System 80*+ design certification; applicable regulations (identifies exemptions from specific portions of 50.34 and part 50 Appendix J)
10 CFR 52, Appendix B, *VIII*	50.12(a), 50.90, 50.109	*System 80*+ design certification; processes for changes and departures
10 CFR 52, Appendix B, *X*	50.4, 50.71(e)	*System 80*+ design certification; records and reporting
10 CFR 52, Appendix C, *II*	50.2, 50.34, 50.36, 50.36a	*AP 600 design certification*; Definitions
10 CFR 52, Appendix C, *IV*	50.36, 50.36a, Part 50	*AP 600 design certification*; additional requirements and restrictions
10 CFR 52, Appendix C, *V*	Part 50, 50.34, 50.55a, 50.62, GDC 17, GDC 19	*AP 600 design certification*; applicable regulations (identifies exemptions from specific portions of 50.34, 50.55a, 50.62 and part 50 Appendix A, GDC 17 and GDC 19)
10 CFR 52, Appendix C, *VIII*	50.12(a), 50.90, 50.109	*AP 600 design certification*; processes for changes and departures
10 CFR 52, Appendix C, *X*	50.4, 50.71(e)	*AP 600 design certification*; records and reporting
10 CFR 52, Appendix D, *II*	50.2, 50.34, 50.36, 50.36a	*AP 1000 design certification*; Definitions
10 CFR 52, Appendix D, *IV*	50.36, 50.36a, Part 50	*AP 1000 design certification*; additional requirements and restrictions
10 CFR 52, Appendix D, *V*	Part 50, 50.34(f), 50.62(c), GDC 17	*AP 1000 design certification*; applicable regulations (identifies exemptions from specific portions of 50.34, 50.62 and part 50 Appendix A, GDC 17)

10 CFR Subpart	Part 50 Subpart	Content of link
10 CFR 52, Appendix D, *VIII*	50.12(a), 50.90, 50.109	*AP 1000 design certification*; processes for changes and departures
10 CFR 52, Appendix D, *X*	50.4, 50.59, 50.71(e)	*AP 1000 design certification*; records and reporting
10 CFR 52, Appendix M	50.4, 50.10, 50.12, 50.22, 50.23, 50.30, 50.33, 50.34, 50.35, 50.40, 50.45, 50.55, 50.56, 50.57, 50.58, Part 50 Appendices C, E, H, J	*Standardization of Design; Manufacture* of Power Reactors; Construction and Operation of Power Reactors Manufactured Pursuant to Commission License
10 CFR 52, Appendix N	50.4, 50.10, 50.33, 50.33a, 50.34, 50.34a, 50.58, Part 50	*Standardization of Power Plant Design*; Licenses to construct and operate power reactors of duplicate design at multiple sites
10 CFR 52, Appendix O	50.4, 50.22, 50.30, 50.33, 50.34, 50.34a, 50.54f	*Standardization of Design*; Staff Review of Standard Designs
10 CFR 52, Appendix Q	50.4, 50.21, 50.22, 50.30, 50.33, 50.34, 50.4	*Pre-Application* Early Review of Site Suitability Issues
10 CFR 54.3	Part 50, 50.2, 50.21, 50.22, 50.71	*Requirements for Operating License Renewal*; definitions
10 CFR 54.4	50.34, 50.48, 50.49, 50.61, 50.62, 50.63, 50.67	*Requirements for Operating License Renewal*; scope
10 CFR 54.7	50.4	*Requirements for Operating License Renewal*; written communications
10 CFR 54.15	50.12	*Requirements for Operating License Renewal*; specific exemptions
10 CFR 54.17	50.4, 50.30, 50.33	*Requirements for Operating License Renewal*; filing of application
10 CFR 54.19	50.33	*Requirements for Operating License Renewal*; content of application - general information
10 CFR 54.21	50.12	*Requirements for Operating License Renewal*; content of application - technical information
10 CFR 54.33	50.36b, 50.54	*Requirements for Operating License Renewal*; continuation of CLB and conditions of renewed license
10 CFR 54.35	Part 50	*Requirements for Operating License Renewal*; requirements during term of renewed license
10 CFR 54.37	50.71(e)	*Requirements for Operating License Renewal*; additional records and record-keeping requirements

10 CFR Subpart	Part 50 Subpart	Content of link
10 CFR 55.1	Part 50	*Operators' Licenses*; purpose
10 CFR 55.2	Part 50	*Operators' Licenses*; scope
10 CFR 55.4	Part 50	*Operators' Licenses*; definitions
10 CFR 55.5	Part 50	*Operators' Licenses*; communications
10 CFR 55.25	50.74(c)	*Operators' Licenses*; incapacity due to disability or illness
10 CFR 60.152, Subpart G	Part 50, Appendix B	*Disposal of HLW in Geologic Repositories*; implementation of quality assurance program
10 CFR 63.73, Subpart D	50.55(e)	*Disposal of HLW at Yucca Mountain*; records, reports, tests and inspections: reports of deficiencies
10 CFR 70.20a, Subpart C	Part 50	*Domestic Licensing of Special Nuclear Material*; general licenses: license to possess SNM for transport
10 CFR 70.22, Subpart D	Part 50, Part 50 Appendix B	*Domestic Licensing of Special Nuclear Material*; License applications: contents of applications
10 CFR 70.23, Subpart D	Part 50, Appendix B	*Domestic Licensing of Special Nuclear Material*; License applications: requirements for the approval of applications
10 CFR 70.24, Subpart D	50.68, Part 50	*Domestic Licensing of Special Nuclear Material*; License applications: criticality accident requirements
10 CFR 70.32, Subpart E	Part 50, 50.90	*Domestic Licensing of Special Nuclear Material*; conditions of licenses
10 CFR 70.50 Subpart G	50.72	*Domestic Licensing of Special Nuclear Material*; SNM control, records, reports and inspections: reporting requirements
10 CFR 71.101	Part 50 Appendix B	*Packaging and Transport of Radioactive Material*; quality assurance requirements
10 CFR 72.3	Part 50	*Licensing Requirements for Independent Storage* of Spent Fuel, HLW, and GTCC waste; definition of ISFSI
10 CFR 72.30	50.75, Part 50	*Licensing Requirements for Independent Storage of Spent Fuel*, HLW, and GTCC waste; financial assurance and record keeping for decommissioning
10 CFR 72.32	50.47	*Licensing Requirements for Independent Storage of Spent Fuel*, HLW, and GTCC waste; emergency plan
10 CFR 72.40	Part 50	*Licensing Requirements for Independent Storage of Spent Fuel*, HLW, and GTCC waste; issuance of license
10 CFR 72.75	Part 50	*Licensing Requirements for Independent Storage of Spent Fuel*, HLW, and GTCC waste; reporting requirements for specific events and conditions
10 CFR 72.140	Part 50 Appendix B	*Licensing Requirements for Independent Storage of Spent Fuel*, HLW, and GTCC waste; QA requirements

10 CFR Subpart	Part 50 Subpart	Content of link
10 CFR 72.184	Part 50	*Licensing Requirements for Independent Storage of Spent Fuel*, HLW, and GTCC waste; safeguards contingency plan
10 CFR 72.210	Part 50	*Licensing Requirements for Independent Storage of Spent Fuel*, HLW, and GTCC waste; general license for storage of spent fuel at power reactor sites
10 CFR 72.212	50.59	*Licensing Requirements for Independent Storage of Spent Fuel*, HLW, and GTCC waste; conditions of general license
10 CFR 72.218	50.54, 50.82	*Licensing Requirements for Independent Storage of Spent Fuel*, HLW, and GTCC waste; termination of licenses
10 CFR 73.1	Part 50	*Physical Protection of Plants and Materials*; purpose and scope
10 CFR 73.2	Part 50	*Physical Protection of Plants and Materials*; definitions
10 CFR 73.20	Part 50	*Physical Protection of Plants and Materials*; general performance objectives and requirements
10 CFR 73.50	Part 50	*Physical Protection of Plants and Materials*; requirements for physical protection of licensed activities
10 CFR 73.55	50.21, 50.22, 50.54, 50.72, 50.90, 50.109	*Physical Protection of Plants and Materials*; requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage
10 CFR 73.56	50.21, 50.22, 50.54, 50.90	*Physical Protection of Plants and Materials*; personnel access authorization for power plants
10 CFR 73.57	Part 50	*Physical Protection of Plants and Materials*; requirements for criminal history checks of individuals granted unescorted access to a nuclear power facility or access to safeguards information by licensees
10 CFR 73.67	Part 50	*Physical Protection of Plants and Materials*; licensee fixed-site and in-transit requirements for SNM of moderate and low strategic significance
10 CFR 73.71	50.72, 50.73	*Physical Protection of Plants and Materials*; reporting of safeguards events
10 CFR 73, Appendix B	Part 50	*Physical Protection of Plants and Materials*; general criteria for security personnel: definitions
10 CFR 73, Appendix C	Part 50 Appendix E	*Physical Protection of Plants and Materials*; licensee safeguards contingency plans
10 CFR 74.13	50.21, 50.22	*Material Control and Accounting of SNM*; Material Status Reports
10 CFR 74.31	Part 50	*Material Control and Accounting of SNM*; Nuclear material control and accounting for special nuclear material of low strategic significance

10 CFR Subpart	Part 50 Subpart	Content of link
10 CFR 74.41	Part 50	*Material Control and Accounting of SNM*; SNM of moderate strategic significance
10 CFR 74.51	Part 50	*Material Control and Accounting of SNM*; formula quantities of strategic SNM: control and accounting for strategic SNM
10 CFR 75.2	50.78	*Safeguards on Nuclear Material* - Implementation of US/IAEA Agreement; Scope
10 CFR 75.4	50.2	*Safeguards on Nuclear Material* - Implementation of US/IAEA Agreement; definitions
10 CFR 95.5	Part 50	*Security Clearance and Safeguarding* of National Security Information and Restricted Data; definitions
10 CFR 100.1	Part 50	*Reactor Site Criteria*; purpose
10 CFR 100.2	Part 50	*Reactor Site Criteria*; scope
10 CFR 100.3	50.2, 50.21, 50.22, Appendix S	*Reactor Site Criteria*; definitions
10 CFR 100.21	50.34	*Reactor Site Criteria*; non-seismic siting criteria
10 CFR 100.23	50.10, Appendix S	*Reactor Site Criteria*; geologic and seismic siting criteria
10 CFR 100, Appendix A	Part 50 GDC 2, 50.10	*Reactor Site Criteria*; seismic and geologic siting criteria for power plants
10 CFR 140.2	Part 50	*Financial Protection Requirements and Indemnity Agreements*; scope
10 CFR 140.3	50.21	*Financial Protection Requirements and Indemnity Agreements*; definitions
10 CFR 140.10	Part 50	*Financial Protection Requirements and Indemnity Agreements*; provisions applicable only to applicants and licensees other than Federal Agencies and Non-Profit Educational Institutions; scope
10 CFR 140.11	Part 50	*Financial Protection Requirements and Indemnity Agreements*; amounts of financial protection for certain reactors
10 CFR 140.12	Part 50	*Financial Protection Requirements and Indemnity Agreements*; amounts of financial protection required for other reactors
10 CFR 140.13	Part 50	*Financial Protection Requirements and Indemnity Agreements*; amount of financial protection required of certain holders of construction permits
10 CFR 140.20	Part 50	*Financial Protection Requirements and Indemnity Agreements*; indemnity agreements and liens

10 CFR Subpart	Part 50 Subpart	Content of link
10 CFR 140.51	Part 50	*Financial Protection Requirements and Indemnity Agreements*; provisions applicable only to Federal Agencies; scope
10 CFR 140.52	Part 50	*Financial Protection Requirements and Indemnity Agreements*; provisions applicable only to Federal Agencies; indemnity agreements
10 CFR 140.72	Part 50	*Financial Protection Requirements and Indemnity Agreements*; provisions applicable only to nonprofit educational institutions; indemnity agreements
10 CFR 150.15	Part 50	*Exemptions and continued regulatory authority* in agreement states and in offshore waters under Section 274, persons not exempt from regulation for storage of GTCC waste
10 CFR 170.2	Part 50	*Fees for Regulatory Services*; scope
10 CFR 170.3	Part 50, 50.21, 50.22, 50.71	*Fees for Regulatory Services*; definitions
10 CFR 170.12	50.71	*Fees for Regulatory Services*; payment of fees
10 CFR 170.21	50.12	*Fees for Regulatory Services*; schedule of fees
10 CFR 170.41	Part 50	*Fees for Regulatory Services*; failure by applicant or licensee to pay fees
10 CFR 171.3	Part 50	*Annual Fees for Reactor Licensees*; scope
10 CFR 171.5	50.21, 50.22, 50.57	*Annual Fees for Reactor Licensees*; definitions
10 CFR 171.15	Part 50	*Annual Fees for Reactor Licensees*; annual fees for reactors licenses and independent spent fuel storage licenses
10 CFR 171.17	Part 50	*Annual Fees for Reactor Licensees*; proration of annual fees

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.

Cleanup Cost Assessment

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[(3)] in the vicinity of a nuclear power plant of prompt fatalities[(4)] 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[(5)] 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.

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

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(3)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).

(4)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.

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

the 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:

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

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

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

TP(1) = total population to 1 mile

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

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

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

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

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

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

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

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

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

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

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

(3) using results from NUREG-1150 [Ref.3] for the Surry PRA (Table 4.3-1) [Ref.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:

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

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:

$$ILR = \sum_1^M [(LFm * LLRFm) / TP(10)] \quad \text{Equation 5}$$

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:

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

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:

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

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

$$ILR = \sum_1^N CPLFm * LLRFm \quad \text{Equation 8}$$

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:

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

$$LLRFm = CDFm * 1.0$$

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 4×10^{-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:

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

The ILR corresponding to a CDF = 10^{-4} per year is less than the latent cancer QHO of 2×10^{-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.

E. EXAMPLE OF LBE AND SAFETY CLASSIFICATION SELECTION PROCESS

E.1 Introduction

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

In the risk-informed approach used in the framework, there are probabilistically selected LBEs and at least one deterministic LBE. The probabilistic LBEs are selected from PRA sequences. These probabilistically selected LBEs not only include sequences that involve a radionuclide release and lead to a dose at the site boundary and at one mile, but may also include sequences that do not involve any release of radionuclides. The process for identifying these probabilistically selected LBEs is included in this appendix. The deterministic LBE is considered for defense-in-depth purposes, as discussed in Subsection 6.2.2.2. An example of the selection of this deterministic event is not included in this appendix.

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

E.2 Process

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

E.2.1 LBE Selection Process

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

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

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

3. For sequences with point estimate frequencies equal to or greater than 1E-8 per year, determine the mean and 95th percentile frequency.

4. Identify all PRA event sequences with a 95th percentile frequency > 1E-7 per year. Event sequences with 95th percentile frequencies less than 1E-7 per year are excluded from further consideration.

5. Group the PRA event sequences with a 95th frequency percentile > 1E-7 per year into event classes.

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

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

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

9. Verify that the selected LBEs meet the deterministic and probabilistic requirements.

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

E.2.2 Selection of Dose Distance and Duration

The framework uses three frequency categories as shown in Table 6.3 of the main report and summarized below in Table E.1.

Table E.1 LBE Frequency Categories

Category	Frequency	Deterministic LBE Criteria
frequent	$10E-2/ry	- no barrier failure - no impact on safety analysis assumptions
infrequent	< 10E-2/ry to $10E-5/ry	- at least one barrier remains - a coolable geometry is maintained
rare	<10E-5/ry to $10E-7/ry	- none
C applies to all internal and external events C events with mean frequency <10-7/ry do not have to be considered in the design for licensing purposes		

Each category has a unique dose evaluation criterion as indicated below:

Frequent95th percentile of the annual dose to a receptor at the exclusion area boundary (EAB) is less than 100 mrem TEDE (total effective dose equivalent)

Infrequent 95th percentile of the worst 2-hour dose at the EAB meets the frequency-consequence curve

Rare 95th percentile of the 24 hour dose at 1 mile from the EAB meets the frequency-consequence curve

It is therefore necessary to know the frequency category of an event sequence in order to establish the applicable dose end state.

E.2.3 Safety-Significant SSCs Selection

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

E.3 Example Plant

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

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

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

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

E.3.1 Initiating Events

This example uses a simplified set of initiating events that is consistent with those contained in the SPAR models. The initiating events identified in Table E.2 are included.

Table E.2 Initiating Events

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

Table E.2 Initiating Events

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

E.3.2 Event Sequences

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

E.3.2.1 Event Sequence Top Events

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

Table E.3 shows the top events used in the front-line event trees that are questioned directly as a result of an initiating event for this PWR example. Note that additional event trees are often questioned, resulting in additional top events (not shown).

Table E.3 Event Sequence Top Events

Top Event	Description	LODCA	LLOCA	LOCCW-A	LOESW	LOOP	MLOCA	SGTR	SLOCA	TRANS	RHR-DIS-V	RHR-HL-V	RHR-SUC-V	SI-CLDIS-V
ACC	RCS Accumulators Re-flood on Demand		Y				Y							

Table E.3 Event Sequence Top Events

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

Table E.3 Event Sequence Top Events

Top Event	Description	LODCA	LLOCA	LOCCW-A	LOESW	LOOP	MLOCA	SGTR	SLOCA	TRANS	RHR-DIS-V	RHR-HL-V	RHR-SUC-V	SI-CLDIS-V
RPS	Reactor Protection System Operates on Demand	Y		Y	Y	Y	Y	Y	Y	Y				
SSC	Secondary Side Cooling					Y								
SG-ISOL	Operator Isolates Affected SG							Y						

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

Table E.4 shows ten different types of top events that are used in the example PRA to model accident progression subsequent to core damage.

Table E.4 Containment Related Top Events

Top Event	Description
CIF	Containment Isolation
RCSDEP-LATE	No Late RCS Depressurization
SGDEP-LATE	No Late Secondary Depressurization
ISGTR	No Induced Steam Generator Tube Rupture
H2	No Containment Failure due to Hydrogen Burn
PREVB-INVREC	In Vessel Recovery before Vessel Breach
RCSPIPE-MELT	No Melt of Surge Line, Hot Legs
DCH	No Containment Failure due to Direct Containment Heating (DCH) with Hydrogen Burn
CMTSTF	No Containment Melt-through via Seal Table Failure
LER	No Large Early Release

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

E.3.2.2 Event Sequence End States

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

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

E.3.3 Dose End States

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

E.3.4 Nomenclature

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

E.4 Example: Identification of LBEs

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

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

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

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

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

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

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

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

This step establishes the complete set of event sequences that will be processed to determine the LBEs. An quantification truncation limit of 1E-15 per year was used. In this example, the 13 initiating events produce a total of 1,536 sequences. Table E.5 summarizes the results.

Table E.5 Accident Sequences

Initiating Event	Number of Sequences	Number of Sequences point estimate > 1E-08	Number of Sequences 95^{th} > 1E-07
IE-LDCA	64	9	7
IE-LLOCA	10	1	1
IE-LOCCW-A	141	5	3
IE-LOESW	190	6	6
IE-LOOP	829	47	24

Table E.5 Accident Sequences

Initiating Event	Number of Sequences	Number of Sequences point estimate > 1E-08	Number of Sequences 95^{th} > 1E-07
IE-MLOCA	13	2	2
IE-SGTR	68	15	13
IE-SLOCA	84	4	4
IE-TRANS	121	18	16
IE-V-RHR-DIS	4	0	0
IE-V-RHR-HLDIS	4	0	0
IE-V-RHR-SUC	4	3	3
IE-V-SI-CLDIS	4	0	0
Total	**1,536**	**110**	**79**

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

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

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

In the example, an uncertainty analysis is performed on the 110 sequences that were determined to be in scope by Step 2. Of these sequences, 79 sequences have a 95^{th} percentile equal to or larger than 1E-7 per year. The 31 sequences that are screened (those sequences less than 1E-7) are shaded in Table E.6.

Note that the characterization of the dose (exposure time and distance) associated with the sequence end state is dependent on the 95^{th} percentile frequency of the sequence. In this example, the 1 mile 24 hour dose was determined for all core damage sequences with a mean frequency greater than1E-8 per year. These are indicated by the term "1 mile" in Table E.6. One core damage event sequence, LOESW 04-01-01, has a 95^{th} percentile frequency greater than 1E-5 per year and is therefore considered to be in the Infrequent category and requires an assessment of the worst 2-hour dose at the exclusion area boundary. This dose is annotated by the term "EAB" in Table E.6.

Table E.6 Accident Sequences for Sequences with a Point Estimate > 1E-8

Initiating Event	Sequence	Description	Point Estimate (per year)	Mean (per year)	95th (per year)	Mean Dose (REM)	95th Dose (REM)
LDCA	01	Loss of a DC bus with all remaining systems successful	2.5E-03	2.51E-03	1.0E-02	<1mR	<1mR
LDCA	10-01-01-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	4.1E-08	3.8E-08	1.6E-07	1 mile 0.6R	1 mile 1.2R
LDCA	10-01-03-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	3.6E-08	3.28E-08	1.4E-07	1 mile 0.6R	1 mile 1.2R
LDCA	10-01-06-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	4.2E-08	3.9E-08	1.7E-07	1 mile 0.6R	1 mile 1.2R
LDCA	10-01-07-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	3.8E-06	3.5E-08	1.5E-07	1 mile 0.6R	1 mile 1.2R
LDCA	10-02-01-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	3.8E-08	3.5E-08	1.5E-07	1 mile 0.6R	1 mile 1.2R
LDCA	10-02-02-01	Loss of a DC bus with no secondary heat removal and induced SGTR	1.8E-08	1.6E-08	7.2E-08	1 mile 100R	1 mile 356R
LDCA	10-02-03-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	4.1E-08	3.8E-08	1.7E-07	1 mile 0.6R	1 mile 1.2R
LDCA	10-02-04-01	Loss of a DC bus with no secondary heat removal and induced SGTR	1.5E-08	1.3E-08	5.8E-08	1 mile 100R	1 mile 356R
LLOCA	01	LLOCA with all systems successful	5.0E-06	5.1E-06	1.9E-05	<1mR	<1mR
LOCCW-A	01	Loss of Component Cooling with RCP seal failure	2.0E-04	2.0E-04	9.6E-04	<1mR	<1mR
LOCCW-A	02	Loss of Component Cooling with RCP seal failure	4.8E-07	4.4E-07	1.8E-06	<1mR	<1mR
LOCCW-A	07	Loss of Component Cooling with failure to cooldown	2.0E-07	2.0E-07	1.0E-06	<1mR	<1mR

Table E.6 Accident Sequences for Sequences with a Point Estimate > 1E-8

Initiating Event	Sequence	Description	Point Estimate (per year)	Mean (per year)	95th (per year)	Mean Dose (REM)	95th Dose (REM)
LOESW	01	Loss of Essential Reactor Cooling Water with RCPs remaining intact	4.0E-04	4.1E-04	1.92E-03	<1mR	<1mR
LOESW	02	Loss of Essential Reactor Cooling with RCP seal failure	7.6E-05	8.1E-05	4.1E-04	<1mR	<1mR
LOESW	03-01-01	Loss of Essential Reactor Cooling with RCP seal failure. Although ERCW is recovered, low pressure recirculation fails.	2.6E-08	2.9E-08	1.28E-07	1 mile 0.4R	1 mile 0.5R
LOESW	04-01-01	Loss of Essential Reactor Cooling with RCP Seal failure. Without cooling low pressure recirculation fails.	2.6E-05	2.5E-05	1.2E-04	EAB NA 1 mile 0.4R	EAB 7R 1 mile 0.5R
LOESW	06-01-01	Loss of Essential Reactor Cooling with RCP Seal failure. Low pressure injection fails, Essential Reactor Cooling is recovered but high pressure recirculation fails.	1.3E-08	1.5E-08	6.1E-08	1 mile 0.4R	1 mile 0.5R
LOESW	09	Loss of Essential Reactor Cooling with failure to cooldown	4.0E-07	3.9E-07	2.0E-06	<1mR	<1mR
LOESW	10	Loss of Essential Reactor Cooling with ERCW recovery and RCP seal failure	7.6E-08	7.8E-09	3.3E-07	<1mR	<1mR
LOESW	13-01-01	Loss of Essential Reactor Cooling with RCP Seal failure. RCS cooldown fails and cooling water is not recovered.	2.6E-08	2.5E-08	7.7E-08	1 mile 0.6R	1 mile 1.2R
LOOP	01	LOOP with all systems successful, 2 hour recovery, no inventory challenge	3.3E-02	3.3E-02	8.5E-02	<1mR	<1mR
LOOP	02-01	LOOP with RCP seal failure	1.6E-06	2.4E-06	9.4E-06	<1mR	<1mR
LOOP	02-02-01	LOOP with RCP seal failure	2.6E-07	2.6E-07	1.0E-06	<1mR	<1mR

Table E.6 Accident Sequences for Sequences with a Point Estimate > 1E-8

Initiating Event	Sequence	Description	Point Estimate (per year)	Mean (per year)	95th (per year)	Mean Dose (REM)	95th Dose (REM)
LOOP	02-03	LOOP with RCP seal failure	1.5E-07	1.1E-07	4.7E-07	<1mR	<1mR
LOOP	02-04-01-01	LOOP with RCP seal failure and failure of high pressure recirculation	1.0E-08	8.3E-09	2.4E-08	1 mile 0.4R	1 mile 0.5R
LOOP	02-06-01	LOOP, 2 hour recovery, inventory challenged (PORVs fail to close) and RCS depressurization to low pressure injection fails	1.3E-08	1.8E-08	6.7E-08	<1mR	<1mR
LOOP	03	LOOP, 2 hour recovery, inventory challenged (PORVs fail to close)	1.2E-07	1.7E-07	6.0E-07	<1mR	<1mR
LOOP	10	LOOP, 2 hr recovery fails, PORVs fail to close, high pressure recirc successful	7.2E-08	6.6E-08	2.6E-07	<1mR	<1mR
LOOP	17-01-01-01	LOOP with AFW failure	2.4E-08	2.6E-08	1.1E-07	1 mile 256R	1 mile 927R
LOOP	17-01-03-01	LOOP with AFW failure	2.1E-08	2.3E-08	9.3E-08	1 mile 256R	1 mile 927R
LOOP	17-01-06-01	LOOP with AFW failure	2.5E-08	2.7E-08	1.1E-07	1 mile 256R	1 mile 927R
LOOP	17-01-07-01	LOOP with AFW failure	2.2E-08	2.4E-08	1.0E-08	1 mile 256R	1 mile 927R
LOOP	17-03-01-01	LOOP with AFW failure	2.2E-08	2.4E-08	9.9E-08	1 mile 256R	1 mile 927R
LOOP	17-03-02	LOOP with AFW failure	1.0E-08	1.2E-08	4.8E-08	1 mile 256R	1 mile 927R
LOOP	17-03-03-01	LOOP with AFW failure	2.4E-08	2.6E-08	1.1E-07	1 mile 256R	1 mile 927R
LOOP	18-01	SBO with secondary heat removal, power recovery and RCP seal integrity maintained	9.8E-06	1.4E-05	5.5E-05	<1mR	<1mR
LOOP	18-02	SBO with secondary heat removal, power recovery and RCP seal integrity maintained	2.8E-06	3.9E-06	1.5E-05	<1mR	<1mR

Table E.6 Accident Sequences for Sequences with a Point Estimate > 1E-8

Initiating Event	Sequence	Description	Point Estimate (per year)	Mean (per year)	95th (per year)	Mean Dose (REM)	95th Dose (REM)
LOOP	18-03-05-01	SBO with battery depletion	2.8E-08	3.9E-08	1.5E-07	1 mile 376R	1 mile 1060R
LOOP	18-03-06-01	SBO with battery depletion	6.9E-07	9.7E-07	3.8E-06	1 mile 376R	1 mile 1060R
LOOP	18-03-10-01	SBO with battery depletion	2.8E-08	3.9E-08	1.5E-07	1 mile 376R	1 mile 1060R
LOOP	18-03-11-01	SBO with battery depletion	6.9E-07	9.7E-07	3.8E-06	1 mile 376R	1 mile 1060R
LOOP	18-04-01	SBO with secondary heat removal, RCP seal failure and power recovery	2.4E-06	2.2E-06	1.0E-05	<1mR	<1mR
LOOP	18-04-07-01-01	SBO with secondary heat removal, RCP seal failure and power recovery. Both high and low pressure injection fail.	1.9E-08	1.4E-08	4.7E-08	1 mile 376R	1 mile 1060R
LOOP	18-05	SBO with secondary heat removal, RCP seal failure and power recovery	7.1E-07	6.4E-07	2.5E-06	<1mR	<1mR
LOOP	18-06-06-01	SBO with secondary heat removal, RCP seal failure and no power recovery	1.7E-07	1.8E-07	7.0E-07	1 mile 376R	1 mile 1060R
LOOP	18-06-11-01	SBO with secondary heat removal, RCP seal failure and no power recovery	1.7E-07	1,8E-07	7.0E-07	1 mile 376R	1 mile 1060R
LOOP	18-07-01	SBO with secondary heat removal, RCP seal failure and power recovery	1.2E-07	1.7E-07	6.5E-07	<1mR	<1mR
LOOP	18-08	SBO with secondary heat removal, RCP seal failure and power recovery	3.5E-08	4.7E-08	1.8E-07	<1mR	<1mR
LOOP	18-10-01	SBO with secondary heat removal, RCP seal failure and power recovery	2.5E-08	2.1E-08	7.4E-08	<1mR	<1mR
LOOP	18-11	SBO with secondary heat removal, RCP seal failure and EDG recovery	1.5E-08	1.4E-08	4.1E-08	<1mR	<1mR

Table E.6 Accident Sequences for Sequences with a Point Estimate > 1E-8

Initiating Event	Sequence	Description	Point Estimate (per year)	Mean (per year)	95^{th} (per year)	Mean Dose (REM)	95^{th} Dose (REM)
LOOP	18-40-01	SBO with secondary heat removal, PORV fails to re-close and power recovery	1.6E-08	1.8E-08	7.4E-08	<1mR	<1mR
LOOP	18-41	SBO with secondary heat removal, PORV fails to re-close and EDG recovery	1.8E-08	2.8E-08	8.1E-08	<1mR	<1mR
LOOP	18-42-05-01	SBO with secondary heat removal, PORV fails to re-close, no power recovery, containment failure due to H2	1.5E-08	1.8E-08	6.5E-08	1 mile 376R	1 mile 1060R
LOOP	18-43-03-01-01-01	SBO with secondary heat removal, PORV fails to re-close, no power recovery, containment failure due to seal table	1.8E-08	2.6E-08	9.9E-08	1 mile 256R	1 mile 927R
LOOP	18-43-03-01-03-01	SBO without secondary heat removal	1.6E-08	2.2E-08	8.6E-08	1 mile 256R	1 mile 927R
LOOP	18-43-03-01-06-01	SBO with failure of secondary heat removal, RCP seal failure and no power recovery	1.8E-08	2.6E-08	1.0E-07	1 mile 256R	1 mile 927R
LOOP	18-43-03-01-07-01	SBO without secondary heat removal	1.7E-08	2.4E-08	9.2E-08	1 mile 256R	1 mile 927R
LOOP	18-43-03-03-03-01	SBO without secondary heat removal	1.8E-08	2.6E-08	1.0E-07	1 mile 256R	1 mile 927R
LOOP	18-44	SBO with failure of secondary heat removal, RCP seal failure and power recovery within 1 hour	1.4E-07	1.7E-07	6.5E-07	<1mR	<1mR
LOOP	18-45-01-06-01	SBO without secondary heat removal	1.6E-08	2.1E-08	8.6E-08	1 mile 256R	1 mile 927R
LOOP	18-45-01-13-01	SBO without secondary heat removal	1.4E-08	1.9E-08	7.5E-08	1 mile 256R	1 mile 927R

Table E.6 Accident Sequences for Sequences with a Point Estimate > 1E-8

Initiating Event	Sequence	Description	Point Estimate (per year)	Mean (per year)	95th (per year)	Mean Dose (REM)	95th Dose (REM)
LOOP	18-45-01-20-01	SBO without secondary heat removal	1.7E-08	2.2E-08	8.8E-08	1 mile 256R	1 mile 927R
LOOP	18-45-01-25-01	SBO without secondary heat removal	1.5E-08	2.0E-08	8.0E-08	1 mile 256R	1 mile 927R
LOOP	18-45-02-06-01	SBO without secondary heat removal	1.5E-08	2.0E-08	8.0E-08	1 mile 256R	1 mile 927R
LOOP	18-45-02-12-01	SBO without secondary heat removal	1.7E-08	2.2E-08	8.7E-08	1 mile 256R	1 mile 927R
LOOP	19-08	ATWS with all systems successful (MFW not credited)	4.0E-08	4.2E-08	1.5E-07	<1mR	<1mR
LOOP	19-09	ATWS with failure of PORVs to re-close (MFW not credited)	1.2E-08	1.2E-08	5.0E-08	<1mR	<1mR
MLOCA	01	MLOCA with all systems successful	4.0E-05	4.1E-05	1.5E-04	<1mR	<1mR
MLOCA	02-01-01	MLOCA with high pressure recirculation failure	1.0E-07	1.0E-07	4.3E-07	1 mile 0.6R	1 mile 1.2R
SGTR	01	SGTR with all systems successful	4.0E-03	4.0E-03	1.6E-02	<1mR	<1mR
SGTR	02	SGTR with failure to isolate the ruptured SG	4.8E-05	5.0E-05	2.4E-04	<1mR	<1mR
SGTR	03-01-01	SGTR with failure to isolate and failure of RHR	9.7E-08	9.5E-08	4.4E-07	1 mile 36R	1 mile 88R
SGTR	03-02-01	SGTR with failure to isolate and failure of RHR	1.2E-07	1.2E-07	5.4E-07	1 mile 36R	1 mile 88R
SGTR	04-01-01	SGTR with failure to depressurize to RHR entry condition	2.2E-08	2.0E-04	9.5E-08	1 mile 36R	1 mile 88R
SGTR	04-02-01	SGTR with failure to isolate and failure to depressurize to RHR entry conditions	2.6E-08	2.5E-08	1.1E-07	1 mile 36R	1 mile 88R
SGTR	05-01-01	SGTR with failure to depressurize < SG RV setpoints	5.1E-08	5.8E-08	2.4E-07	1 mile 36R	1 mile 88R

Table E.6 Accident Sequences for Sequences with a Point Estimate > 1E-8

Initiating Event	Sequence	Description	Point Estimate (per year)	Mean (per year)	95th (per year)	Mean Dose (REM)	95th Dose (REM)
SGTR	05-02-01	SGTR with failure to depressurize to < SG RV setpoints	6.2E-08	7.1E-08	2.9E-07	1 mile 36R	1 mile 88R
SGTR	06	SGTR with failure ro depressurize before SG reliefs lift	4.4E-05	4.4E-05	2.1E-04	<1mR	<1mR
SGTR	07	SGTR with failure to isolate the ruptured SG and failure to depressurize before SG reliefs lift	5.5E-07	5.5E-07	2.4E-06	<1mR	<1mR
SGTR	08-01-01	SGTR with failure to depressurize before SG reliefs lift, failure to isolate the rupture SG and failure or RHR	1,6E-07	1.7E-07	5.8E-07	1 mile 36R	1 mile 88R
SGTR	11-01-01	SGTR with failure to depressurize before and after SG reliefs lift	4.0E-07	3.85E-07	1.8E-06	1 mile 36R	1 mile 88R
SGTR	11-02-01	SGTR with failure to depressurize before and after SG reliefs lift	4.8E-07	4.7E-07	2.2E-06	1 mile 36R	1 mile 88R
SGTR	12	SGTR with failure of high pressure injection	1.5E-08	1.6E-08	6.9E-08	<1mR	<1mR
SGTR	43-01	SGTR with failure of secondary heat removal	4.1E-07	4.6E-07	1.9E-06	1 mile 105R	1 mile 366R
SLOCA	01	SLOCA with all systems successful	4.0E-04	4.1E-04	1.9E-03	<1mR	<1mR
SLOCA	02	SLOCA with the failure of RHR and successful high pressure recirculation	1.6E-06	1.6E-06	7.9E-06	<1mR	<1mR
SLOCA	04	SLOCA with failure of cooldown and high pressure recirculation	4.0E-07	3.9E-07	2.0E-06	<1mR	<1mR
SLOCA	03-01-01	SLOCA with the failure of RHR and high pressure recirculation	1.8E-07	1.9E-07	8.7E-07	1 mile 0.6R	1 mile 1.2R
TRANS	01	TRANS with all system successful	7.0E-01	7.0E-01	1.3	<1mR	<1mR

Table E.6 Accident Sequences for Sequences with a Point Estimate > 1E-8

Initiating Event	Sequence	Description	Point Estimate (per year)	Mean (per year)	95th (per year)	Mean Dose (REM)	95th Dose (REM)
TRANS	02	TRANS with failure PORVs to reseat	5.0E-07	4.3E-07	1.4E-06	<1mR	<1mR
TRANS	18-01-01-01	TRANS with failure of secondary heat removal	4.5E-07	4.7E-07	2.0E-06	1 mile 0.6R	1 mile 1.2R
TRANS	18-01-02-01	TRANS with failure of secondary heat removal & induced SGTR	1.1E-08	1.2E-08	5.0E-08	1 mile 100R	1 mile 356R
TRANS	18-01-03-01	TRANS with failure of secondary heat removal	3.9E-07	4.1E-07	1.7E-06	1 mile 0.6R	1 mile 1.2R
TRANS	18-01-04-01	TRANS with failure of secondary heat removal & induced SGTR	6.2E-08	6.9E-08	2.9E-07	1 mile 100R	1 mile 356R
TRANS	18-01-06-01	TRANS with failure of secondary heat removal	4.6E-07	4.8E-07	2.0E-06	1 mile 0.6R	1 mile 1.2R
TRANS	18-01-07-01	TRANS with failure of secondary heat removal	4.1E-07	4.4E-07	1.8E-06	1 mile 0.6R	1 mile 1.2R
TRANS	18-01-08-01	TRANS with failure of secondary heat removal & induced SGTR	3.7E-08	3.9E-08	1.6E-07	1 mile 100R	1 mile 356R
TRANS	18-02-01-01	TRANS with failure of secondary heat removal	4.1E-07	4.4E-07	1.8E-06	1 mile 0.6R	1 mile 1.2R
TRANS	18-02-02-01	TRANS with failure of secondary heat removal & induced SGTR	2.0E-07	2.2E-07	9.3E-07	1 mile 100R	1 mile 356R
TRANS	18-02-03-01	TRANS with failure of secondary heat removal	4.5E-07	4.7E-07	2.-E-06	1 mile 0.6R	1 mile 1.2R
TRANS	18-02-04-01	TRANS with failure of secondary heat removal & induced SGTR	1.6E-07	1.8E-07	7.5E-07	1 mile 100R	1 mile 356R
TRANS	19-08	ATWS with all systems successful (MFW not credited)	1.4E-06	1.4E-06	4.8E-06	<1mR	<1mR
TRANS	19-09	ATWS with stuck open PORVs	4.3E-07	4.2E-07	1.9E-06	<1mR	<1mR

Table E.6 Accident Sequences for Sequences with a Point Estimate > 1E-8

Initiating Event	Sequence	Description	Point Estimate (per year)	Mean (per year)	95^{th} (per year)	Mean Dose (REM)	95^{th} Dose (REM)
TRANS	19-14-01-01	ATWS with failure to emergency borate	2.9E-08	2.9E-08	1.3E-07	1 mile 0.4R	1 mile 0.5R
TRANS	19-16-01-01-01	ATWS with RCS pressure boundary failure	3.4E-08	3.4E-08	1.4E-07	1 mile 0.4R	1 mile 0.5R
TRANS	19-16-03-01-01	ATWS with RCS pressure boundary failure	2.2E-08	2.3E-08	9.0E-08	1 mile 0.4R	1 mile 0.5R
V-RHR-SUC	03	RHR Suction ISLOCA with successful mitigation	6.1E-07	4.0E-06	8.8E-06	<1mR	<1mR
V-RHR-SUC	04-01	RHR Suction ISLOCA with failure to isolate	1.2E-08	9.7E-08	1.4E-08	1 mile 998R	1 mile 3548R
V-RHR-SUC	05-01	RHR Suction ISLOCA with failure to diagnose	1.5E-07	9.9E-07	1.6E-06	1 mile 998R	1 mile 3548R

Step 4 Identify all PRA event sequences with a 95^{th} percentile frequency > 1E-7 per year.

This step identifies those sequences that are to be included in the event class grouping process. Sequences less than 1E-7 per year are screened from the process. The remaining in-scope sequences are those in Table E.6 that are not shaded.

Step 5 Group the PRA event sequences with a 95^{th} percentile frequency > 1E-7 per year into event classes.

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

Table E.7 PRA Sequences Grouping

Initiating Event	Sequence	Description	Event Class	Mean (per year)	95th (per year)	95th Dose (REM)
LDCA	01	Loss of a DC bus with all remaining systems successful	LBE-01	2.51E-03	1.0E-02	<1mR
LDCA	10-01-01-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	LBE-02	3.8E-08	1.6E-07	1.2R
LDCA	10-01-03-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	LBE-02	3.28E-08	1.4E-07	1.2R
LDCA	10-01-06-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	LBE-02	3.9E-08	1.7E-07	1.2R
LDCA	10-01-07-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	LBE-02	3.5E-08	1.5E-07	1.2R
LDCA	10-02-01-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	LBE-02	3.5E-08	1.5E-07	1.2R
LDCA	10-02-03-01	Loss of a DC bus with no secondary heat removal and no induced SGTR	LBE-02	3.8E-08	1.7E-07	1.2R
LLOCA	01	LLOCA with all systems successful	LBE-03	5.1E-06	1.9E-05	<1mR
LOCCW-A	01	Loss of Component Cooling with RCP seal failure	LBE-04	2.0E-04	9.6E-04	<1mR
LOCCW-A	02	Loss of Component Cooling with RCP seal failure	LBE-05	4.4E-07	1.8E-06	<1mR
LOCCW-A	07	Loss of Component Cooling with failure to cooldown	LBE-06	2.0E-07	1.0E-06	<1mR
LOESW	01	Loss os Essential Reactor Cooling Water with RCPs remaining intact	LBE-04	4.1E-04	1.92E-03	<1mR
LOESW	02	Loss of Essential Reactor Cooling with RCP seal failure	LBE-05	8.1E-05	4.1E-04	<1mR

Table E.7 PRA Sequences Grouping

Initiating Event	Sequence	Description	Event Class	Mean (per year)	95th (per year)	95th Dose (REM)
LOESW	03-01-01	Loss of Essential Reactor Cooling with RCP seal failure. Although ERCW is recovered, low pressure recirculation fails.	LBE-07	2.9E-08	1.28E-07	7R
LOESW	04-01-01	Loss of Essential Reactor Cooling with RCP Seal failure. Without cooling low pressure recirculation fails.	LBE-07	2.5E-05	1.2E-4	7R
LOESW	09	Loss of Essential Reactor Cooling with failure to cooldown	LBE-06	3.9E-07	2.0E-06	<1mR
LOESW	10	Loss of Essential Reactor Cooling with ERCW recovery and RCP seal failure	LBE-08	7.8E-09	3.3E-07	<1mR
LOOP	01	LOOP with all systems successful, 2 hour recovery, no inventory challenge	LBE-09	3.3E-02	8.5E-02	<1mR
LOOP	02-01	LOOP with RCP seal failure	LBE-10	2.4E-06	9.4E-06	<1mR
LOOP	02-02-01	LOOP with RCP seal failure	LBE-10	2.6E-07	1.0E-06	<1mR
LOOP	02-03	LOOP with RCP seal failure	LBE-10	1.1E-07	4.7E-07	<1mR
LOOP	03	LOOP, 2 hour recovery, inventory challenged (PORVs fail to close)	LBE-11	1.7E-07	6.0E-07	<1mR
LOOP	10	LOOP, 2 hr recovery fails, PORVs fail to close, high pressure recirc successful	LBE-11	6.6E-08	2.6E-07	<1mR
LOOP	17-01-01-01	LOOP with AFW failure	LBE-12	2.6E-08	1.1E-07	927R
LOOP	17-01-06-01	LOOP with AFW failure	LBE-12	2.7E-08	1.1E-07	927R
LOOP	17-03-03-01	LOOP with AFW failure	LBE-12	2.6E-08	1.1E-07	927R
LOOP	18-01	SBO with secondary heat removal, power recovery and RCP seal integrity maintained	LBE-13	1.4E-05	5.5E-05	<1mR
LOOP	18-02	SBO with secondary heat removal, power recovery and RCP seal integrity maintained	LBE-13	3.9E-06	1.5E-05	<1mR

Table E.7 PRA Sequences Grouping

Initiating Event	Sequence	Description	Event Class	Mean (per year)	95th (per year)	95th Dose (REM)
LOOP	18-03-05-01	SBO with battery depletion	LBE-14	3.9E-08	1.5E-07	1060R
LOOP	18-03-06-01	SBO with battery depletion	LBE-14	9.7E-07	3.8E-06	1060R
LOOP	18-03-10-01	SBO with battery depletion	LBE-14	3.9E-08	1.5E-07	1060R
LOOP	18-03-11-01	SBO with battery depletion	LBE-14	9.7E-07	3.8E-06	1060R
LOOP	18-04-01	SBO with secondary heat removal, RCP seal failure and power recovery	LBE-15	2.2E-06	1.0E-05	<1mR
LOOP	18-05	SBO with secondary heat removal, RCP seal failure and power recovery	LBE-15	6.4E-07	2.5E-06	<1mR
LOOP	18-06-06-01	SBO with secondary heat removal, RCP seal failure and no power recovery	LBE-16	1.8E-07	7.0E-07	1060R
LOOP	18-06-11-01	SBO with secondary heat removal, RCP seal failure and no power recovery	LBE-16	1,8E-07	7.0E-07	1060R
LOOP	18-07-01	SBO with secondary heat removal, RCP seal failure and power recovery	LBE-15	1.7E-07	6.5E-07	<1mR
LOOP	18-08	SBO with secondary heat removal, RCP seal failure and power recovery	LBE-15	4.7E-08	1.8E-07	<1mR
LOOP	18-43-03-01-06-01	SBO with failure of secondary heat removal, RCP seal failure and no power recovery	LBE-16	2.6E-08	1.0E-07	927R
LOOP	18-44	SBO with failure of secondary heat removal, RCP seal failure and power recovery within 1 hour	LBE-17	1.7E-07	6.5E-07	<1mR
LOOP	19-08	ATWS with all systems successful (MFW not credited)	LBE-18	4.2E-08	1.5E-07	<1mR
MLOCA	01	MLOCA with all systems successful	LBE-19	4.1E-05	1.5E-04	<1mR

Table E.7 PRA Sequences Grouping

Initiating Event	Sequence	Description	Event Class	Mean (per year)	95th (per year)	95th Dose (REM)
MLOCA	02-01-01	MLOCA with high pressure recirculation failure	LBE-20	1.0E-07	4.3E-07	1.2R
SGTR	01	SGTR with all systems successful	LBE-21	4.0E-03	1.6E-02	<1mR
SGTR	02	SGTR with failure ro isolate the ruptured SG	LBE-22	5.0E-05	2.4E-04	<1mR
SGTR	03-01-01	SGTR with failure to isolate and failure of RHR	LBE-23	9.5E-08	4.4E-07	88R
SGTR	03-02-01	SGTR with failure to isolate and failure of RHR	LBE-23	1.2E-07	5.4E-07	88R
SGTR	04-02-01	SGTR with failure to isolate and failure to depressurize to RHR entry conditions	LBE-23	2.5E-08	1.1E-07	88R
SGTR	05-01-01	SGTR with failure to depressurize < SG RV setpoints	LBE-24	5.8E-08	2.4E-07	88R
SGTR	05-02-01	SGTR with failure to depressurize to < SG RV setpoints	LBE-24	7.1E-08	2.9E-07	88R
SGTR	06	SGTR with failure ro depressurize before SG reliefs lift	LBE-25	4.4E-05	2.1E-04	<1mR
SGTR	07	SGTR with failure to isolate the ruptured SG and failure to depressurize before SG reliefs lift	LBE-25	5.5E-07	2.4E-06	<1mR
SGTR	08-01-01	SGTR with failure to depressurize before SG reliefs lift, failure to isolate the rupture SG and failure or RHR	LBE-24	1.7E-07	5.8E-07	88R
SGTR	11-01-01	SGTR with failure to depressurize before and after SG reliefs lift	LBE-24	3.85E-07	1.8E-06	88R
SGTR	11-02-01	SGTR with failure to depressurize before and after SG reliefs lift	LBE-24	4.7E-07	2.2E-06	88R
SGTR	43-01	SGTR with failure of secondary heat removal	LBE-26	4.6E-07	1.9E-06	366R
SLOCA	01	SLOCA with all systems successful	LBE-27	4.1E-04	1.9E-07	<1mR

Table E.7 PRA Sequences Grouping

Initiating Event	Sequence	Description	Event Class	Mean (per year)	95th (per year)	95th Dose (REM)
SLOCA	02	SLOCA with the failure of RHR and successful high pressure recirculation	LBE-28	1.6E-06	7.9E-06	<1mR
SLOCA	04	SLOCA with failure of cooldown and high pressure recirculation	LBE-28	3.9E-07	2.0E-06	<1mR
SLOCA	03-01-01	SLOCA with the failure of RHR and high pressure recirculation	LBE-29	1.9E-07	8.7E-07	1.2R
TRANS	01	TRANS with all system successful	LBE-30	7.0E-01	1.3	<1mR
TRANS	02	TRANS with failure PORVs to reseat	LBE-27	4.3E-07	1.4E-06	<1mR
TRANS	18-01-01-01	TRANS with failure of secondary heat removal	LBE-31	4.7E-07	2.0E-06	1.2R
TRANS	18-01-03-01	TRANS with failure of secondary heat removal	LBE-31	4.1E-07	1.7E-06	1.2R
TRANS	18-01-04-01	TRANS with failure of secondary heat removal and induced SGTR	LBE-29	6.9E-08	2.9E-07	356R
TRANS	18-01-06-01	TRANS with failure of secondary heat removal	LBE-31	4.8E-07	2.0E-06	1.2R
TRANS	18-01-07-01	TRANS with failure of secondary heat removal	LBE-31	4.4E-07	1.8E-06	1.2R
TRANS	18-01-08-01	TRANS with failure of secondary heat removal and induced SGTR	LBE-29	3.9E-08	1.6E-07	356R
TRANS	18-02-01-01	TRANS with failure of secondary heat removal	LBE-31	4.4E-07	1.8E-06	1.2R
TRANS	18-02-02-01	TRANS with failure of secondary heat removal and induced SGTR	LBE-29	2.2E-07	9.3E-07	356R
TRANS	18-02-03-01	TRANS with failure of secondary heat removal	LBE-31	4.7E-07	2.-E-06	1.2R
TRANS	18-02-04-01	TRANS with failure of secondary heat removal and induced SGTR	LBE-29	1.8E-07	7.5E-07	356R
TRANS	19-08	ATWS with all systems successful (MFW not credited)	LBE-18	1.4E-06	4.8E-06	<1mR
TRANS	19-09	ATWS with stuck open PORVs	LBE-18	4.2E-07	1.9E-06	<1mR

Table E.7 PRA Sequences Grouping

Initiating Event	Sequence	Description	Event Class	Mean (per year)	95th (per year)	95th Dose (REM)
TRANS	19-14-01-01	ATWS with failure to emergency borate	LBE-32	2.9E-08	1.3E-07	0.5R
TRANS	19-16-01-01-01	ATWS with RCS pressure boundary failure	LBE-32	3.4E-08	1.4E-07	0.5R
V-RHR-SUC	03	Rhr Suction ISLOCa with successful mitigation	LBE-33	4.0E-06	8.8E-06	<1mR
V-RHR-SUC	04-01	RHR Suction ISLOCA with failure to isolate	LBE-34	9.7E-08	1.4E-08	3548R
V-RHR-SUC	05-01	RHR Suction ISLOCA with failure to diagnose	LBE-34	9.9E-07	1.6E-06	3548R

Additional discussion of the grouping process can be found in Step 6 following Table E.8.

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

The selected event sequence defines the accident behavior and consequences for the LBE that represent this event class. If several events within the event class have similar consequences, then a bounding event is selected. If there is not a clear bounding event, then the event with the lowest frequency is selected. Note that the frequency of the event class is determined separately from the bounding consequence event. See Step 7. Table E.8 lists the resulting bounding events for the example PWR.

Table E.8 Licensing Basis Events

LBE	Description	Frequency Bases	Consequence Bases	Mean (per year)	95th (per year)	Category	95th Dose
LBE-01	Loss of a DC Bus with all remaining systems successful	LDCA 01	LDCA 01 (1 Event)	2.5E-03	1.0E-02	Frequent	<1mR
LBE-02	Loss of DC with no secondary heat removal, early secondary depressurization and no induced SGTR	LDCA 10-01-06-01	LDCA 10-01-03-01 (6 Events)	3.9E-08	1.7E-07	Rare	1.2R
LBE-03	LLOCA with all systems successful	LLOCA 01	LLOCA 01 (1 Event)	5.1E-06	1.9E-05	Infrequent	<1mR
LBE-04	Loss of Essential Reactor Cooling Water with RCPs intact	LOESW 01	LOESW 01 (2 Events)	4.1E-04	1.9E-03	Infrequent	<1mR
LBE-05	Loss of Essential Reactor Cooling Water with RCP seal failure	LOESW 02	LOESW 02 (2 Events)	8.1E-05	4.1E-04	Infrequent	<1mR

Table E.8 Licensing Basis Events

LBE	Description	Frequency Bases	Consequence Bases	Mean (per year)	95th (per year)	Category	95th Dose
LBE-06	Loss of Essential Reactor Cooling Water with failure to cooldown	LOESW 09	LOESW 09 (2 Events)	3.9E-07	2.0E-06	Rare	<1mR
LBE-07	Loss of Essential Reactor Cooling Water with RCP seal failure and low pressure recirculation failure	LOESW 04-01-01	LOESW 04-01-01 (1 Event)	2.5E-05	1.2E-04	Infrequent	7R
LBE-08	Loss of Essential Reactor Cooling Water with recovery and RCP seal failure	LOESW 10	LOESW 10 (1 Event)	7.8E-08	3.3E-07	Rare	<1mR
LBE-09	LOOP with all systems successful , 2 hr recovery no inventory challenge	LOOP 01	LOOP 01 (1 Event)	3.3E-02	8.5E-02	Frequent	<1mR
LBE-10	LOOP with RCP seal failure (Bounding LOOP: stage 2 seal failure and Loop recovery fails)	LOOP 02-01	LOOP 02-03 (3 Events)	2.4E-06	9.4E-06	Rare	<1mR
LBE-11	LOOP, 2 hr recovery fails, PORVs fail to close, high pressure recirculation successful	LOOP 03	LOOP 10 (2 Events)	1.7E-07	6.0E-07	Rare	<1mR
LBE-12	LOOP with AFW failure (Bounding LOOP: RCP seals intact, early SG depressurization)	LOOP 17-01-06-01	LOOP 17-03-03-01 (3 Events)	2.7E-08	1.1E-07	Rare	927R
LBE-13	SBO with secondary heat removal, power recovery and RCP seal integrity maintained	LOOP 18-01	LOOP 18-01 (2 Events)	1.4E-05	5.5E-05	Infrequent	<1mR
LBE-14	SBO with battery depletion (Bounding LOOP: no RCS depressurization, vessel breach)	LOOP 18-03-06-01	LOOP 18-03-10-01 (4 Events)	9.7E-07	3.8E-06	Rare	1060R
LBE-15	SBO with secondary heat removal, RCP seal failure and power recovery	LOOP 18-04-01	LOOP 18-04-01 (4 Events)	2.2E-06	1.0E-05	Infrequent	<1mR
LBE-16	SBO with secondary heat removal, RCP seal failure and no power recovery (Bounding: no RCS depressurization, RCP Stage 2 failure)	LOOP 18-06-11-01	LOOP 18-06-11-01 (3 Events)	1.8E-07	7.0E-07	Rare	1060R
LBE-17	SBO with failure of secondary heat removal, RCP seal failure and power recovery within 1 hour	LOOP 18-44	LOOP 18-44 (1 Event)	1.7E-07	6.5E-07	Rare	<1mR
LBE-18	ATWS with all systems successful (MFW not credited)	TRANS 19-08	TRANS 19-08 (3 Events)	1.4E-06	4.8E-06	Rare	<1mR
LBE-19	MLOCA with all systems successful	MLOCA 01	MLOCA 01 (1 Event)	4.1E-05	1.5E-04	Infrequent	<1mR

Table E.8 Licensing Basis Events

LBE	Description	Frequency Bases	Consequence Bases	Mean (per year)	95th (per year)	Category	95th Dose
LBE-20	MLOCA with high pressure recirculation failure	MLOCA 02-01-01	MLOCA 02-01-01 (1 Event)	1,1E-07	4.3E-07	Rare	1.2R
LBE-21	SGTR with all systems successful	SGTR 01	SGTR 01 (1 Event)	4.0E-03	1.6E-02	Frequent	<1mR
LBE-22	SGTR with failure to isolate the ruptured SG	SGTR 02	SGTR 02 (1 Event)	5.0E-05	2.4E-04	Infrequent	<1mR
LBE-23	SGTR with failure to isolate and failure of RHR	SGTR 03-02-01	SGTR 03-02-01 (3 Events)	1.2E-07	5.4E-07	Rare	88R
LBE-24	SGTR with failure to depressurize before SG reliefs lift, failure to isolate the ruptured SG and failure of RHR	SGTR 11-02-01	SGTR 08-01-01 (5 Events)	4.7E-07	2.2E-06	Rare	88R
LBE-25	SGTR with failure to depressurize before SG reliefs lift	SGTR 06	SGTR 06 (2 Events)	4.4E-05	2.1E-04	Infrequent	<1mR
LBE-26	SGTR with failure of secondary heat removal	SGTR 43-01	SGTR 43-01 (1 Event)	4.6E-07	1.9E-06	Rare	366R
LBE-27	SLOCA with all systems successful	SLOCA 01	SLOCA 01 (2 Events)	4.1E-04	1.9E-03	Infrequent	<1mR
LBE-28	SLOCA with the failure of RHR and successful high pressure recirculation (Bounding event: failure of HP recirculation)	SLOCA 02	SLOCA 03-01-01 (2 Events)	1.6E-06	7.9E-06	Rare	<1mR
LBE-29	Transient with failure of secondary heat removal and induced SGTR	TRANS 18-02-02-01	TRANS 18-02-02-01 (5 Events)	2.2E-07	9.3E-07	Rare	356R
LBE-30	Transient with all systems successful	TRANS 01	TRANS 01 (1 Event)	6.7E-01	1.2	Frequent	<1mR
LBE-31	Transient with failure of secondary heat removal (Bounding: SG depressurization with induced SGTR)	TRANS 18-01-06-01	TRANS 18-01-03-01 (6 Events)	4.8E-07	2.0E-06	Rare	1.2R
LBE-32	ATWS with RCS pressure boundary failure (Bounding: ATWS with failure to emergency borate)	TRANS 19-16-01-01-01	TRANS 19-14-01-01 (2 Events)	3.4E-08	1.4E-07	Rare	0.5R
LBE-33	RHR Suction ISLOCA with successful mitigation	V-RHR-SUC 03	Y-RHR-SUC 03 (1 Event)	3.8E-06	8.8E-06	Rare	<1mR
LBE-34	RHR Suction ISLOCA with failure to diagnose	V-RHR-SUC 05-01	V-RHR-SUC 05-01 (2 Events)	9.9E-07	1.5E-06	Rare	3548R

As can be seen from Table E.8, 34 LBEs have been identified with each representing between one and six event sequences. Twelve LBEs address only a single event sequence. For the remaining 22 sequences, a bounding event was selected to represent the event class.

A discussion on LBE-02 is provided in order to illustrate the selection and grouping process. LBE-02 represents 6 events sequences, each initiated by the loss of a DC bus followed by the failure of auxiliary feedwater. Although feed and bleed is available at the example plant, this function was set to guaranteed failure, as it is not safety-related. For all six events, containment isolation remains intact and an induced steam generator tube rupture is avoided. The six events are differentiated by the status of RCS and secondary system pressure. For the four sequence 10-01 events, the steam generators are initially maintained at normal pressure. For the two sequence 10-02 events, early secondary system depressurization occurs. The additional variations of these sequences is associated with late depressurization of the RCS and secondary systems. The variations are shown in Table E.9.

Table E.9 LBE-02 Bounding Event Selection

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

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

It should also be noted that event grouping does not have to be limited to sequences with the same initiating event. LBE-18 is an example of an event class that crosses between initiating events. LBE-18 represents three anticipated transient without scram (ATWS) events. One of the events is initiated as a result of a loss of offsite power event with the resulting failure of the control rods to insert into the reactor core. The other two sequences are initiated by a transient. These events are shown in Table E.10

Table E.10 LBE-18 Bounding Event Selection

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

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

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

The frequency of an event class is determined by setting the LBE's mean frequency to the highest mean frequency of the event sequences in the event class and its 95th percentile frequency to the highest 95th percentile frequency of the event sequences in the event class. Note that the mean and 95th percentile frequencies can come from different event sequences. The example results are shown in Table E.8. In the example, the mean and 95th percentile frequency for each LBE come from the same event sequence.

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

The defense-in-depth requirements are a function of the frequency ranges. This binning is required in order to determine the LBE deterministic requirements. These frequency ranges and their associated requirements are shown in Table E.1. Table E.8 shows the results of this binning process.

Step 9 Verify that the selected LBEs meet the probabilistic and deterministic probabilistic requirements.

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

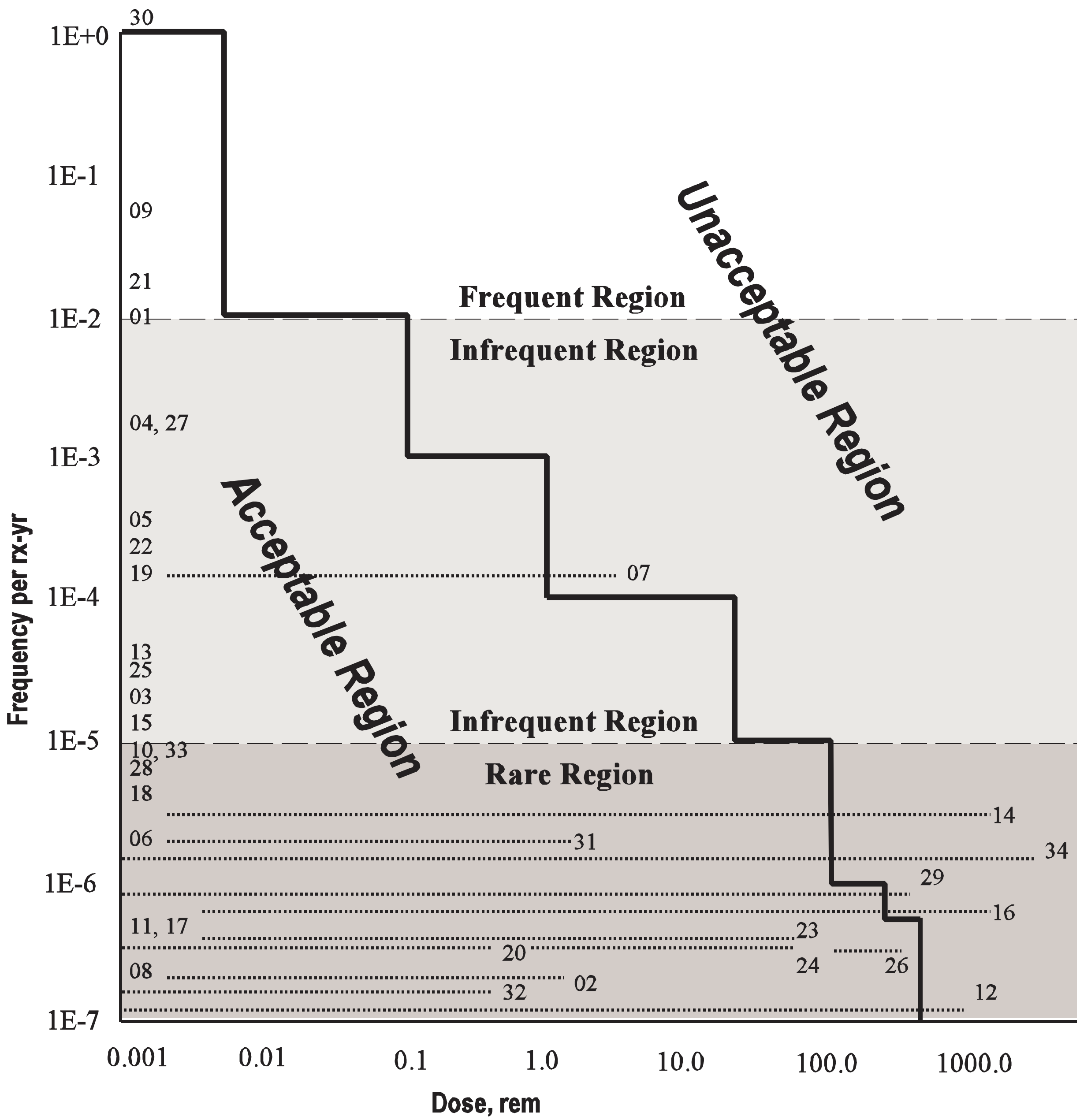

Figure E.1 Frequency-Consequence Curve with 95th Percentile Values

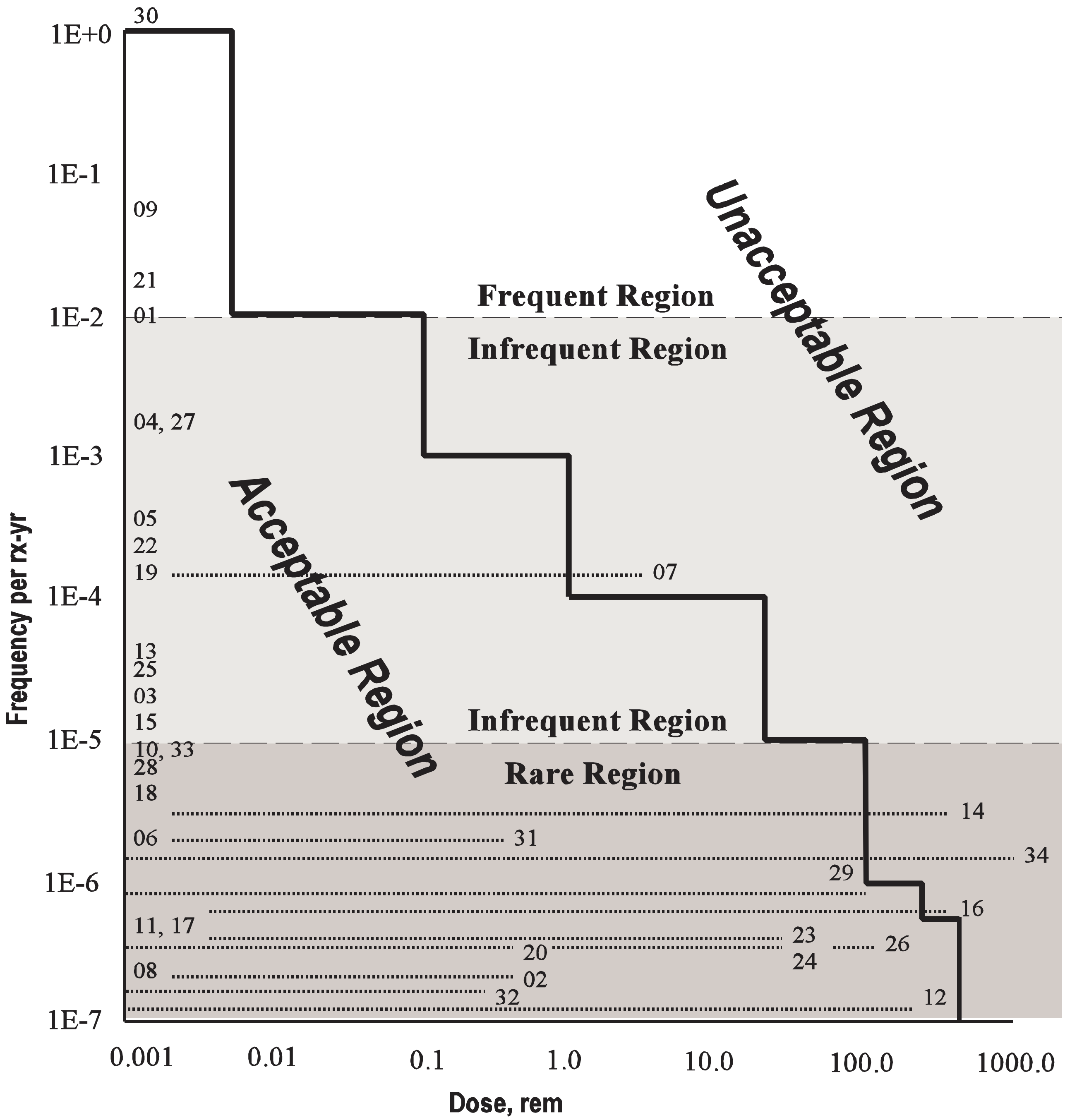

Figure E-2 Frequency -Consequence Curve with Mean Values

The framework has additional deterministic requirements for LBEs classified as Frequent or Infrequent. The example in this appendix has four Frequent Events, ten Infrequent LBEs and twenty Rare LBEs. Tables E.11 and E.12 show the deterministic requirements for Frequent and Infrequent LBEs, respectively, and show how the example's LBEs compare with the deterministic requirement.

Table E.11 Deterministic Requirements for LBEs Categorized as Frequent

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

Table E.12 Deterministic Requirements for LBEs Categorized as InFrequent

LBE	Description	At Least One Barrier Remains	Coolable Geometry Remains	Dose Meets F-C Curve	Comments
LBE-03	LLOCA with all systems successful	MEETS	MEETS	MEETS	
LBE-04	Loss of Essential Reactor Cooling Water with RCPs intact	MEETS	MEETS	MEETS	
LBE-05	Loss of Essential Reactor Cooling Water with RCP seal failure	MEETS	MEETS	MEETS	
LBE-07	Loss of Essential Reactor Cooling Water with RCP seal failure and low pressure recirculation failure	MEETS	DOES NOT MEET	DOES NOT MEET	This event sequence results in core damage and exceeds the F-C curve. The RCS barrier is breached due to RCP seal failure and fuel cladding barrier fails due to failure of low pressure recirculation. Containment isolation is achieve and maintained.
LBE-13	SBO with secondary heat removal, power recovery and RCP seal integrity maintained	MEETS	MEETS	MEETS	
LBE-15	SBO with secondary heat removal, RCP seal failure and power recovery	MEETS	MEETS	MEETS	
LBE-19	MLOCA with all systems successful	MEETS	MEETS	MEETS	

Table E.12 Deterministic Requirements for LBEs Categorized as InFrequent

LBE	Description	At Least One Barrier Remains	Coolable Geometry Remains	Dose Meets F-C Curve	Comments
LBE-22	SGTR with failure to isolate the ruptured SG	MEETS	MEETS	MEETS	
LBE-25	SGTR with failure to depressurize before SG reliefs lift	MEETS	MEETS	MEETS	
LBE-27	SLOCA with all systems successful	MEETS	MEETS	MEETS	

E.5 Comparison with Current Design Bases Events

E.5.1 Design Bases Events for Example Plant

This section describes the conditions or design basis events (DBEs) analyzed in the example plant's FSAR Chapter 15 analysis. The development of these original DBEs is consistent with Regulatory 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." The following five conditions, shown in Table E.13, were analyzed in the example plant's FSAR:

Table E.13 DBE Condition Categories

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

Table E.13 DBE Condition Categories

Condition	Title	Description
E	Environmental Faults	Faults that provide the limiting events for environmental consequences of an event.

Table E.14 lists the Condition II, III, IV and E events. Condition I events are normal operation and operational transients (e.g., power operation, start up, hot shutdown, cold shutdown, refueling). As stated in the example plant's FSAR, Condition I occurrences occur frequently or regularly, and they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III, and IV). In this regard, analysis of each fault condition described in Table E.14 is generally based on a conservative set of initial conditions corresponding to the most adverse set of conditions which can occur during Condition I operation. An explicit evaluation of each Condition I event is not provided in the FSAR.

Table E.14 Example PWR Chapter 15 Events

Event	Title	Description	Cat
1.1	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition	A rod cluster control assembly withdrawal of rod cluster control assemblies resulting in a power excursion. Such a transient could be caused by a malfunction of the reactor control or control rod drive systems. This is the maximum rate of reactivity addition (greater than the boron dilution event).	II
1.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power	Same as D.1.1, except at-power.	II
1.3	Rod Cluster Control Assembly Misalignment	Rod cluster control assembly misalignment includes: a dropped full-length assembly, a dropped full-length assembly bank, and statically misaligned full length assembly.	II
1.4	Uncontrolled Boron Dilution	The Chemical and volume Control System (CVCS) is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.	II
1.5	Partial Loss of Forced Reactor Coolant Loop	A partial loss of coolant flow accident can result from a mechanical or electrical failure in a reactor coolant pump, or from a fault in the power supply to the pump. If the reactor is at-power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. The necessary protection against a partial loss of coolant flow is provided by the low primary coolant flow reactor trip, which is actuated by two out of three low flow signals in any reactor coolant loop.	II
1.6	Startup of an Inactive Reactor Coolant Loop	Starting of an idle reactor coolant pump without bringing the inactive loop hot leg temperature close to the core inlet temperature would result in the injection of cold water into the core, which causes a rapid reactivity insertion and subsequent power increase.	II

Table E.14 Example PWR Chapter 15 Events

Event	Title	Description	Cat
1.7	Loss of External Electrical Load and/or Turbine Trip	Major load loss on the plant can result from loss of external electrical load or from a turbine trip. For either case, off-site power remains available for the continued operation of plant components, such as reactor coolant pumps. The case of loss of all AC power (station blackout) is analyzed in section D.1.9. Following the loss of generator load, an immediate fast closure of the turbine control valves will occur. For a turbine trip, the reactor would be tripped directly (unless below approximately 50% power) from a signal derived from the turbine autostop oil pressure and turbine stop valves.	II
1.8	Loss of Normal Feedwater	Event assumes that the reactor trips on low-low level in any steam generator and that only one motor driven auxiliary feedwater pump is available one minute after the low-low steam generator level signal is initiated. Secondary system steam relief is achieved through the self-actuated safety valves.	II
1.9	Loss of All Off-Site Power to the Station Auxiliaries	Event assumes that only one motor-driven auxiliary feedwater pump is available one minute after the low-low steam generator level signal is initiated in any steam generator.	II
1.10	Excessive Heat Removal Due to Feedwater System Malfunctions	Excessive feedwater flow could be caused by a full opening of one or more feedwater regulator valves due to a feedwater control system malfunction or an operator error. The feedwater flow from a fully open regulator valve is terminated by the steam generator high-high signal, which closes all feedwater regulator valves and feedwater isolation valves and trips the main feedwater pumps.	II
1.11	Excessive Load Increase Incident	This accident could result from either an administrative violation, such as excessive loading by the operator. or an equipment malfunction in the steam dump control or turbine speed control.	II
1.12	Accidental Depressurization of the Reactor Coolant System (inadvertent opening of pressurizer spray valve)	The most severe core condition resulting from an accidental depressurization of the RCS is associated with an inadvertent opening of a pressurizer safety valve. The reactor will be tripped by one of the following RPS signals: 1) pressurizer low pressure, or 2) overtemperature ^{a}T.	II
1.13	Accidental Depressurization of Main Steam System (inadvertent opening of a single dump, relief or safety valve)	The most severe core condition resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening of a single steam dump, relief or safety valve. The following systems provide the necessary protection against an accidental depressurization of the main steam system: 1) safety injection system actuation, 2) the overpower reactor trip, and 3) redundant isolation of the main feedwater lines.	II

Table E.14 Example PWR Chapter 15 Events

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

Table E.14 Example PWR Chapter 15 Events

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

Table E.14 Example PWR Chapter 15 Events

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

Table E.14 Example PWR Chapter 15 Events

Event	Title	Description	Cat
4.7	Environmental Consequences of a Postulated Rod Ejection Accident	Bounded by Loss of Coolant Accident.	E

E.5.2 Comparison of DBEs and LBEs

The DBEs frequency categories can be loosely compared with the framework's categories as shown in the Table E.15.

Table E.15 DBE and LBE Categories

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

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

E.5.2.1 Comparison of Events by Category

Moderate Frequency (Category II)/Frequent Category

In the (moderate) frequency category, the events identified by the two methods are similar. As shown in Table E.16, many of the FSAR events are mapped to the framework's transient initiating event indicating the need for this event to be bounding for all the initiators that are grouped into the transient initiating event category. One event, DB Event 1.12, appears to best map to the infrequent framework event of small LOCA (Sequence SLOCA 01). Two framework events, a steam generator tube rupture (Sequence SGTR 01) and the loss of a DC Bus (LDCA-01) are not included as frequent events in the FSAR.

Table E.16 Moderate Frequency (Category II) Event Comparison

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

Infrequent Category (Category III)

Table E.17 shows the Category III events. There are significant differences between the approaches in this category. First, the framework example includes small, medium and large LOCA event sequences in this category. For all three initiating events, no degradation of the mitigating systems is assumed (for these events in this category). Small LOCA with failure of residual heat removal is included in the rare event category. The SPAR model that is the bases for the framework example does not include main steam line breaks due to the limited contribution these initiators typically have on overall plant risk. It is expected that a fully developed framework PRA would have these steam line break initiators. Table E.17 provides a list of Category III events with the related LBE.

Table E.17 Infrequent (Category III) Event Comparison

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

Limiting Fault (Category IV)/Rare

There are six limiting fault DBEs identified in example plant's FSAR as shown in Table E.18. One is shutdown related and not addressed by the current selection of at-power LBEs. Both the large break LOCA and main steam line breaks are identified as limiting fault DBEs with the large break LOCA being identified as the limiting event for containment design and siting. In framework's selection process, only one large break LOCA scenario was identified. Unlike the DBE which considers a simultaneous LOOP and LOCA with a single failure, the large break LOCA LBE does not consider the occurrence of a LOOP event and has all safety functions available.

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

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

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

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

Table E.18 Infrequent (Category IV) Event Comparison

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

Environmental Consequences of Accidents

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

Table E.19 provides a list of environmental events with the related LBE.

Table E.19 Environmental Consequences Event Comparison

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

E.6 Conclusion

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

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

When the results of the framework events are compared against the framework's acceptance criteria, six LBEs are identified as exceeding the F-C curve when using the 95th percentile for both frequency and consequence, and two events are identified as not meeting the deterministic requirements. Considering the exclusion of some rare DBA event combinations and a more restrictive performance criteria for the 6 of 34 LBEs that do not satisfy the requirements of the F-C curve (and considering the addition of the framework's deterministic event as described in Chapter 6), the level of safety achieved by the framework selection process and associated acceptance criteria appears to be commensurate with that required for current plants. Some rare event sequences are excluded while other more frequent events are included.

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

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.

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

This appendix builds on existing PRA quality requirements delineated in Regulatory Guide 1.200 and the currently available PRA standards. The high-level requirements provided in these documents were reviewed and modified to make them generic for different reactor types, modes of operation, accident initiators, and other radioactive sources besides the reactor core. In addition, some of the requirements were generalized to address different accident end states and associated risk metrics. The supporting requirements in the PRA standards were also reviewed and in some cases, the content of a supporting requirement was deemed to contain an important requirement, not specifically addressed in other high-level requirements, that justified its elevation to a high-level requirement.

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. The licensee may choose to perform a fully integrated PRA that examines all accident initiators or perform separate PRAs for internal and external events. In either case, the identified PRA requirements are applicable.

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.

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.

Table F-2 Technical elements of a PRA.

Level of Analysis	Technical Element

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

The third level, Consequence Assessment, analyses the transport of radioactive material through the environment and assesses the health and economic consequences resulting from accidents. An analysis that includes all three levels described in Table F-2 allows for the assessment of risk since it provides both the frequency and consequence of potential accident sequences. For existing LWRs, a PRA of the reactor core that includes the Accident Sequence Development, Release Analysis, and Consequence Assessment technical elements is referred to as a Level 3 PRA.

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

F.3 Accident Sequence Development Technical Elements

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

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

Initiating event analysis identifies and characterizes the initiating events that can upset plant stability and challenge critical safety functions during all plant operating states (i.e., full-power, shutdown, and transitional states). 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. These requirements are applicable for both internal and external events.

Table F-3 Initiating event requirements.

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

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

Success criteria analysis is used to distinguish the path between success and failure for components, human actions, trains, systems, structures and sequences given an initiating event. In all cases, the success criteria 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. They are applicable to success criteria evaluations required for the analysis of internal and external initiators.

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

Table F-4 Success criteria 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.

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

If, as delineated in this framework, accident sequences will be used to define the LBEs and determine the safety significance of systems, the accident sequences delineated will be more than those that result in either a mitigated state or severe core damage as is currently done in LWR PRAs. Sequences resulting in intermediate states of core damage and/or levels of radioactive release will also have to be delineated and quantified. The delineation of these sequences may require that different levels of system success criteria be defined and delineated as separate events in the PRA models. An important 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).

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.

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. The requirements are applicable for the analysis of systems required to mitigate either internal and external initiating events.

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

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

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

Table F-7 Human reliability analysis 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.

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 required in the analysis of all accident initiators are shown in Table F-8.

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.

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.

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

Table F-9 Accident sequence quantification 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.

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

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

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

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-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 for each POS (i.e., the set of knowledge regarding the flood area, source, flood rate and capacity, operator actions, and SSC damage) that accounts for flood propagation, flood mitigation systems, and operator actions, and identifies susceptible SSCs.

FSE-5	Temporary configurations of barriers during different POSs that affect flood propagation and mitigation are included in the development of flood scenarios for each POS.
FSE-6	Screen out potential flood areas using acceptable criteria (e.g., none of the flood scenarios can cause a reactor trip or affects accident mitigating systems).

F.3.3 Internal Fire PRA

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

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

Table F-12 Flood sequence quantification 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.

FSQ-7	Identify significant contributors (including assumptions, initiating events, POSs, accident sequences, component failures, and human errors) to the required end-states and ensure that all flood significant sequences are traceable and reproducible.
FSQ-8	Characterize and quantify the uncertainties in the results including parameter and model uncertainty and the contribution from assumptions. Understand their potential impact on the results.

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

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

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

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

The models used to analyze fire growth, fire suppression, and fire-induced component and barrier damage must be consistent with actual nuclear power plant fire experience, tests, and experiments. Data used in the analyses should reflect plant-specific experience to the extent practical. The high level 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).
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	Follow the Systems Analysis requirements and include plant-specific experience and reflect scenario-specific conditions in the modeling of fire suppression systems. Address the dependency between various forms of automatic and manual suppression and account for fire-effects on manual suppression.
FD-7	Fire models and data used in the fire damage analysis are consistent with actual fire experience (when available) and experiments.
FD-8	Temporary configurations of barriers and suppression systems during different POSs are included in the fire damage analysis for scenarios specific to the POS.

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

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

Table F-16 Fire 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 in the response analysis.
PR-4	Identify potential circuit interactions which can interfere with safe shutdown.
PR-5	Modify human recovery failure events to account for fire-related impacts and quantify any fire-specific operator action.
PR-6	Estimate the required end-state frequency for each fire-induced scenario. Quantify the fire scenarios to obtain the desired risk metrics in accordance with the requirements identified for the internal event PRA accident sequence quantification but accounting for the combined effects of failures caused by fires and by random equipment failures or unavailability due to test or maintenance.
PR-7	Identify significant contributors (including assumptions, initiating events, POSs, accident sequences, component failures, and human errors) to the required end-states and ensure that all fire significant sequences are traceable and reproducible.

PR-8	Characterize and quantify the uncertainties in the results including parameter and model uncertainty and the contribution from assumptions. Understand their potential impact on the results.

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

F.3.4 Seismic PRA

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

Seismic hazard analysis estimates the frequency of different intensities of earthquakes based on a site-specific evaluation reflecting recent data and site-specific information. The analysis can be based on either historical data or a phenomenological model, or a mixture of the two. If existing studies are used to establish the seismic hazard, it is necessary to confirm that the basic data and interpretations that were used are still valid in light of current information. What ever the source of data, the hazard analysis should reflect the composite distribution of the informed technical community. Necessary inputs to the analysis include geological, seismological, and geophysical data, local site topography, surficial geologic and geotechnical properties. All sources of potentially damaging earthquakes and all credible mechanisms influencing vibratory ground motion should be accounted for in the hazard analysis. In addition, the effects of the local site response (e.g., topography and site geotechnical properties) should be included. Other seismic hazards such as fault displacement, landslide, soil liquefaction, or soil settlement should be reviewed to determine if they need to be included in the seismic PRA. Uncertainties in each step of the hazard analysis are propagated and included in the final hazard estimates for the site. The high level 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 hazard analysis is used, confirm that the data and information is still valid.
SH-2	The hazard analysis uses pertinent site information (e.g., geological, seismological, and geophysical data; site topography) and historical information.
SH-3	The hazard analysis considers all sources of potentially damaging earthquakes that can affect the seismic hazard at the site.
SH-4	The hazard analysis accounts for all credible mechanisms influencing vibratory ground motion that can occur at the site.
SH-5	Perform screening to address other seismic hazards, such as; fault displacement, landslide, soil liquefaction, or soil settlement, that need to be included in the seismic PRA.

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

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

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

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	Integrate the seismic hazard frequencies and the seismic fragilities into the plant system model.
SS-5	Quantify the seismic scenarios to obtain the desired risk metrics in accordance with the 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-6	Modify human recovery failure events to account for seismic-related impacts and include any seismic-specific recovery action.
SS-7	Identify significant contributors (including assumptions, initiating events, POSs, accident sequences, component failures, and human errors) to the required end-states and ensure that all significant sequences are traceable and reproducible.
SS-8	Characterize and quantify the uncertainties in the results including parameter and model uncertainty (using sensitivity analysis) and the contribution from assumptions. Understand their potential impact on the results.

F.3.5 Risk Assessment of Other External Events

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

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

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

Several current US LWRs sites may be submitted for possible future reactor sites. Existing sites will have very similar external events to consider but the results of the external events on the future reactors must be evaluated independently from the LWR on the site. The consequences the external event has on the future reactor may be different from the LWR and the systems in the future reactor will have different capabilities. Specifically, the impact of the external event on passive systems used in future reactors will have 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.

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-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 and justify the screening criteria used to eliminate external events from the scope of the PRA. Apply the screening criteria based on the plant's design and licensing basis relevant to the external event.
SB-3	Perform bounding evaluations of external events during all POSs, if required for comparison to quantitative screening criteria.
SB-4	Perform walkdowns of the plant and surrounding site to confirm the basis for screening of any external event.

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.

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 involving any source of radioactive material initiated by internal and external events during all modes of operation.

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	Integrate the external event hazard frequencies and the SSC fragilities into the plant system model.
SQ-5	Quantify the external event scenarios to obtain the desired risk metrics in accordance with the 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-6	Modify human recovery failure events to account for external event-related impacts and include any recovery actions specific to the external event.
SQ-7	Identify significant contributors (including assumptions, initiating events, POSs, accident sequences, component failures, and human errors) to the required end-states and ensure that all significant sequences are traceable and reproducible.
SQ-8	Characterize and quantify the uncertainties in the results including parameter and model uncertainty (using sensitivity studies) and the contribution from assumptions. Understand their potential impact on the results.

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 related to the reactor core would be; the status of coolant injection systems, the status of heat removal systems, the recoverability of failed systems after an undesirable end-state, and the interdependence of various systems. Grouping of accident sequences with similar behavior can be performed to reduce the amount of analysis required in the accident progression phase of the PRA. The accident progression analysis also models the effects accident phenomena (e.g., high temperatures or pressure) has on the available plant systems and human actions necessary to prevent containment failure or bypass. In addition, the effects of the internal and external accident initiators on these systems and human actions and the potential for additional random system failures are also included in the analysis.

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

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

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

Table F-24 Accident progression analysis requirements.

Item	Requirement

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

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

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

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. Include impacts resulting from system alignments during different POSs.
ST-3	Include accident sequence specific characteristics in the calculations that affect the timing, form and magnitude of radioactive material released from the fuel, coolant, and confinement.
ST-4	Characterize the source term with respect to the time, elevation, and energy of the release and the amount, form, and size of the radioactive material released to the environment.
ST-5	Characterize and quantify the uncertainties in the results including parameter and model uncertainty (using sensitivity studies) and the contribution from assumptions. Understand their potential impact on the results.

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

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

F.5 Consequence Assessment Technical Elements

The 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 involving any source of radioactivity, initiated by internal and external events during all modes of operation.

Consequence analysis evaluates the offsite consequences of an accidental release of radioactive material from a nuclear power plant expressed in terms of human health, environmental, and economic measures. The consequence measures of most interest focus on impacts on human health. Specific measures of accident consequences developed in a PRA can include: the number of early fatalities, the number of early injuries, the number of latent cancer fatalities, population dose at various distances from the plant, individual dose at various distances from the plant, individual early fatality risk defined in the early fatality QHO, individual latent cancer risk defined in the latent cancer QHO, and land contamination. The last three are of primary interest in the proposed 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, evacuation and sheltering plans, population data, and other required data and assumptions.
OC-3	Justify and document all parameter values and assumed parameter values.

OC-4	Ensure that the consequence code has been validated and verified.
OC-5	Characterize and quantify the uncertainties in the results including parameter and model uncertainty (using sensitivity studies) and the contribution from assumptions. Understand their potential impact on the results.

Health and economic risk estimation is the final step in a PRA that proceeds all the way to a Consequence Assessment. It integrates both the frequency and consequence results for accident sequences to compute the selected measures of risk. The high level 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.

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

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

G. SELECTION OF TOPICS FOR WHICH REQUIREMENTS ARE NEEDED

G.1 Introduction

In Chapter 8, the general process for the identification of topics for which technology-neutral requirements are needed was discussed. The purpose of this appendix is to apply this process to each of the five protective strategies described in Chapter 5 and to the administrative area. Section G.2 below describes the application of the process to the five protective strategies and Section G.3 describes its application to the administrative area.

G.2 Topics for the Protective Strategies

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

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

The protective strategies represent a high level defense-in-depth structure for developing the requirements in that each one represents a line of defense against the uncontrolled release of radioactive material and adverse impact on the health and safety of workers and the public. The process for identification of the scope and content of the detailed technical requirements for each protective strategy is described in Sections G.2.1 through G.2.5 below.

G.2.1 Physical Protection

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

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

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

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

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

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

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

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

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

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

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

- Physical protection needs to account for uncertainties and provide appropriate safety margins. Requiring security be considered integral with design, including a safety and security assessment assessing beyond DBT threats, will help address uncertainties and provide safety margin, thus providing high assurance of protection of public health and safety.

- Physical protection needs to be directed toward preventing an unacceptable release of radioactive material to the environment. In this regard, the security assessment should include an analysis of the release of radioactive material as a metric for decisions.

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

Table G-1 summarizes the logic tree of Figure G-1 by identifying questions that must be addressed by the technology-neutral requirements to ensure that the pathways that could lead to failure of the physical protection protective strategy are adequately covered in the requirements.

The table is organized by the three top level pathways of the logic tree and the answers to the questions in the table are the topics which must be covered by the requirements. The answers (i.e., topics) are arranged by whether they apply to design, construction, or operation.

As can be seen in Table G-1, many of the requirements needed to address this protective strategy already exist in 10 CFR 73. The framework and technology-neutral requirements would not change these requirements (i.e., any future design using the technology neutral requirements would also have to meet 10 CFR 73 requirements). However, for defense-in-depth reasons, Table G-1 does propose, that in addition to 10 CFR 73 future designs also consider physical protection in an integrated fashion as part of the design. This would require designers to perform a safety and security assessment on their designs against a range of threats, including beyond the DBT, based upon a set of security performance standards (as proposed in SECY-05-0120), and discussed in Section 6.4 of the framework. In addition, security considerations can affect the design of plant systems, structures and components with respect to their:

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

Therefore, security considerations must also be factored into the design.

Accordingly, the technology-neutral requirements need to include a requirement for such a safety and security assessment, including security performance standards. Chapter 6 (Section 6.4) discusses the security performance standards and each application to build a nuclear power plant under the technology-neutral requirements needs to include a safety and security assessment. Guidance on conducting a safety and security assessment will be provided in a separate document.

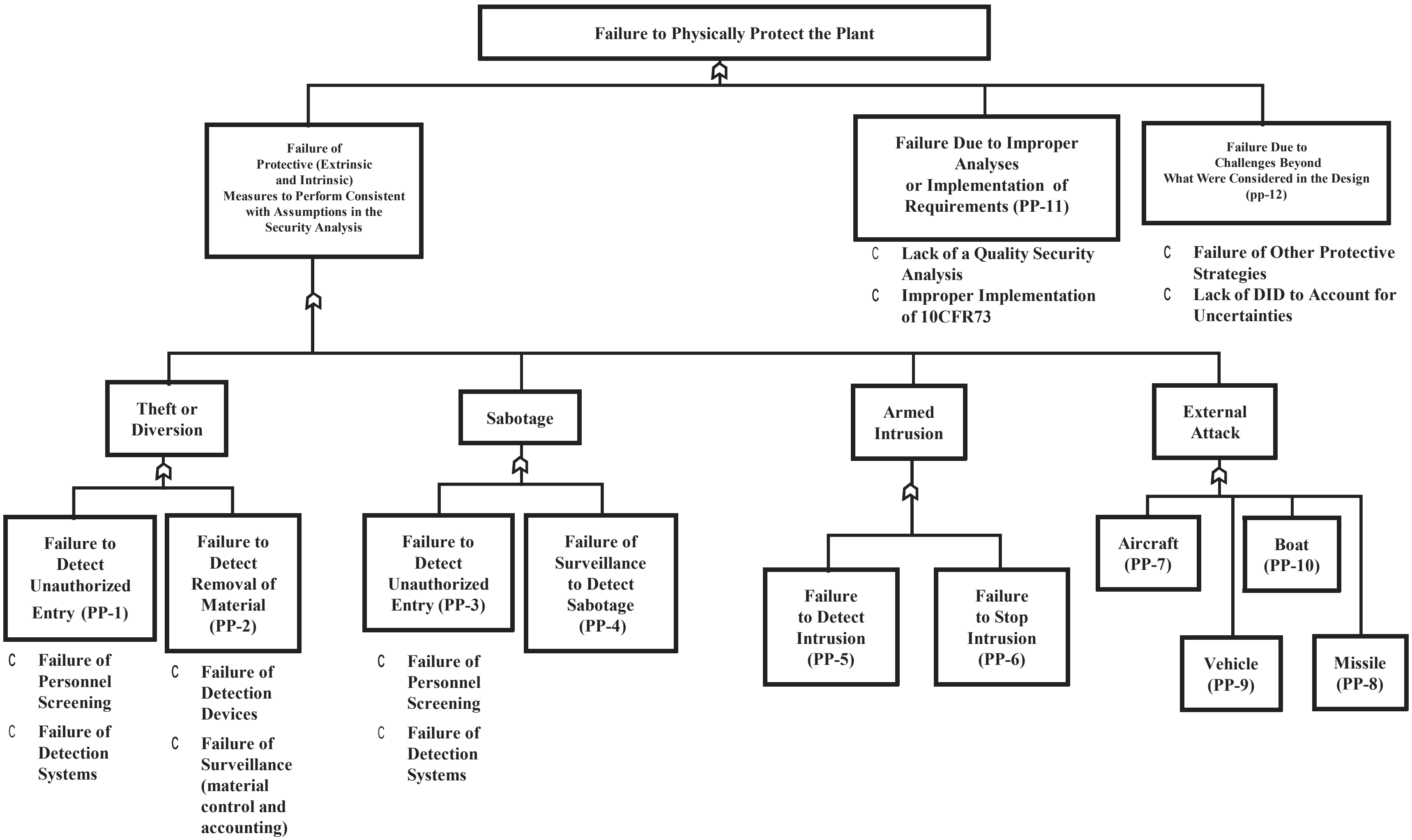

Figure G-1 Logic tree for the physical protection strategy.

Table G-1 Physical protection.

Protective Strategy Questions	Topics to be Addressed in the Requirements		
	Design	**Construction**	**Operation**
Failure of Protective Measures for Theft/Diversion			
• How should theft and diversion be detected? (PP-1) - detection systems	• Conduct security assessment integral with design, including security performance standards.	• N/A	• Implement results of security assessment, plus 10 CFR 73 requirements.
• How should unauthorized removal of material be detected? (PP-2) - personnel screening - detection systems	• 10 CFR 73 requirements.		• Detection and surveillance to check for loss (material control and accounting)
Failure of Protective Measures for Sabotage			
• How should unauthorized entry be prevented? (PP-3) - verify trustworthiness of personnel (i.e., personnel, screening) - detection systems	• Conduct security assessment integral with design, including security performance standards. • 10 CFR 73 requirements.	• Access Control	• Implement results of security assessment, plus 10 CFR 73 requirements.
• How can sabotage be detected? (PP-4)	• N/A	• QA, QC and surveillance to check for sabotage	• Surveillance to check for sabotage.
Failure of Protective Measures for Armed Intrusion			
• How can armed intrusion be detected? (PP-5)	• Conduct security assessment integral with design, including security performance standards.	• N/A	• Implement results of security assessment, plus 10 CFR 73 requirements.
• How can armed intrusion be stopped? (PP-6)	• 10 CFR 73 requirements.		

Table G-1 Physical protection.

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

N/A = Not applicable

G.2.2 Stable Operation

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

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

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

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

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

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

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

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

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

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

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

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

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

- Event sequences considered in the design that could disrupt stable plant operation must not be of such a nature as to defeat the protective systems, barrier integrity and protective actions strategies simultaneously. Accordingly, events with the potential to defeat all of these strategies need to be kept to a frequency of less than 10^{-7}/plant year. Such events might include reactor pressure vessel rupture, combustible gas explosion, or energetic recriticality. Reducing the frequency of such events to less than 10^{-7}/plant year will help ensure that no single event can defeat all protective strategies.

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

- Event sequences with the potential to simultaneously defeat the protective systems, barrier integrity and protective actions strategies need to have a frequency of less than 10^{-7}/plant year.

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

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

G.2.2.1 Design Stage

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

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

Reliability Assurance Program

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

Special Treatment

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

- QA / QC
- EQ (for temperature, humidity, radiation, etc.)
- Seismic qualification

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

Siting

Each design needs to have a link to the siting dose criteria. For current LWRs this is established through demonstrating that the releases that occur from design basis accidents do not exceed the dose criteria defined in 10 CFR 50.34(a)(1)(ii)(D) for the worst 2 hours at the exclusion area boundary (EAB) and for 30 days at the outer edge of low population zone (LPZ), as defined in 10 CFR 100.

The relationship between the technology-neutral requirements and 10 CFR 100 “Siting Requirements of Nuclear Power Plants” is intended to be one where the requirements of 10 CFR 100 would continue to apply and the technology-neutral requirements would contain requirements on the dose calculation necessary to demonstrate the “worst” 2-hour dose and the dose at the outer edge of the Low Population Zone are less than 25 rem TEDE (same requirement as is currently in 10 CFR 50.34). The dose calculation would be based upon the deterministic accident (discussed in Section G.2.4) selected to meet defense-in-depth principle # 5, which requires a controlled leakage barrier, independent from the fuel and RCS, with a capability to limit releases of radioactive material to the environment to acceptable levels. As discussed in Section G.2.4, the deterministic accident would be selected to address uncertainties in source term and would be analyzed mechanistically. However, it needs to be recognized that the technology-neutral requirements also require a range of low probability accidents (rare event category) to be analyzed and meet the doses represented by the F-C curve. Accordingly, design acceptability includes consideration of accidents beyond what has traditionally been considered.

Research and Development

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

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

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

Use of Prototype Testing

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

- What would be the objective of the test program:
  - Which aspects of plant safety can be addressed by demonstration plant testing?
  - Which types of analytical tools could be validated?
  - What phenomena could be addressed?
- What would be the scope of the test program:
  - How would the test program be selected?
  - Would it be conducted during initial startup only?
  - How would plant aging, irradiation, burnup effects be tested?
  - Would tests cover the full range of the accidents or only partial ranges, with the remainder done by analysis?
  - What instrumentation would be required?
- Are any special provisions needed in case the tests do not go as planned (e.g., containment, EP, has to be on a remote site, DOE site, etc.)?
- How would equipment reliability assumptions be verified?
- What acceptance criteria would be necessary (e.g., scope, treatment of uncertainties)?

- Would there be any limitations on future design changes?
- If the initial demonstration plant is to be licensed, how would this be accomplished?

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

G.2.2.2 Construction Stage

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

- factory fabrication to produce modules that can be installed in the field, thus reducing the amount of field fabrication;
- utilize components fabricated outside the U.S. and possibly to non-U.S. codes and standards; and
- in the case of HTGRs, have safety highly dependent upon the quality of the fuel fabrication and inspection process.

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

Factory Fabrication

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

Fabrication Outside the United States

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

Fuel Quality

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

G.2.2.3 Operating Stage

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

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

- training;
- use of procedures;
- radiation protection from routine operation (e.g., ALARA);
- maintenance;
- work control;
- configuration control; and
- surveillance, testing, ISI.

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

- radiation protection,
- worker protection during accidents;
- staffing;
- technical specifications;
- human factors; and
- corrective actions.

Additional discussion regarding each of these is provided in the paragraphs below.

Radiation Protection

The design also needs to consider limiting radiation doses to workers and the public from routine operation consistent with 10 CFR 20. This includes implementing the concept of “As

Low As Reasonably Achievable (ALARA) for workers and for releases to the environment. In this regard, 10 CFR 50, Appendix I provides guidance on permissible releases to the environment for LWRs. The technology-neutral requirements will need to develop criteria or generic guidance to apply the ALARA concept to other technologies.

Staffing

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

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

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

Technical Specifications

Technical specification limits for the new reactor technologies will need to be established at the technology-specific and design specific level. A scheme that utilizes insights from the PRA will need to be developed. This scheme would involve selecting events from the frequent, infrequent and rare categories that represent risk significant deviations from normal operations. Risk insights should be used to establish what SSCs are included in the technical specifications and what the limits on unavailability and allowable outage times should be. In addition, the success criteria from the PRA should be reviewed for application to TS limits. Lessons learned from efforts to risk-inform the technical specifications for currently operating LWRs should be considered in developing the requirements and any implementing guidance. It is likely that some experience will be needed in order to gain confidence in the limits that would be established by such a scheme.

Human Factors

A design that employs good human factors and man-machine interface practices will contribute to stable and safe plant operations. In this regard, guidelines have been developed for good human factors designs practices and good control room design practices for LWRs. These are found in NUREG-0711, “Human Factors Engineering Program”, and could be used as guidance to supplement the requirements. However, in general the requirements should, in a technology-neutral manner, address good human factors engineering practices that promote carrying out operations in a timely and accurate fashion, such as:

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

- procedures, and
- training.

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

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

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

Corrective Actions

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

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

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

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

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

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

Safety-Security Interface

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

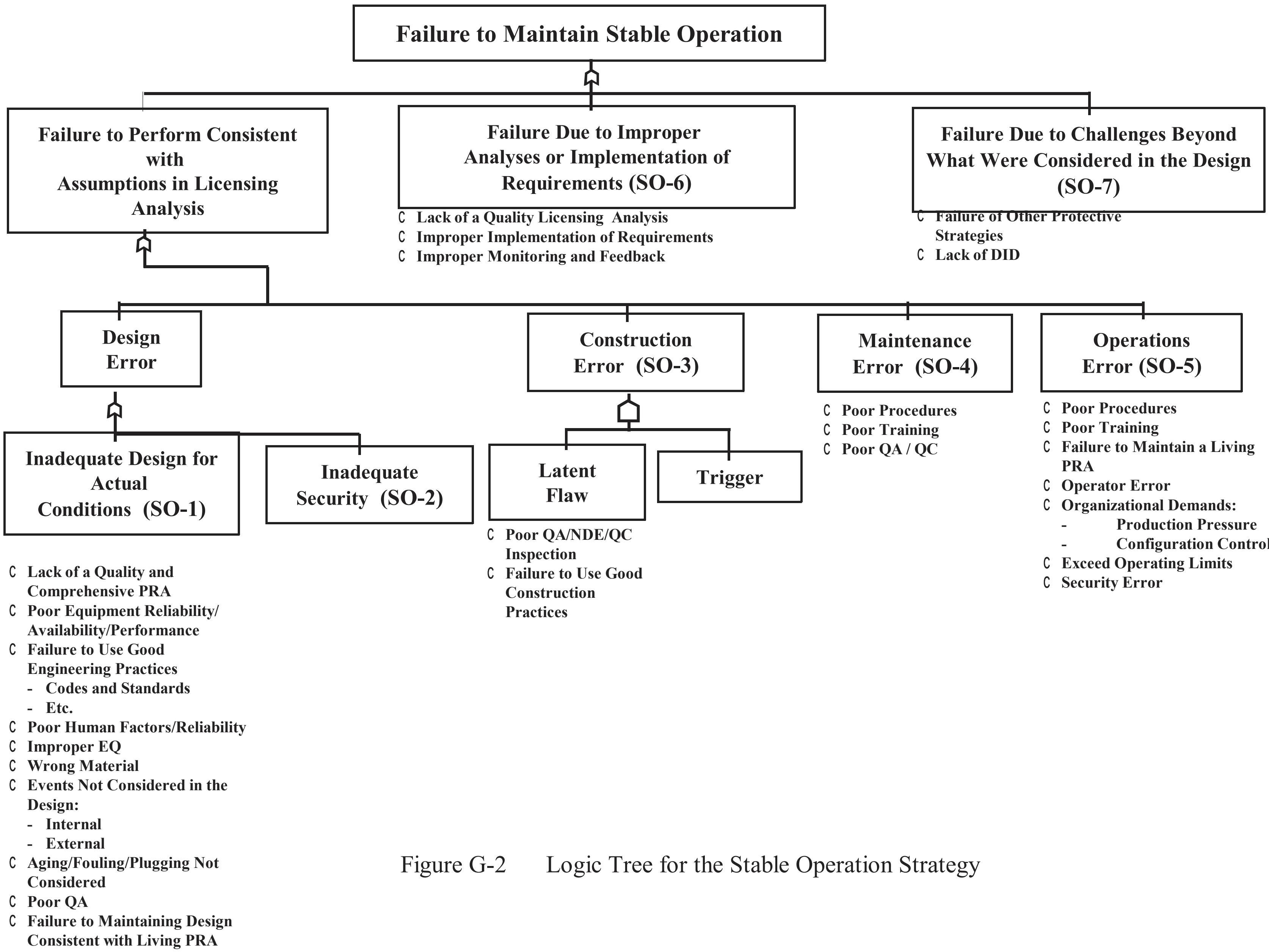

Figure G-2 Logic Tree for the Stable Operation Strategy

Table G-2 Stable operation.

Protective Strategy Questions	Topics to be Addressed in the Requirements		
	Design	**Construction**	**Operation**
Failure to Maintain Assumptions - Design Error			
• What needs to be done to ensure the design is adequate for the expected actual conditions? (SO-1)	• use event and LBE selection criteria in Chapter 6 • follow siting requirements (10 CFR 100) and consider effect of site specific events • ensure proper scope and quality of licensing analysis and consideration of uncertainties • use of good engineering practices: - use of consensus design codes and standards - good human factors design (e.g., automatic vs. operator action) - I&C qualification-software V&V - QA - proper EQ - flow blockage prevention - reactor inherent protection (e.g., no positive power coefficient) - qualified materials - qualified safety analysis tools • criticality prevention • Use of prototype testing • Research and Development • safety classification (see Chapter 6) • fire protection • prevention of brittle fracture • leak before break • consider plant aging, corrosion, etc. in the design	• N/A • N/A • N/A • QA/QC • Testing • Inspection • N/A • N/A • N/A • N/A • N/A • N/A • N/A • N/A • N/A • N/A • N/A • N/A	• monitoring and feedback into the design • maintenance • training • procedures • ISI • IST • staffing • aging management program

Table G-2 Stable operation.

Protective Strategy Questions	Topics to be Addressed in the Requirements		
	Design	Construction	Operation
	• specify reliability goals consistent with PRA : -reliability assurance program -specify goals on initiating event frequency • maintain design consistent with living PRA		• monitoring and feedback • monitoring and feedback
• What needs to be done to provide adequate security? (SO-2)	• see physical protection protective strategy	• see physical protection protective strategy	• see physical protection protective strategy
Failure to Maintain Assumptions - Construction Error			
• What needs to be done to prevent construction or manufacturing flaws? (SO-3)	• Specify construction /manufacturing methods to be used.	• Use of good construction/ manufacturing practices, including attention to factory fabrication and fabrication outside the U.S. • QA/QC • NDE • Inspection	• Surveillance • ISI • Testing
Failure to Maintain Assumptions - Maintenance Errors			
• What needs to be done to prevent maintenance errors? (SO-4)	• N/A	• N/A	• procedures • maintenance training • maintenance QA/QC
Failure to Maintain Assumptions - Operation Error			
• What needs to be done to limit operational errors? (SO-5)	• Consider human factors and man-machine interface as part of the design.	• N/A	• Utilize good operating practices: - training - procedures - maintenance - configuration and work control - use of simulators • technical specifications • security • personnel qualification

Table G-2 Stable operation.

Protective Strategy Questions	Topics to be Addressed in the Requirements		
	Design	Construction	Operation
Failure Due to Improper Analyses or Implementation of Requirements			
• How can failures due to improper analyses or implementation of requirements be prevented? (SO-6)	• Ensure quality analysis and that plant is designed consistent with licensing analysis, including event selection criteria in Chapter 6. • QA	• Ensure plant is constructed consistent with design. • QA/QC	• Ensure plant is maintained and operated consistent with licensing analysis. • Ensure fuel and replacement part quality is maintained over the life of the plant • Monitoring and feedback • Technical specifications
Challenges Beyond What were Considered in the Design			
• How can challenges beyond what were considered in the design (i.e., uncertainties) be accounted for? (SO-7)	• Apply other protective strategies and DID principles.	• N/A	• Surveillance • Monitoring and feedback
	• Frequency of events that could simultaneously defeat the protective systems, barrier integrity and protective actions strategies should be kept below 10^{-7} per plant year.	• N/A	• N/A
	• Consideration of uncertainties in PRA, LBE section and setting performance limits.	• N/A	• N/A

N/A = Not Applicable

G.2.3 Protective Systems

The protective systems protective strategy ensures that, should a challenge occur, systems are in place that will mitigate the resulting event sequences, i.e., arrest the sequences with no damage or minimize damage to the suite of barriers considered in the barrier integrity protective strategy.

The pathways leading to functional failure of a set of protective systems are shown in the logic tree of Figure G-3. The scope of the protective systems covered by this strategy include the front line protective systems and their support systems: those systems that provide needed services to the front line protective systems (e.g., I&C, electric power, and cooling). Note that the actual definition of protective system sets that must fail to lead to the actual loss of a protective function will depend on the details of final system design. At the top level, the major pathways leading to functional failure of protective systems are (1) failure of the protective systems to perform consistent with assumptions in the licensing analyses, (2) failures due to improper analyses or implementation of requirements, and (3) failures due to challenges beyond what were considered in the design. Each of these top level pathways is discussed further below.

Items that contribute to failures in the first top level pathway are design errors, construction (which includes manufacturing) errors, maintenance and operational errors. Design errors can lead to system failure by not properly including the events or conditions under which protective systems must function, the system performance needed to respond to these events, or the support systems needed into the design. Such design errors can result from poor QA, wrong assumptions on equipment performance or reliability / availability or not using good engineering practices in the design. Failures can also result from inadequate support systems or poor design for security. Accordingly, good QA is needed along with the use of good engineering practices and validated analytical tools. Also, protective systems should receive a safety classification consistent with their safety importance to ensure they are available and operable when needed during the operating stage.

Construction and manufacturing errors can also lead to protective systems failure by introducing latent flaws or by not thoroughly testing the systems for conditions under which they are to operate. The latent flaws can be the result of poor inspection, poor QA or QC, use of the wrong material or fabrication techniques or sabotage. Accordingly, the use of good construction and QA / QC practices are important to preventing failures.

Maintenance errors can also contribute to failure of protective systems. Maintenance programs that are incomplete may miss important contributors to failure such as plant aging, corrosion, etc. Poor training, procedures, spare parts, or QA / QC can cause maintenance errors and allow them to go undetected. Accordingly, maintenance programs should be comprehensive, including items such as aging management, and use of trained personnel and verified procedures.

Operations errors can also cause failure of protective systems. Such errors can result from poorly trained operators, poor procedures, poor work or configuration control or sabotage. Accordingly, the requirements must address these factors.

The second major pathway to failure of protective systems is that associated with failures due to improper analyses or implementation of requirements. Accordingly, ensuring quality analyses, the use of validated analytical tools and QA, along with items such as monitoring/feedback, technical specifications and safety classification should be used to ensure proper analyses and implementation of requirements during design and operation.

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

Applying the defense-in-depth principles to this protective strategy leads to the following:

- Protective systems can respond to intentional acts as well as inadvertent events. As described in Section G.2.1, "Physical Protection," security related issues and events need to be considered as an integral part of the design process. As discussed in Section G.2.1, a safety and security assessment should be done integral with design to assess whether or not protective systems design should be modified to make them less vulnerable to intentional acts or better able to mitigate intentional acts.

- Protective systems are provided that prevent events from leading to major plant damage as well as preventing the uncontrolled release of radioactive material to the environment should major plant damage occur. Applicants need to propose availability and reliability goals for the protective systems in consideration of the expected frequency of the events they are intended to respond to. Protective systems responding to events expected to occur one or more times during the life of the plant (frequent events in Chapter 6) should have high availability and reliability, whereas protective systems that are in the design to respond to events not expected to occur (infrequent and rare events in Chapter 6) may have a lower availability and reliability. To ensure this concept is implemented, the requirements need to require the designer to propose availability and reliability goals for the protective systems commensurate with the above, with overall plant risk goals and with assumptions used in the PRA.

- Key plant safety functions (i.e., reactor shutdown and decay heat removal) are not dependent upon a single protective system. Accordingly, it is envisioned that each of those functions, be accomplished by redundant, independent and diverse means, with each means having reliability and availability goals commensurate with overall plant risk goals. This represents a structuralist approach to defense-in-depth for these important functions to account for unquantified uncertainties, including common cause failure. It is intended that the requirement for redundant, diverse and independent means for reactor shutdown and decay heat removal be applied in the following manner:

  - The design should ensure that for frequent and infrequent event sequences, redundant, diverse and independent means for reactor shutdown and decay heat removal are available. For frequent events, the reliability and availability of the redundant, independent and diverse shutdown and decay heat removal systems should be sufficient such that no frequent event will make them inoperable. For infrequent events, which may involve loss of one decay heat removal path or means of reactor shutdown, the other path or means should have sufficient reliability and availability to be considered functional and ensure that the acceptance criteria for infrequent event sequences are met.

  - This functional requirement would not apply to event sequences in the rare category.

- In assessing the performance of protective systems, uncertainties in reliability, and performance need to be accounted for and appropriate safety margins provided. For new types of equipment or equipment with little or no operating experience at the conditions it will experience, a reliability assurance program (see Section G.2.2) needs to be provided to demonstrate and monitor equipment to ensure the assumptions of reliability, availability and performance used in the PRA and safety analyses are met. As discussed in Chapter 6, regulatory limits that are related to the failure of a piece of safety significant equipment, barrier or function should be set at the lower end of the expected uncertainty band so as to have an insignificant probability of failure as long as the limit is not exceeded, thus providing margin to the actual expected failure point. Also, the source term to be used in the safety analysis is to be that associated with the 95% confidence level (i.e., 95% of the ST is expected to be below the value used in the safety analysis). Use of the 95% value is intended to provide margin for the difficulty in modeling and in calculating the various phenomena associated with fission product release and transport. Finally, as discussed in Chapter 6, the dose calculated for LBEs is to be compared to the F-C curve using the 95% confidence value of the calculation. The use of the 95% value of the calculation is, among other things, intended to demonstrate the conservation of the PRA calculations (i.e., margin between the PRA analysis results and the F-C curve).

In addition to the items discussed above, two other areas that will inherently result in safety margins are worth noting. These areas are (1) the use of consensus codes and standards in the design of components and structures provides additional safety margins due to the conservatism built into their design rules and (2) the use of the NRC Safety Goal QHOs as the level of safety to be achieved provides margin to the "adequate protection" standard for licensing.

- The unacceptable release of radioactive material must be prevented. Accordingly, a means to prevent the uncontrolled release of radioactive material needs to be included in the design, consistent with the barrier integrity protective strategy (See Section G.2.4).

- Plant siting can affect the types and performance of safety systems since site specific hazards may be different. Site specific hazards and conditions need to be considered in the design consistent with 10 CFR 100 and the licensing analysis.

The above defense-in-depth considerations are reflected in the topics which the requirements must address, as shown in Table G-3.

Table G-3 identifies the questions that need to be answered to address each of the potential causes of protective system failure. The answers to these questions are organized by whether they apply to design, construction or operation and identify the topics which the technology-neutral requirements must address to ensure the success of this protective strategy. These topics are directed toward ensuring that quality analyses is used in the design process, that good engineering practices are used in the design and construction, that the equipment is tested, maintained and inspected over the life of the plant and that plant operations are conducted in a fashion that assures high reliability and availability of the protective systems (e.g., use of procedures and training need to be employed to minimize human errors). These considerations also apply to safety-significant support systems as well as the front line protective systems.

Finally, in assessing the performance of the protective systems (and the performance resulting from the other protective strategies) the design should meet the F-C curve and the QHOs, as described in Chapter 6. The F-C curve is to be met by each accident sequence in the PRA and in the LBE analysis. The QHOs represent an overall assessment of plant risk (considering all plant operating states and SSCs, including spent fuel storage). It is intended that the QHOs be assessed in an integrated fashion such that all new reactors on a site must meet the QHOs considering their risk in a cumulative fashion.

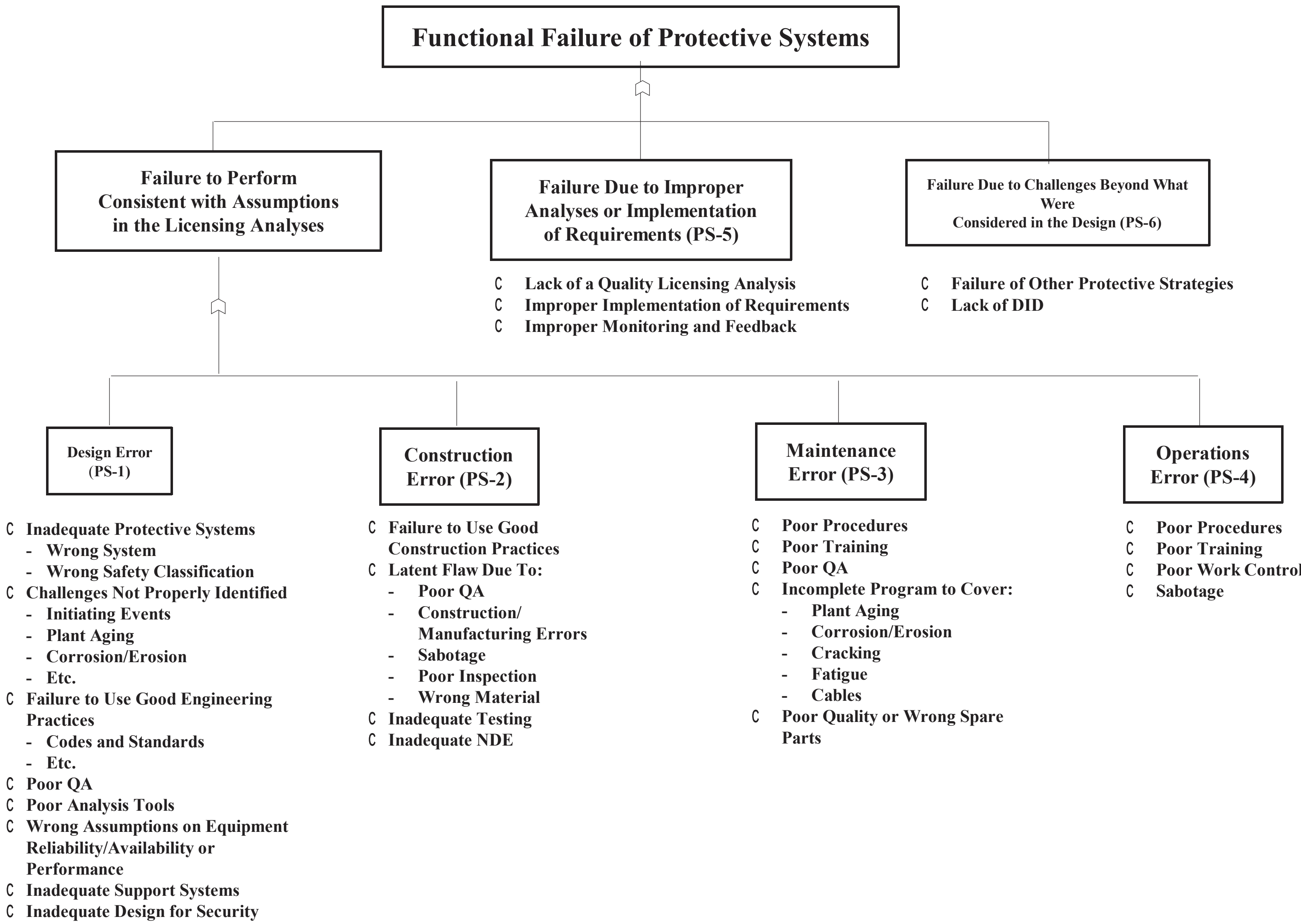

Figure G-3 Logic tree for the protective systems strategy.

Table G-3Protective systems.

Protective Strategy Questions	Topics to be Addressed in the Requirements		
	Design	**Construction**	**Operation**
Failure to Perform Consistent with Assumptions - Design Errors			
• How should systems be designed to ensure adequate performance and safety? (PS-1)	• Use licensing analysis to determine protective and support system needs (i.e., need quality licensing analysis) • Meet F-C curve • Meet QHOs, including integrated risk • Meet LBE acceptance criteria (Chap 6) • Use good engineering practices: - consensus design codes and standards - I&C qualification - software V&V - QA - qualified materials - EQ - combustible gas control - coolant/water/ fuel reaction control - qualified analytical tools - quality licensing analysis to determine performance and reliability needed • Safety classification (see Chapter 6) • Consider plant aging/ corrosion, etc. • Designer to specify reliability/availability goals consistent with PRA	• N/A • N/A • N/A • N/A • N/A • N/A • N/A • N/A	• Use living PRA to feedback operational experience into design. • N/A • N/A • N/A • N/A • Tech specs • Surveillance • Monitoring and feedback

Table G-3Protective systems.

Protective Strategy Questions	Topics to be Addressed in the Requirements		
	Design	**Construction**	**Operation**
Failure to Perform Consistent with Assumptions - Construction Error			
• What needs to be done to prevent construction errors? (PS-2)	• Specify good construction / fabrication practices as part of the design.	• Use good construction/ fabrication practices: • consensus codes and standards • QA / QC • access control	• N/A
Failure to Perform Consistent with Assumptions - Maintenance Errors			
• What needs to be done to prevent maintenance errors? (PS-3)	• N/A	• N/A	• procedures • training • QA/QC • comprehensive maintenance program, including: - plant aging - cables - corrosion - etc. • quality spare parts
Failure to Perform Consistent with Assumptions - Operation Errors			
• What needs to be done to prevent operations errors? (PS-4)	• Consider human factors and man-machine interface as part of design (e.g., automatic vs. operator actions)	• N/A	• procedures • training • use of simulator • technical specifications • surveillance • ISI • testing • good work control
Failures Due to Improper Analyses or Implementation			
• How can failures due to improper analyses or implementation of requirements be prevented? (PS-5)	• Ensure quality analysis and that plant is designed consistent with PRA and safety analysis. • QA	• Ensure plant is constructed consistent with design. • QA/QC	• technical specifications • monitoring and feedback

Table G-3Protective systems.

Protective Strategy Questions	Topics to be Addressed in the Requirements		
	Design	Construction	Operation
Failures Due to Challenges Beyond What Were Considered in the Design			
• How can challenges beyond what were considered in the design (i.e., uncertainties) be accounted for? (PS-6)	• provide 2 independent redundant and diverse ways to shutdown the reactor and remove decay heat	• N/A	• N/A
	• reliability assurance program	• N/A	• N/A

N/A = Not applicable

G.2..4 Barrier Integrity

The barrier integrity protective strategy is intended to ensure that the design provides sufficient physical (or chemical) barriers to prevent the uncontrolled release of radioactive material. The number and nature of the barriers will be technology and design dependent. Barrier integrity depends on adequate design, construction, maintenance and operation and, in some cases, on the success of protective systems. The logic tree of Figure G-4 lays out the events that can lead to functional failure of the barriers. If at least one barrier remains, the public is protected and workers are given a measure of protection. Barrier integrity applies to barriers associated with the reactor as well as spent fuel storage. Figure G-4 begins by identifying three major top level pathways that can lead to failure. These are:

- Failure to perform consistent with assumptions in the licensing analyses;
- Failures due to improper analyses or implementation of requirements; and
- Failures due to challenges beyond what were considered in the design.

Each of these is discussed in more detail in the following paragraphs.

The first major pathway (Failure to Perform Consistent with Assumptions in the Licensing Analyses) can be affected by design, construction, maintenance or operation errors, as discussed below.

Design errors leading to barrier failure can occur because the design is inadequate for the actual conditions that occur or conditions in excess of the design conditions occur. Failure can also occur by a failure of security, i.e., a loss of physical protection. Other design factors affecting barrier integrity are failure to consider barrier degradation mechanisms or poor QA / QC.

Construction and manufacturing errors are another source of barrier failure. Using good construction practices and having a good QA and QC program during the construction phase is essential to ensuring the plant is built as intended. Inspection, NDE and testing of barriers as construction proceeds are means to ensure the plant has been built as intended. Manufacturing processes for the fuel need to be controlled and qualified to ensure that fuel performance is consistent with design assumptions.

Maintenance errors are another potential source of barrier failure. These can occur due to leaving equipment in the wrong position, making a work error (e.g., forgetting to install a seal), not being trained or not following procedures. Accordingly, good work control, training and procedures are needed as well as a post maintenance test program to verify that barrier integrity is established. Finally, the maintenance program must cover all important degradation mechanisms that can affect barrier integrity.

Preventing operational errors is also important to maintaining barrier integrity. Poor procedures, training or work control could lead to barrier bypass or loss of integrity. To help prevent these errors, good training programs, verified procedures, surveillance, ISI and testing are needed. Also, sabotage is a potential source of barrier failure.

The second major pathway to barrier failure is associated with failures due to improper analyses or implementation of requirements. The licensing analysis will determine what barriers need to be in the design and how they should be designed. For normal operation and anticipated operational occurrences, reliable barriers to retain the fission products in the reactor and reactor coolant in the coolant system are necessary to meet the low levels of radioactive material release specified for normal operation. To ensure reliable barriers, the barriers should be designed and built to accepted consensus design codes using materials qualified for the intended service and accepted quality assurance measures.

For off-normal conditions, the event selection criteria discussed in Chapter 6 can be used to define the event scenarios and conditions which must be considered in designing the barriers. These criteria categorize event scenarios into those that are expected to occur one or more times during the life of the plant (frequent events), those that may occur once in a population of plants (infrequent events) and those considered in assessing overall plant risk and emergency preparedness (rare events).

Deterministic acceptance criteria for frequent and infrequent events have been developed in Chapter 6. Criteria on plant risk have also been developed in Chapter 6. To ensure the barriers perform as intended, they need to be qualified for the service conditions expected. This may involve research and development to verify fuel performance and equipment qualification (EQ) to verify the performance of mechanical items. Also, the analysis of barrier performance under normal and off-normal conditions will require safety analysis tools that need to be validated against experimental data. Depending upon the importance of the barriers to meeting the acceptance criteria, they may be assigned a safety classification (as described in Chapter 6) that will help ensure their performance availability and operability is maintained over the life of the plant.

It is also important that the assumptions associated with the analysis be properly implemented and controlled. Accordingly, items such as monitoring/feedback, technical specifications and safety classification needs to capture the key assumptions and provide control over the plant configuration and operation.

For the third major pathway (unanticipated challenges and failures), protection is provided by the other protective strategies (i.e., they are additional lines of defense) and by application of the defense-in-depth principles to account for completeness uncertainty.

Applying the defense-in-depth principles to the barrier integrity protective strategy leads to the following:

- The number of barriers and their design need to be based upon both intentional as well as inadvertent events. By requiring the design be done in an integral fashion considering security (see Section G.2.1), the barriers need to consider both.

- The barriers need to be designed with both accident prevention and mitigation in mind. Accident prevention will be achieved by ensuring that the barriers are designed to be highly reliable and can withstand a range of off-normal conditions. High reliability needs to be achieved by the use of good engineering practices (such as the use of consensus design codes and standards, qualification of materials, QA, etc.) in the design and performing surveillance, inspection and testing during the plant lifetime. Barriers also need to be designed to maintain their integrity for events expected to occur during the plant lifetime such that their failure does not become an initiating event.

  Accident mitigation will be achieved by ensuring the barriers perform their function of containing radioactive material. The events for which they must perform their function, their design and their degree of leak tightness will be design dependent, as will the total number of barriers needed. Minimum requirements for barriers are discussed below.

- Defense-in-depth requires that key safety functions not be dependent upon a single element of design, construction, operation or maintenance. Application of this principle to barrier integrity implies multiple barriers are needed, since containment of radioactive material is considered a key safety function. Accordingly, at least two barriers to the release of radioactive material need to be provided, since the failure of one of these barriers (e.g., the reactor coolant system barrier) could be an initiating event. In general, the barriers, in conjunction with other plant features, need to be capable of limiting dose to the public consistent with the frequency consequence curve in Chapter 6.

- In the design and safety analysis, uncertainties in reliability and performance need to be accounted for and appropriate safety margins provided. As discussed in Chapter 6, regulatory limits that are related to the failure of a piece of safety equipment, barrier or function should be set at the lower end of the expected uncertainty band so as to have an insignificant probability of failure as long as the limit is not exceeded, thus providing margin to the actual expected failure point. However, not all uncertainties can be quantified. Therefore, it is considered reasonable to require each design to have additional capability (beyond the two barriers described above) to mitigate against accident scenarios that result in the release of larger amounts of radioactive material by providing margin to account for unquantified uncertainties that result in a larger source term available for release to the environment (e.g., security related events). Accordingly, as a structuralist defense-in-depth provision, each design needs to have a containment functional capability (i.e., the capability to establish a controlled low leakage barrier) in the event plant conditions result in the release of radioactive material from the core and reactor coolant system in excess of anticipated conditions. The specific conditions regarding the leak tightness, temperature, pressure and time available to establish the containment functional capability will be design specific. The design of the containment functional capability is to be based upon a process that defines an event representing a serious challenge to fission product retention in the core and coolant system. The event needs to be agreed upon between the applicant and the NRC consistent with the technology and safety characteristics of the design. The event could represent an event where fission product retention in the core and coolant system suddenly changes due to small changes elsewhere, a low probability event from the PRA, a security related event or an assumed fuel damage event.

For LWRs, core melt accidents will likely continue to be used to establish the design conditions for the containment functional capability. For non-LWRs, examples of the types of events that could be considered for establishing the design conditions for the controlled leakage barrier are:

- HTGRs
    - graphite fire in the core
    - water ingress to the core
    - loss of coolant accident in conjunction with poor quality fuel

- LMRs
    - flow blockage in the core
    - large liquid metal fire
    - loss of normal heat removal in conjunction with poor quality fuel

The selection of the event to be used to establish the design conditions for the containment functional capability is not intended to impose a traditional LWR type containment on all technologies, but rather to allow each technology to have designs that reflect their unique safety characteristics while providing margin for uncertainties in the source term available for release to the environment (e.g., venting to the atmosphere early in an accident scenario may be acceptable for some technologies).

The selected event should be analyzed mechanistically to determine the timing, magnitude and form of radionuclide released into the reactor building, and the resulting temperature, pressure and other environmental factors (e.g., combustible gas) in the building over the course of the event. The timing of closure and the allowable leak rate should then be established such that the worst two-hour exposure at the EAB and the exposure at the outer edge of the LPZ for the duration of the event do not exceed 25 rem TEDE. Chapter 6 contains additional guidance regarding analysis of this event.

- Barriers need to prevent the unacceptable release of radioactive material. Accordingly, to account for uncertainties (see paragraph above), the reactor needs to have a containment functional capability independent from the fuel and RCS, as discussed above.

- Barrier integrity interfaces with siting in that some aspects of barrier performance may be determined by site characteristics (e.g., meteorology, population distribution). Likewise, barrier integrity can also affect the type and extent of off-site protective measures needed. These need to be accounted for in the design.

The above defense-in-depth considerations have been factored into the requirement topics shown in Table G-4.

Table G-4 shows a set of questions and answers associated with the Barrier Integrity protective strategy. The questions are organized by the top level branches of the logic diagram and the answers (i.e., the topics which must be covered by the requirements) are arranged by whether they apply to design, construction or operation.

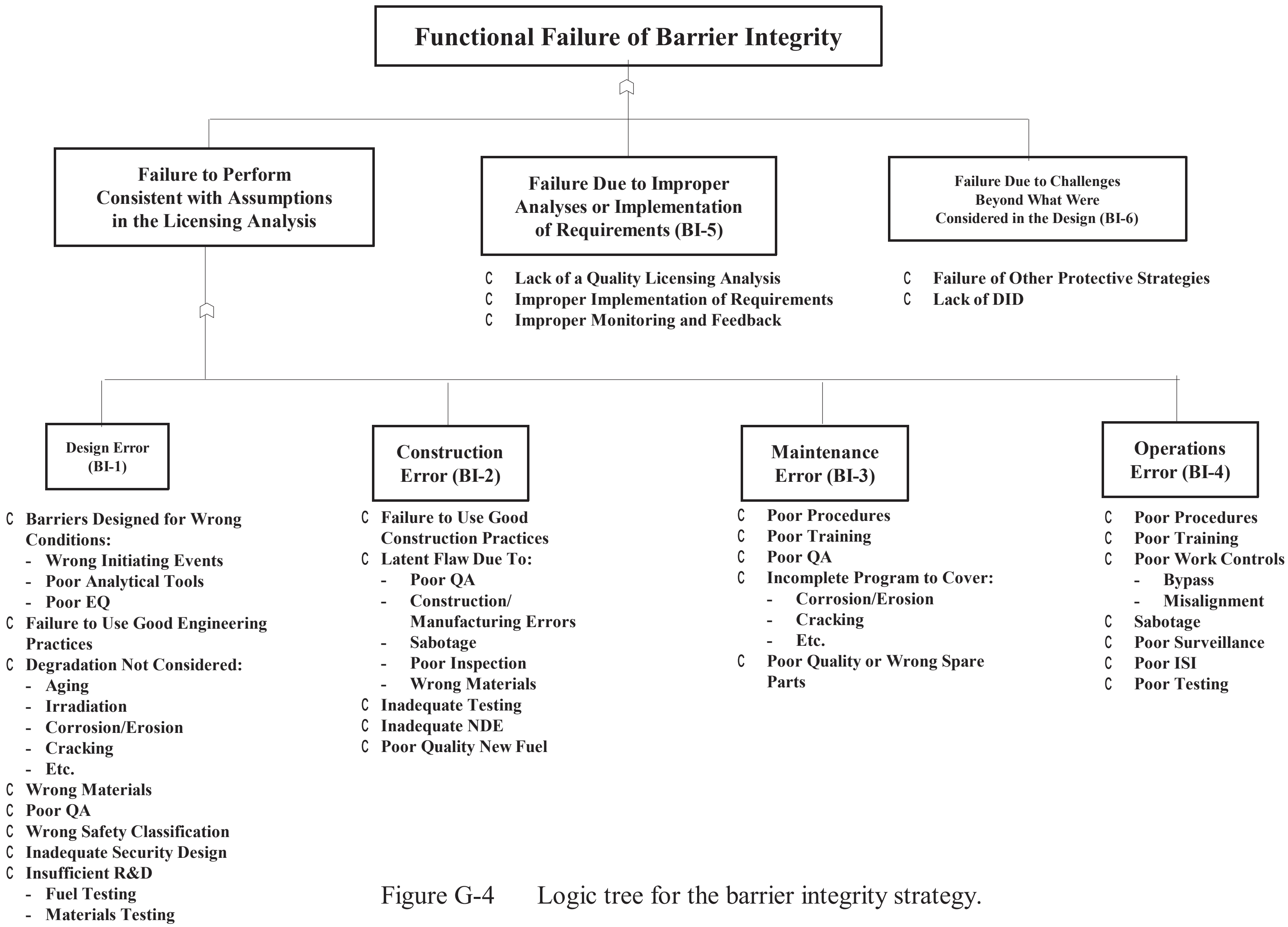

Figure G-4 Logic tree for the barrier integrity strategy.

Table G-4Barrier integrity.

Protective Strategy Questions	Topics to be Addressed in the Requirements		
	Design	**Construction**	**Operation**
Failure to Perform Consistent with Assumptions - Design Errors			
• How should adequate barrier design (integrity and reliability) be assured? (BI-1)	• Design barriers consistent with: - Chapter 6 event selection criteria - Chapter 6 LBE acceptance criteria (probabilistic, e.g., F-C curve, and deterministic) - Safety classification - EQ - Consider degradation mechanisms • Provide barriers for: - fission product retention (in the fuel) - coolant retention (in the reactor I cooling system) - Other capability, as necessary to meet safety objectives • Use good engineering practices: - quality assurance - materials qualification - use of accepted design codes and standards - use of validated safety analysis tools - consider aging and other degradation phenomena	• N/A	• N/A

Table G-4Barrier integrity.

Protective Strategy Questions	Topics to be Addressed in the Requirements		
	Design	**Construction**	**Operation**
Failure to Perform Consistent with Assumptions - Construction Errors			
• What needs to be done to prevent construction errors? (BI-2)	• specify construction/ manufacturing techniques at the design stage.	• use good construction/ manufacturing practices: - consensus construction codes and standards - QA/QC - inspection - testing - NDE - assure fuel quality over the life of the plant • access control • surveillance	• N/A
Failure to Perform Consistent with Assumptions - Maintenance Error			
• What needs to be done to prevent maintenance errors? (BI-3)	• N/A	• N/A	• verified procedures • good training • QA/QC • have a comprehensive maintenance program • use quality spare parts
Failure to Perform Consistent with Assumptions - Operations Error			
• What needs to be done to prevent operational errors? (BI-4)	• Use good HF and HMI engineering • Use fault tolerant designs	• N/A	• verified procedures • good training • use of simulator • good work control • good surveillance • ISI • testing
Failures Due to Improper Analyses or Implementation of Requirements			
• How can failures due to improper analyses or implementation of requirements be prevented? (BI-5)	• Use verified analytical tools • Quality PRA and safety analyses • Ensure plant is designed consistent with PRA and safety analysis. • QA	• Ensure plant is constructed consistent with design. • QA/QC	• technical specifications • safety classification • monitoring and feedback

Table G-4Barrier integrity.

Protective Strategy Questions	Topics to be Addressed in the Requirements		
	Design	**Construction**	**Operation**
Failures Due to Challenges Beyond What Were Considered in the Design			
• How can challenges beyond what were considered in the design (i.e., uncertainties) be accounted for? (BI-6)	• at least 2 barriers for the reactor • provisions to establish a containment functional capability independent of fuel and RCS for the reactor.	• N/A • N/A	• technical specifications • technical specifications

N/A = Not applicable

G.2.5Protective Actions

The protective actions strategy ensures that adequate systems, equipment, and practices are in place to control and terminate the accident progression, to minimize damage to the barriers, to limit the release of radionuclides, to protect workers, and to limit public health effects. Protective actions generally include EOPs, accident management and on-site and off-site emergency preparedness.

Figure G-5 is a logic tree showing the pathways that can lead to failure of protective actions. At the top level, three major pathways to failure are: (1) failure to take protective actions consistent with assumptions in the licensing analysis, (2) failure due to improper analyses or implementation of requirements, or (3) failures due to challenges beyond what were considered in the design. Each of these top level pathways is discussed further below.

In the first top level pathway (Failure to Take Protective Action Consistent with Assumptions in the PRA and Safety Analysis), failure can be associated with either on-site or off-site actions, as shown in Figure G-5. Failure of on-site protective actions can be associated with operations, hardware or software, training or design. Off-site failures can occur in areas regulated by the NRC or in areas controlled by other agencies. For example, state and local officials are responsible for many aspects of the off-site response (e.g., evacuation).

On-site failures due to operational problems can result in failure to terminate the accident (thus making conditions on-site, and possibly off-site, worse) or failure to adequately protect operating personnel. Operating personnel are vital to plant safety and are called on to perform safety related actions during design basis and beyond-design-basis events (e.g., accident management actions). Accordingly, protection of the operating staff during accidents also needs to be considered in the design and operation of future reactors.

General Design Criteria (GDC) 19 of 10 CFR Part 50 Appendix A currently requires main control rooms to be designed to ensure habitability under a variety of conditions, including design basis accident conditions. The conditions which must be considered include a postulated source term representative of a LWR core melt accident (or an alternate source term) and chemical releases. As a result, LWR main control rooms are provided with shielding and habitability systems that ensure the safety of the operators during the postulated conditions. Accordingly, the technology-neutral requirements should include a similar provision for protection of control room staff during accidents, recognizing the use of the PRA to select the accident scenarios which must be considered and the use of scenario specific source terms.

However, no corresponding requirements exist in 10 CFR 50 for protection of operating staff outside the main control room, who may be called upon to perform accident management actions and communicate with other staff during accident situations. In the development of accident management programs for existing LWRs (which were developed on the basis of a voluntary industry initiative), it was recognized that access by the operating staff to certain portions of the plant was essential to carry out the planned actions. Accordingly, NEI, in its "Severe Accident Issue Closure Guidelines" document (NEI-91-04, Rev. 1, dated December 1994) on the development of accident management programs, identified operational and phenomenological conditions as factors which must be assessed in planning and implementing operator accident management actions.

For new plants, the technology-neutral requirements should require that the procedures and accident management programs consider the environment (e.g., temperature, radiation) in which local operator actions take place and ensure that the design (e.g., shielding, access) and procedures sufficiently protect all the operators so that the actions can be safely accomplished without serious injury. For radiation exposure (during such activities), the limits in 10 CFR Part 20.1201, "Occupational Dose Limits" should be used for frequent event scenarios and 10 CFR Part 20.1206, "Planned Special Exposures" should be used as the measure to prevent serious injury for personnel outside the control room during frequent and rare event scenarios. Regulatory Guide 8.38 provides additional guidance in this area regarding access to high radiation areas. For personnel inside the control room, limits similar to those in GDC-19 could be used. Scenario specific source terms may be used in the assessment, consistent with those used in other accident analyses. Other accepted limits should be applied for other hazards (temperature, chemicals, etc.).

On-site hardware or software problems can lead to unintended actions and/or poor decisions. Accordingly, measures to ensure reliable equipment and software are needed. Poor training can also lead to the same consequences as poor operations or poor hardware/software. Training programs need to be complete and conducted periodically to keep operating personnel skills current. Design problems can result in needed equipment not being present, instrumentation and/or communication not sufficient to understand the accident, personnel access and habitability restricted more than anticipated or personnel injury or death. Therefore, during the design stage, accident scenarios (including those related to security failure) must be considered integral with the design and measures to ensure good EOPs and accident management need to be provided.

Off-site preparedness failures can lead to failure to take measures needed to protect the public. Such failures could be due to hardware problems (e.g., failure to notify), poor planning (e.g., traffic jams delay evacuation) or an insufficient implementation for the accident consequences. Off-site organizational failures can also lead to failures to adequately protect the public. Such failure can be due to poor coordination among off-site authorities, poor communication, poor training or poor decisions (i.e., not implementing the appropriate protective measures at the appropriate time).

The second top level pathway is associated with failures due to improper analyses or implementation of requirements. Quality analyses and the use of verified analytical tools are essential. In addition, the EOPs and AM procedures should be developed in an integrated fashion with the design so that the design can provide reasonable measures for AM and ensure the procedures are consistent with the PRA and safety analysis.

For the third top level pathway (failures due to challenges beyond what were considered in the design), protection is provided by the application of the defense-in-depth principles to account for completeness uncertainty. Applying the defense-in-depth principles to this protective actions strategy leads to the following:

- The development of protective actions needs to consider intentional acts as well as inadvertent events. The physical protection protective strategy (Section G.2.1) provides further guidance on evaluating security integral with design.

- Protective actions need to include measures to terminate the accident progression (referred to as EOPs, and accident management) and pre-planned measures to mitigate the accident consequences (referred to as emergency preparedness). The EOPs, AM procedures and EP need to be developed in an integrated fashion with the design.

- The accomplishment of protective actions must not rely on a single element of design, construction, maintenance or operation. As such, normal operating, EOPs, accident management and EP procedures need to be developed so as not to have key safety functions dependent upon a single human action or piece of equipment.

- Protective actions need to be developed in consideration of uncertainties and appropriate safety margins provided. As a structural defense-in-depth measure, emergency preparedness needs to be included in the design and operation to account for unquantified uncertainties.

- Prevention of unacceptable releases of radioactive material need to be part of the AM program.

- Plant siting will affect EP and needs to be considered in developing EP plans.

The above defense-in-depth considerations are reflected in Table G-5.

Table G-5 below summarizes each of these pathways in the form of questions, the answers to which identify the topics that the technology-neutral requirements must address to prevent pathway failure. The answers (i.e., topics) are arranged according to whether they apply to design, construction or operation.

As can be seen from Table G-5, there are a number of topics that should be addressed in the requirements to assure an adequate protective actions strategy. Some of these (e.g., drills, training) can utilize the technology-neutral requirements contained in 10 CFR 50, while others will need to be developed in a technology-neutral fashion consistent with a risk-informed approach. A major item in this regard would be a requirement for the development of the design (and its associated systems and instrumentation) in an integrated fashion with the development of EOP and AM procedures. Such an integrated process would help ensure that the procedures address all of the relevant accident scenarios in the PRA (and scenarios from security considerations) and that the design includes features that facilitate AM.

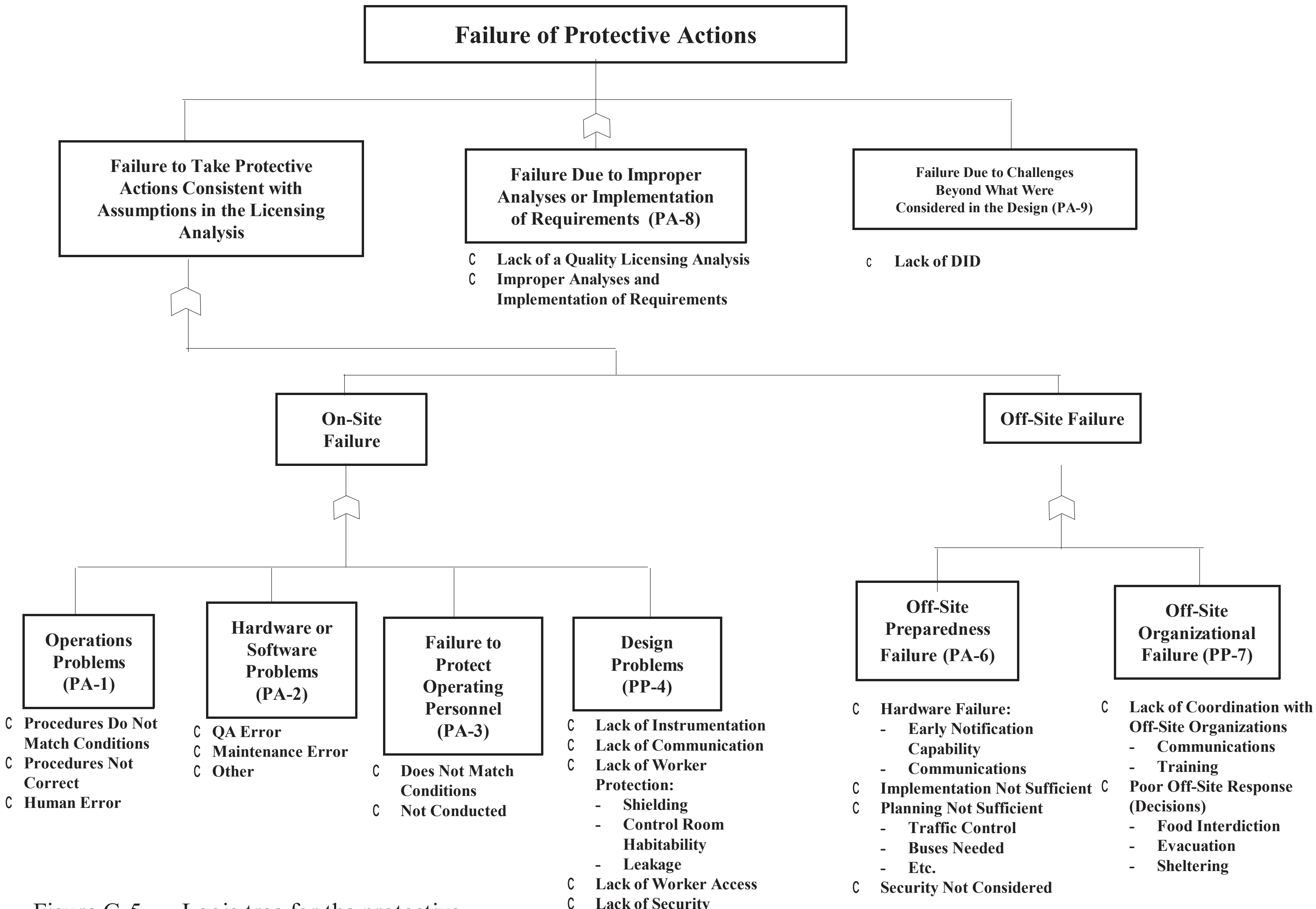

Figure G-5 Logic tree for the protective actions strategy.

Table G-5 Protective actions.

Protective Strategy Questions	Topics to be Addressed in the Requirements		
	Design	Construction	Operation
Failure to take Protective Actions Consistent with Assumptions: On-Site Failure			
• How can operations problems be prevented? (PA-1)	• N/A	• N/A	• Develop comprehensive training programs and require periodic training. • Use of simulator • Use of verified procedures
• How can hardware and software be assured to be operable? (PA-2)	• Reliability assurance program for hardware • Software V and V	• Testing • QA/QC • N/A	• Maintenance program • Testing
• How can it be assured operating personnel are properly protected? (PA-3)	• Provide appropriate shielding and habitability for the control room and other areas needing access.	• N/A	• Establish comprehensive worker protection programs, training and monitoring. • Ensure 10 CFR 20 requirements are complied with.
• How can design deficiencies/ problems be prevented? (PA-4)	• Develop EOP and AM design features and procedures integral with design, including identifying equipment, instrumentation, and communication needs. • Provide alternate shutdown location	• N/A • N/A	• N/A • N/A
• How can adequate on-site preparedness be assured? (PA-5)	• Develop on-site EP plans and procedures integral with design	• N/A	• N/A • Training • Procedures • Coolant drills and training to demonstrate effectiveness of on-site EP
Failure to Take Protective Actions Consistent with Assumptions - Off-Site Failure			
• How can adequate off-site preparedness be assured? (PA-6)	• Provide adequate emergency planning • Consider security related events	• N/A	• Conduct drills and training to demonstrate effectiveness of off-site EP • Integrate security and preparedness

Table G-5 Protective actions.

Protective Strategy Questions	Topics to be Addressed in the Requirements		
	Design	Construction	Operation
• How can adequate off-site organizational performance be assured? (PA-7)	• Provide reliable communication equipment	• N / A	• Conduct drills and training to demonstrate effectiveness of EP
Failures Due to Improper Analyses or Implementation of Requirements			
• How can failures due to improper analyses or implementation of requirements be prevented? (PA-8)	• Quality licensing analyses • Use verified analytical tools • Develop EOPs and AM procedures integral with design. • QA	• N/A • N/A • N/A • N/A	• Ensure training program is comprehensive and conducted periodically. • Use of simulator • N/A
Failures Due to Challenges Beyond What Were Considered in the Design			
• How can challenges beyond what were considered in the design (i.e., uncertainties) in protective actions be accounted for? (PA-9)	• Consider security related events beyond the DBT. • Develop EOPs and AM integral with design. • Do not have key safety functions dependent upon a single human action or piece of hardware.	• N/A • N/A • N/A	• Consider security related events beyond the DBT. • Training • Drills • EP • Procedures

N/A = Not applicable

G.2.6 Summary of Topics for the Protective Strategies

Sections G.2.1 through G.2.5 identify the topics that the technology-neutral requirements must address to ensure the success of the protective strategies. Some of the topics identified are applicable to more than one protective strategy (e.g., QA, training, etc.). Accordingly, a summary table (Table G-6) has been prepared that consolidates the technical topics from Tables G-1 through G-5, eliminating any duplication. Table G-6 also organizes the topics in a more logical fashion (i.e., by subject) and identifies the appropriate question numbers from Table G-1 through G-5 that identified that topic.

It needs to be recognized that Table G-6 presents a broad, high level overview of the topics which the technology-neutral technical requirements must address. Many details need to be developed in the course of writing the requirements. Accordingly, reference to the appropriate section in the framework is also shown in Table G-6 for additional guidance.

As described in Sections G.2.1 through G.2.5, the defense-in-depth principles from Chapter 4 were applied to each protective strategy to ensure adequate treatment of uncertainties. Application of the defense-in-depth principles to each of the protective strategies (as described in Sections G.2.1 through G.2.5) has also led to the identification of a number of specific topics to address uncertainties. Although included in Table G-6, these are also summarized in Table G-7 to illustrate the defense-in-depth provisions identified by the application of the DID principles in Chapter 4. The technology-neutral requirements also need to include the defense-in-depth principles and process, so that applicants and licensees are required to implement a defense-in-depth review on their designs.

Table G-6 Technical Topics for technology-neutral requirements.

Topic	Framework Technical Description
(A) Topics Common to Design, Construction and Operation	
1) QA/QC (Questions PP-4, SO-1, SO-3, SO-4, SO-6, PS-1, PS-2, PS-3, PS-5, BI-1, BI-2, BI-3, BI-5, PA-2, PA-8)	Appendix G - Section G.2.2
2) PRA scope and quality (PP-11, SO-1, SO-6, PS-1, PS-5, BI-1, BI-5, PA-8)	Chapters 6, 7 and Appendix F
(B) Physical Protection	
1) General (10 CFR 73) (PP-1 through 11)	Appendix G - Section G.2.1
2) Perform security assessment integral with design (PP-1 through 12)	Appendix G - Section G.2.1
3) Security performance standards (PP-1 through 12)	Section 6.4
(C) Good Design Practices	
1) Plant Risk (PS-1, BI-1): - Frequency-Consequence curve - QHOs (including integrated risk)	Chapter 6
2) Criteria for selection of LBEs (SO-1)	Chapter 6
3) LBE acceptance criteria (PS-1): • frequent events (dose, plant damage) • infrequent events (dose, plant damage) • rare events (dose) • link to siting	Chapter 6
4) Keep initiating events with potential to defeat two or more protective strategies $<10^{-7}$/plant year (SO-7)	Appendix G - Section G.2.2
5) Criteria for safety classification and special treatment (SO-1, PS-1, BI-1, BI-5)	Chapter 6
6) Equipment Qualification - (SO-1, PS-1)	Section G.2.2
7) Analysis guidelines (SO-1) • realistic analysis, including failure assumptions • source term	Chapter 6
8) Siting and site specific considerations (SO-1)	Appendix G - Section G.2.2

Table G-6 Technical Topics for technology-neutral requirements.

Topic	Framework Technical Description
9) Use consensus design codes and standards (SO-1, PS-1, BI-1)	Appendix G - Section G.2.2
10) Materials and equipment qualification (SO-1, PS-1, BI-1)	Appendix G - Section G.2.2
11) Provide 2 redundant, diverse, independent means for reactor shutdown and decay heat removal (PS-6)	Appendix G - Section G.2.3
12) Minimum - 2 barriers to FP release (BI-1, BI-6)	Appendix G - Section G.2.3
13) Containment functional capability (BI-6)	Appendix G - Section G.2.4
14) No key safety function dependent upon a single human action or piece of hardware (PA-9)	Appendix G - Section G.2.5
15) Need to consider degradation and aging mechanisms in design (SO-1, PS-1, BI-1)	Appendix G - Section G.2.2
16) Reactor inherent protection (i.e., no positive power coefficient, limit control rod worth, stability, etc.) (SO-1)	Appendix G - Section G.2.2
17) Human factors considerations (SO-1, SO-5, PS-4, BI-4)	Appendix G - Section G.2.2
18) Fire protection (SO-1)	Appendix G - Section G.2.2
19) Control room design (PA-3)	Appendix G - Section G.2.5
20) Alternate shutdown location (PA-4)	Appendix G - Section G.2.5
21) Flow blockage prevention (SO-1)	Appendix G - Section G.2.2
22) Specify reliability and availability goals consistent with PRA (SO-1, PS-1, PS-6, PA-2) - establish Reliability Assurance Program - specify goals on initiating even frequency	Appendix G - Section G.2.2
23) Use of prototype testing (SO-1)	Appendix G - Section G.2.2
24) Research and Development (SO-1)	Appendix G - Section G.2.2
25) Combustible gas control (PS-1)	Appendix G - Section G.2.3
26) Coolant/water/fuel reaction control (PS-1)	Appendix G - Section G.2.3
27) Prevention of brittle fracture (SO-1)	Appendix G - Section G.2.2
28) Leak before break (SO-1)	Appendix G - Section G.2.2
29) I and C System (SO-1, PS-1, PA-2) • analog • digital • HMI	Appendix G - Section G.2.2
30) Criticality prevention (SO-1)	Appendix G - Section G.2.2

Table G-6 Technical Topics for technology-neutral requirements.

Topic	Framework Technical Description
31) Protection of operating staff during accidents (PA-3)	Appendix G - Section G.2.5
32) Qualified analysis tools (SO-1, SO-6, PS-1, PS-5, BI-1, BI-5, PA-8)	Chapter 6
(D) Good Construction Practices	
1) Use accepted codes, standards, practices (SO-3, PS-2, BI-2)	Appendix G - Section G.2.2
2) Security (See (B) above)	Appendix G - Section G.2.1
3) NDE (SO-3, BI-2)	Appendix G - Section G.2.2
4) Inspection (SO-1, SO-3, BI-2)	Appendix G - Section G.2.2
5) Testing (SO-1, BI-2)	Appendix G - Section G.2.2
(E) Good Operating Practices	
1) Radiation protection during routine operation (PA-3)	Appendix G - Section G.2.2
2) Maintenance program (SO-1, SO-5, PS-3, BI-3, PA-2)	Appendix G - Section G.2.2
3) Personnel qualification (SO-5)	Appendix G - Section G.2.2
4) Training (SO-1, SO-4, SO-5, PS-3, PS-4, BI-3, PA-1, PA-5, PA-6, PA-7, PA-8, PA-9)	Appendix G - Section G.2.2
5) Use of Procedures (SO-1, SO-4, SO-5, PS-3, PS-4, BI-3, BI-4, PA-1, PA-5)	Appendix G - Section G.2.2
6) Use of simulators (SO-5, PS-4, BI-4, PA-1, PA-8)	Appendix G - Section G.2.2
7) Staffing (SO-1)	Appendix G - Section G.2.2
8) Aging management program (SO-1)	Appendix G - Section G.2.2
9) Surveillance, including materials surveillance program (SO-3, SO-7, PS-1, PS-4, BI-4)	Appendix G - Section G.2.2
10) ISI (SO-1, SO-3, PS-4, BI-4)	Appendix G - Section G.2.2
11) Testing (SO-1, SO-3, PS-4, BI-4, PA-2)	Appendix G - Section G.2.2
12) Technical specifications, including environmental (SO-5, SO-6, PS-1, PS-4, PS-5, BI-5, BI-6)	Appendix G - Section G.2.2
13) Develop EOP and AM procedures integral with design PA-4, PA-9)	Appendix G - Section G.2.5
14) Develop EP integral with design (PA-5, PA-6)	Appendix G - Section G.2.5
15) Monitoring and feedback (SO-1, SO-6, SO-7, PS-1, PS-5, BI-5)	Appendix G - Section G.2.2

Table G-6 Technical Topics for technology-neutral requirements.

Topic	Framework Technical Description
16) Work and configuration control (SO-5, BI-4, PS-4)	Appendix G - Section G.2.2
17) Living PRA (SO-1, PS-1)	Chapter 7
18) Maintain fuel and replacement part quality (SO-6, PS-3, BI-3)	Appendix G - Section G.2.2
19) Security (See B above)	Appendix G - Section G.2.1

Table G-7 Defense-in-Depth (DID) provisions.

DID Principle	Physical Protection	Stable Operation	Protective Systems	Barrier Integrity	Protective Actions
1) Consider intentional and inadvertent events	Integral Design Process	Integral Design Process	Integral Design Process	Integral Design Process	Integral Design Process
2) Consider prevention and mitigation in design	Security Assessment	Applicant should propose cumulative limit on IE frequencies.	Accident prevention and mitigation:	Accident prevention and mitigation:	Develop EOPs and AM integral with design EP
3) Not dependent upon a single element of design, construction, maintenance, operation	Security Assessment	Ensure events that can fail multiple PS are $<10^{-7}$ /plant year.	Provide 2 independent, redundant diverse means for: reactor shutdown and DHR.	Provide at least 2 barriers:	No key safety function dependent upon a single human action or piece of hardware
4) Account for uncertainties in performance and provide safety margins	Security Assessment and Consideration of Beyond DBTs	Reliability Assurance Program (RAP). Provide safety margins in performance limits.	Applicant to propose reliability and availability goals and RAP. Provide safety margin in regulatory limits.	Provide containment functional capability independent from fuel and RCS. Provide safety margin in regulatory limits.	EP Use 95% ST in calculations for safety margin.
5) Prevent unacceptable release of radioactive material	Security Assessment	Ensure events that can fail (stable oper, PS and BI) PS are $<10^{-7}$ / plant year.	N / A	Provide containment functional capability independent from fuel and RCS	AM
6) Siting	Security Assessment	Applicant should propose limits on ext. event cumulative frequencies.	N / A	N / A	EP

N/A = Not applicable

G.3 Administrative Requirement Topics

As discussed earlier in this document, the framework is to define the scope and content, and provide the overall technical basis for a new part to 10 CFR containing technology-neutral, risk-informed and performance-based requirements for new plant licensing which can serve as an alternative to 10 CFR 50. Accordingly, as an alternative to 10 CFR 50, the new part should address the administrative aspects of licensing using the new process, similar to the administrative aspects of 10 CFR 50. Where possible, existing administrative requirements should be used provided they are technology-neutral. However, the administrative aspects of this new part will have some differences from those in 10 CFR 50 because of the technology-neutral, risk-informed and performance-based nature of the new part. In either case, the administrative requirements need to be complete, so as to make the technology-neutral set of requirements a stand alone alternative to 10 CFR 50.

Administrative requirements have an impact on safety in that they define processes, documentation and practices that are necessary to ensure accurate and adequate information is developed, maintained and reviewed such that there is assurance that the plant is designed, constructed, operated and maintained in accordance with the safety analysis. The administrative requirements also ensure sufficient information is provided to the regulator to allow independent verification of plant safety. In effect, this serves as an administrative defense-in-depth measure by providing an independent check on plant safety.

Figure G-6 is a logic tree that illustrates schematically the various elements of administration whose failure could impact safety. Each of the branches on the tree is discussed below with respect to identifying what must be done to ensure success of the branch. This then leads to identifying what topics the administrative requirements must address to be complete. Table G-8 then provides the questions resulting from Figure G-6, the answers to which identify the topics that need to be addressed by the administrative requirements.

The first branch on the tree is associated with ensuring that the information necessary for licensing decisions is adequate. The licensing decisions that require information are:

- the initial application to build and operate a nuclear power plant;
- any amendments to the license after the initial OL is granted; and
- any exemptions to the regulations for initial licensing or subsequent amendments.

Each of these licensing actions requires certain types of information which the administrative requirements should address. However, due to the risk-informed and performance-based nature of the requirements, where PRA information will play a central role in establishing the safety case, the types of information required for each of these decisions will be different that what is required under 10 CFR 50. In developing the requirements, such information needs will need to be defined.

Issues that will need to be addressed include:

- What information from the PRA should be part of the initial application, license amendment requests and exemption requests? (See Chapter 7 for guidance.)

- What level of design, construction and operational detail needs to be submitted?

- What supporting research and development information needs to be submitted?

The second branch on the tree relates to maintaining the plant configuration up to date. This would include having a change control process that requires adequate review and approval of proposed changes and clearly identifies what changes require NRC approval and which do not. Since the regulatory structure for new plant licensing makes use of a living PRA, the selection of basis events (and the selection of SSCs for special treatment) may not be a one-time licensing step, carried out at the time of plant licensing and remaining fixed. (See Chapter 6 for a description of the selection process.) Instead, it can be expected that both the selection of LBEs and the safety classification of SSCs may change over the lifetime of the plant as operational experience and other new information add to, and reshape, the risk insights from the living PRA. This potential for change in the LBEs and safety classification over time, due to the use of a living PRA, has to be addressed. The frequency and manner of updating the living PRA will have to be determined in a way that allows for regulatory stability and predictability, including compatibility with the design certification process in 10 CFR 52. Accordingly, the requirement will need to address a process for changes to the licensing basis. It needs to be noted the licensing basis is also dependent on defense-in-depth, therefore, while the risk insights may change, the licensing basis may not necessarily change. Also, if the design has received design certification, the interface between the change control process and the design certification rule-making needs to be defined. To develop a change control process that accommodates the above, the following guidelines are to be used.

- The results of the "living" PRA update should be compared to the plant licensing basis. Where changes in the licensing basis are needed to be consistent with the PRA update, they should be submitted to NRC for approval in a timely fashion.

- For plants built according to a certified design, if any of the proposed changes modify the certified (Tier 1 or Tier 2) portion of the design, a rule change to amend the certification should be processed and backfit considerations used to determine whether other plants of that same design need to make conforming changes.

- All other changes can be made by the licensee, with appropriate documentation available for NRC audit.

Plant configuration can also be affected by inadequate record keeping. This could be due to incomplete or out of date documentation. Requirements for record keeping also need to be established.

The third branch of the tree relates to information and processes necessary for oversight. This will require (1) the licensee to report certain information to NRC (e.g., events, inspection results, performance indicators, etc.) in an accurate and timely fashion, (2) the licensee to monitor certain aspects of plant performance and take corrective action (via design or operation) when necessary and (3) the NRC to initiate enforcement or backfit action if licensee performance or action is judged inadequate. Requirements addressing what is expected from the licensee and what will trigger NRC actions will be necessary.

Table G-9 summarizes the topics which the administrative requirements need to address based on the above. Other administrative requirements not related to safety will also be needed and these can be identified by a careful review of 10 CFR 50 and by including the appropriate requirement from 10 CFR 50 in the technology-neutral requirements, provided it is technology-neutral.

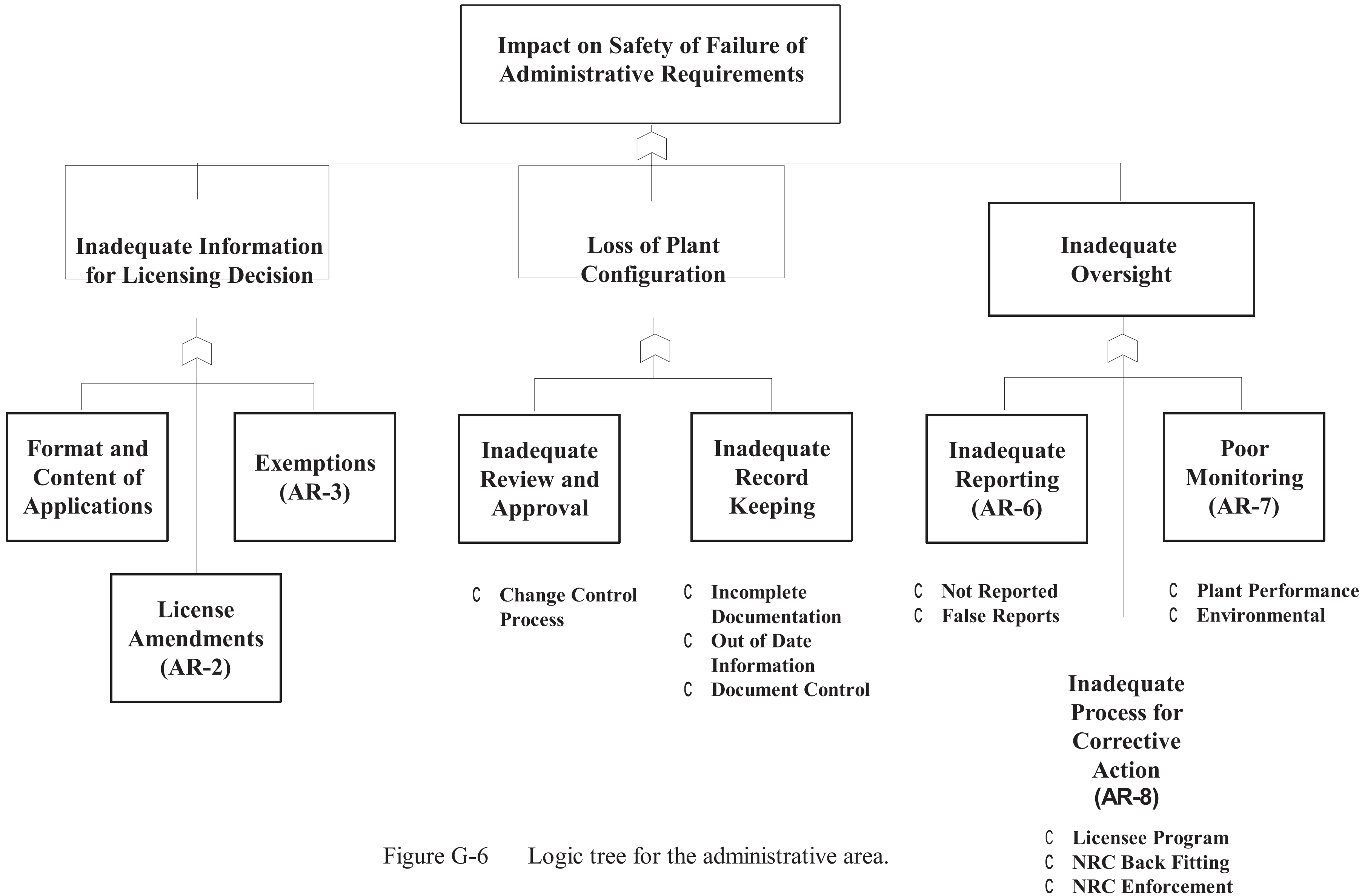

Figure G-6 Logic tree for the administrative area.

Table G-8 Administrative requirement topics.

Questions	Topics to be Addressed in the Requirements		
	Design	Construction	Operation
Inadequate Information for Licensing Decisions			
• What information needs to be submitted to support initial licensing? (AR-1)	• Standard format and content of applications	• Standard format and content of applications	• Standard format and content of applications
• What in formation needs to be submitted to support license amendments? (AR-2)	• N / A	• N / A	• Standard format and content of applications
• What information needs to be submitted to support exemptions? (AR-3)	• Standard format and content of applications	• Standard format and content of applications	• Standard format and content of applications
Loss of Plant Configuration			
• What is needed to ensure appropriate review and approval of plant changes? (AR-4)	• Change control process	• Change control process	• Change control process
• What information needs to be maintained? (AR-5)	• Identify documentation to be maintained (i.e., recordkeeping) • Documentation control process	• Identify documentation to be maintained (i.e., recordkeeping) • Documentation control process	• Identify documentation to be maintained (i.e., recordkeeping) • Documentation control process
Inadequate Oversight			
• What information is needed to support NRC oversight? (AR-6)	• N / A	• Reporting requirements	• Reporting Requirements
• What information is the licensee expected to monitor? (AR-7)	• N / A	• Inspection • Testing	• Plant performance • Environmental Releases
• What corrective action processes are needed? (AR-8)	• N / A	• Licensee program • NRC enforcement	• Licensee program • NRC enforcement • NRC backfitting

N / A = Not Applicable

Table G-9 Administrative topics for Technology-Neutral Requirements.

TOPIC	FRAMEWORK DESCRIPTION
• Standard format and content of application (AR-1)	• Appendix G - Section G.3
• Change control process (AR-4)	• Appendix G - Section G.3
• Record keeping (AR-5)	• Appendix G - Section G.3
• Documentation control (AR-5)	• Appendix G - Section G.3
• Reporting (AR-6)	• Appendix G - Section G.3
• Monitoring and feedback (AR-7): - plant performance - environmental releases - testing results	• Appendix G - Section G.3
• Corrective action program (AR-8)	• Appendix G - Section G.3
• Backfitting (AR-8)	• Appendix G - Section G.3
• License Amendments (AR-2)	• Appendix G - Section G.3
• Exemptions (AR-3)	• Appendix G - Section G.3
• Other legal and process items (e.g.) - anti-thrust - termination of license - etc.	• Appendix G - Section G.3 and Appendix H

H. APPLICABILITY OF 10 CFR 50

As discussed in Chapter 8, the development of technology-neutral requirements should build upon previous work as much as possible. Accordingly, 10 CFR 50 needs to be reviewed to see where it would be appropriate to directly carry over its requirements into the proposed 10 CFR 53. Two main areas where this would appear to be appropriate are:

- those legal, financial and process requirements that are technology-neutral and were not identified by the technical considerations discussed in Chapter 8 and Appendix G, and

- those technical requirements that are currently technology-neutral.

Any initial assessment of 10 CFR 50 has been made to identify where 10 CFR 50 requirements can be used directly in the proposed 10 CFR 53. The results of this assessment are shown in Table H-1. As can be seen from Table H-1, there are many 10 CFR 50 requirements that are candidates for inclusion in 10 CFR 53.

Table H-1 Initial assessment of applicability of 10 CFR 50 requirements.

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

Table H-1 Initial assessment of applicability of 10 CFR 50 requirements.

US 10 CFR Part 50		Technology Neutral Framework
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
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

Table H-1 Initial assessment of applicability of 10 CFR 50 requirements.

US 10 CFR Part 50	Technology Neutral Framework
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
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

Table H-1 Initial assessment of applicability of 10 CFR 50 requirements.

US 10 CFR Part 50	Technology Neutral Framework
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
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

Table H-1 Initial assessment of applicability of 10 CFR 50 requirements.

US 10 CFR Part 50	Technology Neutral Framework
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 Addendum to Table H-1
6. Owner/Management Competency and Fitness Requirements	
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 H-1 Initial assessment of applicability of 10 CFR 50 requirements.

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

Table H-1 Initial assessment of applicability of 10 CFR 50 requirements.

US 10 CFR Part 50	Technology Neutral Framework
Appendix B: Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants	• Consider use of 10 CFR 50 words
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
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

Table H-1 Initial assessment of applicability of 10 CFR 50 requirements.

US 10 CFR Part 50	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
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

Table H-1 Initial assessment of applicability of 10 CFR 50 requirements.

US 10 CFR Part 50		Technology Neutral Framework
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
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

Addendum to Table H-1

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

The following GDCs are considered technology-neutral and candidates for inclusion into 10 CFR 53. All other GDCs are considered LWR specific or need some other modification if they are to be transferred to 10 CFR 53. Appendix K provides additional information on where GDCs address topics similar to those identified by the process described in Chapter 8.

GDC 1: Quality Standards and Records
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 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 62: Prevention of Criticality in Fuel Storage and Handling
GDC 63: Monitoring Fuel and Waste Storage

I. 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 5). An example of a typical performance objective is maintaining cladding integrity. In the conventional regulatory approach this objective is considered to be accomplished through a prescriptive approach of limiting cladding temperature and oxidation conditions to 2200 F and 17% respectively. In a performance-based approach, a different set of criteria, perhaps using a combination of qualitative and quantitative may be found to better fulfill the high-level guidelines.

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

Purpose – To define a performance objective for the SSC and/or operator actions in such a way that one or more performance measures and criteria can be proposed for consideration.

Step 1a: What is the topic area with which the performance objective is associated?

This question is likely addressed during the review under Chapter 8, where the risk objectives are classified as falling under design, construction and operation. Additionally, from a regulatory standpoint, the objectives may fall under the categories public risk, worker risk and environmental risk. There could be significant differences in the information gathering and stakeholder identification depending on what is being addressed. A well defined performance objective is a pre-requisite for an effective performance measure. If a single performance objective will not be effective for establishing the requirements for the SSC, an Objectives Hierarchy (see NUREG/BR-0303) may need to be prepared.

Step 1b: Which of the NRC's performance goals does the performance objective address?

Clarifying the performance goal also improves the clarity with which NRC decision preferences may be incorporated in the consideration of performance measures or criteria. From the NRC's Strategic Plan (NUREG-1614, Vol. 3, August 2004) the two performance goals likely to be involved are "*Ensure protection of public health and safety and the environment*" and "*Ensure that NRC actions are effective, efficient, realistic, and timely*".

Step 1c: What are the expected outcomes and results from successful performance relative to the objective?

In general, the expected outcome is that the SSC performs its intended safety function adequately, and that the performance can be appropriately verified through regulatory oversight. In addition, this question addresses which part of the regulatory structure is appropriate for implementing the objective. In general, a regulation in the Code of Federal Regulations is likely to address higher level goals or objectives. Guidance documents are more likely to be directed at detailed or component level objectives.

I.2 Step 2 – Identifying the Safety Functions

Purpose – To identify the safety functions and systems that affect the performance objective (directly or indirectly).

Step 2a: What are the safety functions or concepts that can impact the performance objective?

The objective of this inquiry is to identify the most important functions. The PRA should be of help in this effort. However, some aspects of system performance may not be modeled in the PRA. Such aspects are generally those that cannot be easily quantified and must be considered qualitatively. It is key that the identification of important functions focus on successful outcomes rather than make assumptions because of inadequacies of the PRA model. In addition, consideration should be given to other aspects of the context which may include expected outcomes being fulfilled by other SSCs.

Step 2b: What equipment/systems/procedures are necessary to satisfy the safety function?

This addresses the technical evaluation that establishes the range of particular SSCs or support systems to be considered; for example, instrumentation, siting, safety conscious work environment, etc. Again, the evaluation can take advantage of the PRA where the modeling is adequate. Often, qualitative factors coupled with expert judgement can be as or more reliable than quantitative models that are not supported by sufficient data. This is especially the case when data from operating experience exists, even if the data is from a related but different industry.

Step 2c: What level of safety (based on appropriate metrics) is required to meet the performance objective?

This addresses the required level of safety that should have been addressed in the Chapter 6 evaluation. For example, the required level of safety for an accident within containment might be one that meets the objective of reducing, to an acceptable level, the risk of early containment failure. Hence, the metric in this case is the conditional containment failure probability. Another example might be that the required level of safety is to maintain at an acceptable level the core damage risk associated with certain configurations typical of specific modes of operations. Again, qualitative evaluations supported by expert judgement or operational data may be required.

I.3 Step 3 – Identifying Safety Margins

Purpose – To evaluate margins and identify performance measures (if any) that satisfy the performance objectives.

Step 3a: How much safety margin is available, and how robust is it, for performance monitoring to provide a basis for granting licensee flexibility?

The generic definition of a "margin" is that it is an expression of a difference between two system states. When the two states are associated with different levels of safety as reflected in the above evaluations related to outcomes, the "margin" becomes a safety margin. For regulatory purposes, the margin that is sought to be maintained is expressed by the first of these being the expected state and the other is one where a regulatory concern exists. The state of regulatory concern can be drawn from the frequency-consequence curve dealt with in Chapter 6 and the margin discussion in Chapter 6.

"Robustness" of a safety margin means that the margin between two performance levels is significantly greater than uncertainty and normal variability in performance. If this condition is met, a very low probability exists of the performance parameter crossing a set limit, unless performance changes in a very significant way. In any case, wherever there is substantial uncertainty, achieving robustness requires that nominal performance levels be set more conservatively than when there is less uncertainty. Depending on the situation, uncertainty can be assessed using explicit models (e.g., PRAs), expert judgment, or actuarial methods based on operating experience.

The identification of performance measures (natural, constructed or combination) begins as a search process within the overall context of the performance objective. It is likely to involve iteration through the steps in this guidance as well as consideration of the factors that were involved in the application of the viability guidelines. The flexibility aspects should include operational flexibility as well as the means to fulfill regulatory responsibilities.

Step3b: What observable characteristics, quantitative and qualitative, exist within the safety functions identified in Step 2?

For example, observable characteristics may come from the results of periodic servicing, testing, and calibration of certain instruments. The operating margin would be based on a comparison between these results and the target values established under a maintenance program. Another example would be observations based on verification (through testing) of design margins of structures.

Step 3c: Can the contemplated constructed measures provide qualitative expressions capable of observation with reasonable objectivity?

As explained in NUREG/BR-0303, natural measures are preferred, but appropriate constructed measures may also prove adequate with proper consideration given to verification and validation. In some cases, a binary constructed measure might well suffice where the measure reflects a positive or negative response to a question such as , “Does a particular attribute exist?”

I.4 Step 4 – Selecting Performance Measures and Criteria

Purpose – To select a complement of performance measures and objective criteria (if possible) that both satisfy the viability guidelines and accomplish the performance objective.

Step 4a: Can the identified observable characteristics, together with objective criteria, provide measures of safety performance and the opportunity to take corrective action if performance is lacking?

This step is a part of the search process. Many technically significant performance objectives will require engineering judgement for exploring qualitative and/or quantitative measures while keeping in mind operational (or other) constraints. Measures of safety performance considered as candidates should be associated with the desired outcomes as directly as possible. Sometimes, it may prove quite effective to use proxy measures. For example, if the accomplishment of a performance objective calls for an analysis, the cost of the analysis may be one of the measures considered as a proxy for efficiency of obtaining the outcome.

Another of the highly desirable features of a good performance measure is that it should be identified at as high a level as practicable. If this feature is not sought, all systems and sub-systems involved in, say, risk-significant configurations might have been targeted for monitoring. The management of risk when various configurations are being considered may include monitoring strategies that target all systems and sub-systems, or a higher-level measure that may prove to be simpler, but as effective. The process of searching for parameters at a high level directs the 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 and expert elicitation.

Step 4c: Is flexibility (for NRC and licensees) available consistent with level of margin?

The approach of setting criteria at as high a level as practicable can allow more flexibility. The benefits of flexibility must be balanced against assurance of opportunity to take appropriate corrective action and practicality of regulatory oversight. The basic principle involved is that more flexibility can be justified by higher levels and robustness of safety margin. Again, an iterative approach may be most suitable for optimum results. This is because questions of margin, corrective action, and flexibility strongly interact with one another. Strong linkages can exist between observable characteristics chosen as the performance measures to be used in a performance-based approach and the assessment of margin based on criteria applied to these parameters. For example, in the area of quality assurance, the quality of emergency backup power provided by a diesel generator would not necessarily be well-reflected just by the criteria that are applied to each component part of the diesel generator. Even if very strict quality criteria are applied to each of the component parts, the overall diesel generator performance may not meet regulatory standards. On the other hand, a diesel generator could adequately meet performance standards even if the component parts are only commercial grade.

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

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

Step 5a: Does the performance-based regulatory requirement provide necessary and sufficient coverage for the performance objective?

One of the important elements of coverage is consideration of defense-in-depth. As described in Chapters 4 and 8, NRC's defense-in-depth philosophy includes consideration of "prevention" and "mitigation" strategies which should operate in proper balance. Such considerations may require the use of more complex approaches based on decision theoretic concepts (also described in NUREG/BR-0303).

Step 5b: Of the performance parameters selected in Step 4, which of them requires that a prescriptive approach be used to meet regulatory needs? Can a combination of performance-based and prescriptive measures be implemented such that the resolution of the regulatory issue is as performance-based as possible?

The search process for performance measures and criteria may reveal various permutations and combinations of prescriptive, less-prescriptive and performance-based strategies for individual components or sub-systems. In some cases, specific prescriptive elements can be incorporated into a less prescriptive regulatory approach. The regulatory structure permits inclusion of prescriptive elements through Technical Specification or License Condition provisions.

Step 5c: Has the regulatory alternative been considered for implementation within each of the levels of the regulatory structure so that an optimum level is proposed?

For example, a prescribed parameter can be included in a Technical Specification or other license condition. It may be possible to provide flexibility in operation for parameters that do not have to be strictly controlled. Also, consideration should be given to incentives for licensees to increase the likelihood of improved safety outcomes.

Step 5d: Are licensees' incentives appropriately aligned, considering the overall complement of performance measures, criteria, the implementation, and the regulatory structure as a whole?

Licensees' flexibility can be coupled with positive and negative incentives. Examples of positive incentives occur when licensees may be able to reduce costs of operation if they meet specified levels of safety or trends in safety of operation. Examples of negative incentives occur when the enforcement policy may cause undesired consequences for the licensee when levels of safety or trends in safety are unfavorable.

Regulation that is based on sampling licensee performance needs to be designed with care, in order to avoid incentivizing performance in one important area at the expense of another, with a net adverse outcome. As a hypothetical example, regulation that sought only to minimize the unavailability of components might create an incentive to reduce maintenance to a level at which unreliability performance would be adversely affected. The regulatory structure itself should be subjected to critical scrutiny for inappropriate incentives.

Step 5e: Is it worth modifying the regulatory structure in the manner proposed, considering the particulars of the regulatory issue?

Among the high-level performance-based guidelines, the assessment guidelines are best suited to make this evaluation. A feedback process involving a wide range of stakeholders may be the most effective way to develop the required information. Such a process may explicitly consider the cost impacts of incorporating requirements in one or other part of the regulatory structure.

J. EXAMPLE REQUIREMENTS

Chapter 8 described the process for taking the structure principles and criteria described in Chapters 2 through 7 and identifying the topics for which requirements need to be written. The application of this process is described in Appendix G and the list of identified topics shown in Table 8-1. The next step is to take these topics from Table 8-1 and develop requirements following the guidance in Chapter 8. The purpose of this appendix is to provide example requirements for some of the topics in Table 8-1 to illustrate the scope, depth and level of detail envisioned in the requirements. The examples chosen are intended to illustrate requirements for the design, construction, operation and administrative areas that could be applicable to any plant design, including those that will likely need technology-specific guidance for implementation, as well as those that will not. Listed below are example requirements.

<u>**Example Requirements Related to Good Design Practices**</u>

- <u>Topic 1 - Plant Risk</u>

Each application to construct and operate a NPP shall include a probabilistic risk assessment that:

(1) includes the risk from full power and low power operation, shutdown, refueling and spent fuel storage (except that from dry cask storage)

(2) includes assessment of internal and external events and uncertainties

(3) shows each accident sequence in the PRA meets the appropriate dose limit on the F-C curve at its mean value

(4) shows overall risk from the NPP (or if more than one NPP from all NPPs on site) meets the QHOs expressed in the Commission's 1986 Safety Goal Policy using mean risk values

(Technology-specific guidance will likely not be required.)

- <u>Topic 3 - LBE Acceptance Criteria</u>

Events selected as licensing basis events (LBEs) shall meet the following acceptance criteria:

(1) LBEs in the frequent category shall:
   (a) not exceed an annualized dose of 100 mrem/yr, at the 95% confidence level
   (b) not result in any fuel damage (no additional release of fission products or fuel and no loss of fuel lifetime)
   (c) not result in any additional barrier failure, beyond the initiating event.

(2) LBEs in the infrequent category shall:
   (a) not exceed the dose criteria represented by the F-C curve in the infrequent frequency range on a per event basis, at the 95% confidence level
   (b) not result in loss of coolable core geometry (no fuel melting or other condition such as fuel temperature that could result in the uncontrolled movement of fission products and/or fuel from their intended location)
   (c) not result in the loss of all barriers to the release of fission products or other radioactive material to the environment

(3) LBEs in the rare category shall not exceed the dose criteria represented by the F-C curve in the rare frequency range on a per event basis, at the 95% confidence level

(Technology-specific guidance will likely be required to define “fuel damage” and “coolable geometry.”)

- Topic 9 - Use of Consensus Codes and Standards

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. Each code or standard used in the design must be submitted to NRC for review.

(Technology-specific guidance will be needed to specify acceptable codes and standards.)

- Topic 16 - Reactor Inherent Protection

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 power and core 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).

(Technology-specific guidance will likely not be needed.)

Example Requirements Related to Good Construction Practices

- Topic 4 - Inspection

During construction, accepted inspection techniques shall be used to verify safety-significant SSCs are installed according to design.

(Technology-specific guidance will be needed to identify acceptable inspection techniques.)

Example Requirements Related to Good Operating Practices

- Topic 5 - Use of Procedures

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 mockups. Procedures shall be controlled and maintained up to date.

(Technology-specific guidance will likely not be needed.)

- Topic 10 - In-Service Inspection

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.

(Technology-specific guidance will likely be needed to identify acceptable ISI techniques.)

Example Requirements Related to Administrative Items

- Topic 6 - Monitoring and Feedback

Each licensee shall establish and maintain a monitoring program to:

(a) determine the reliability and availability of all safety significant equipment. This information shall be periodically fed back into the licensing analysis so as to maintain the licensing analysis up to date. This information shall also be compared to the reliability and availability goals established during design and, where these goals are not met, corrective action shall be taken.

(b) measure the release of radioactive material to the environment from normal operation, frequent and infrequent events. This information shall be compared to established limits and corrective action taken when limits are exceeded.

(Technology-specific guidance will likely not be needed.)

The above example requirements are for illustration purposes only and are subject to change as comments are received on the framework and as work to develop a complete set of requirements continues.

K. COMPLETENESS CHECK

K.1 Introduction

As described in Chapter 8, a top down process has been used to identify the topics for which requirements are needed to have a stand alone 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 8-1, which lists the technical topics which the technology-neutral requirements must address.

A similar process was followed for the administrative requirements, as described in Appendix G 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 also shown in Table 8-1.

To help ensure that the list of technical and administrative topics shown in Table 8-1 is complete, a check was made against other documents containing requirements for reactor safety. Specifically, the following documents were compared against Table 8-1:

- 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”

This Appendix documents the results of the completeness check.

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

K.2 Comparison Against 10 CFR 50

Table K-1 shows the results of the comparison against 10 CFR 50. Table K-1 addresses all requirements in 10 CFR 50. Table K-1 (and Table K-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 8-1.

For the administrative topics, Table 8-1 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.

Table K-1 10 CFR 50 comparison.

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	• 10 CFR 50 requirement needs to be included.
50.2	Definitions	• Review for applicability.
50.3	Interpretation Assigns legal interpretation authority to NRC General Counsel	• 10 CFR 50 requirement needs to be included.
2. Oversight/Enforcement		
50.7	Employment Protection Protects employees of licensees against discrimination and retribution for providing information to NRC, Congress, etc.	• 10 CFR 50 requirement needs to be included.
50.8	Information Collection Requirements Requires NRC to submit information collection requirements to OMB for approval to collect the information	• 10 CFR 50 requirement needs to be included.
50.9	Completeness Requirements	• 10 CFR 50 requirement needs to be included.
50.10	License Requirements (Construction and Operation) Establishes license requirement Identifies facilities which are required to obtain an NRC license and which are not	• 10 CFR 50 requirement needs to be included.
50.11	Exceptions and Exemptions from License Requirements	• 10 CFR 50 requirement needs to be included.
50.12	Specific Exemptions	• Included
50.35	Issuance of Construction Permits	• 10 CFR 50 requirement needs to be included.
50.39	Public Inspection of License Requirement	• 10 CFR 50 requirement needs to be included.
50.50	Issuance of Licenses and Construction Permits Technical Specifications, Conditions, and Limitations	• 10 CFR 50 requirement needs to be included.
50.51	Continuation of License Set time limits on term of license Holds licensee responsible for site after permanent shutdown	• 10 CFR 50 requirement needs to be included.

Table K-1 10 CFR 50 comparison.

US 10 CFR Part 50		Technology Neutral Framework
50.53	Jurisdictional Limits	• 10 CFR 50 requirement needs to be included.
50.58	Publishing and Hearing Requirements to Issue Construction Permits	• 10 CFR 50 requirement needs to be included.
50.76	Licensee Change of Status, Financial Qualifications Requires licensee to inform NRC 75 days before ceasing to exist	• 10 CFR 50 requirement needs to be included.
50.78	Installation information and verification Requires licensees to submit to IAEA inspection when directed by NRC	• 10 CFR 50 requirement needs to be included.
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	• 10 CFR 50 requirement needs to be included.
50.90	Application for Amendment of License or Construction Permit	• Included
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	• 10 CFR 50 requirement needs to be included.
50.92	Issuance of Amendments Identifies issues which are to be considered when evaluating a request for a license change	• Included
50.100	Revocation, Suspension, and Modification of Licenses and Construction Permits	• 10 CFR 50 requirement needs to be included.
50.101	Retaking Possession of Special Nuclear Fuel The NRC may retake fuel upon revocation of license.	• 10 CFR 50 requirement needs to be included.
50.102	Commission Orders for Operation After Revocation Allows Commission to require a plant to be operated after licenses have been revoked	• 10 CFR 50 requirement needs to be included.
50.103	Suspension and Operation in War or National Emergency	• 10 CFR 50 requirement needs to be included.
50.110	Violations Grants power to NRC to seek injunction for violations of Atomic Energy Act, NRC regulations, or violations of License	• 10 CFR 50 requirement needs to be included.
50.111	Criminal Penalties	• 10 CFR 50 requirement needs to be included.
3. Management Requirements/Confidence		

Table K-1 10 CFR 50 comparison.

US 10 CFR Part 50	Technology Neutral Framework
50.30 Filing Procedure, Oath or Affirmation	• 10 CFR 50 requirement needs to be included.
50.33a Anti Trust Limitation	• 10 CFR 50 requirement needs to be included.
50.40 Common Standards Compliance requirement Requirement for licensee to be technically and financially qualified Operation does not infringe on defense or public health	• 10 CFR 50 requirement needs to be included.
50.81 Creditor Regulations Sets conditions under which a creditor may posses a lien on a utilization and production facility	• 10 CFR 50 requirement needs to be included.
Appendix C: A Guide for the Financial Data and Related Information Required to Establish Financial Qualifications for Facility Construction Permits	• 10 CFR 50 requirement needs to be included.
Appendix L: Information Requested by the Attorney General for Antitrust Review of Facility Construction Permits and Initial Operating Licenses	• 10 CFR 50 requirement needs to be included.
4. Tracking and Records Schema/Requirements	
50.4 Written Communications Communication Delivery Requirements and Procedures Distribution Requirements Communication Requirements Required Submissions	• 10 CFR 50 requirement needs to be included.
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 to technology-neutral framework.
50.22 Class 103 License Commercial and industrial license	• 10 CFR 50 requirement needs to be included.
50.23 Construction Permits	• 10 CFR 50 requirement needs to be included.
50.31 Allowance for Combining Applications	• 10 CFR 50 requirement needs to be included.
50.32 Elimination of Repetition	• 10 CFR 50 requirement needs to be included.
50.33 Contents of Application (General Requirements)	• Included
50.41 Additional Standards for Class 104 License	• Not applicable to technology-neutral framework.

Table K-1 10 CFR 50 comparison.

US 10 CFR Part 50	Technology Neutral Framework
50.42 Additional Standards for Class 103 License Usefulness Requirement Antitrust Restriction Open Communication Requirement	• 10 CFR 50 requirement needs to be included.
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	• 10 CFR 50 requirement needs to be included.
50.70 Inspections Requires licensees to submit to routine inspection Requires licensee to provide reasonable space accommodation to inspectors	• 10 CFR 50 requirement needs to be included.
50.71 Maintenance of Records, Making Reports Defines items which must be records Sets requirements for quality of records Sets reporting periods for specific records	• Included
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	• 10 CFR 50 requirement needs to be addressed.
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	• 10 CFR 50 requirement needs to be included.
50.75 Reporting and Record Keeping for Decommissioning Planning Establishes reasonable assurance that funds will be available for decommissioning process	• 10 CFR 50 requirement needs to be included.
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	• 10 CFR 50 requirement needs to be included.
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.

Table K-1 10 CFR 50 comparison.

US 10 CFR Part 50	Technology Neutral Framework
5. Safety Objectives	
Appendix A: General Design Criteria for Nuclear Power Plants	• See Addendum to Table K-1
6. Owner/Management Competency and Fitness Requirements	
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	• 10 CFR 50 requirement needs to be included.
7. Confidence in Personnel	
50.5 Deliberate Misconduct	• 10 CFR 50 requirement needs to be included.
50.74 Notification of Change in Operator or Senior Operator Status Reporting Requirement	• 10 CFR 50 requirement needs to be included.
50.120 Training and Qualification of Nuclear Power Plant Personnel Requirement to have a training program Training program standards Personnel required to receive training Training review and update requirements	• Included
8. Confidence in Engineering	
50.34 Contents of Application (Technical Requirements)	• Included
50.36 Technical Specifications	• Included
50.45 Standards for Construction Permits	• 10 CFR 50 requirement needs to be included.

Table K-1 10 CFR 50 comparison.

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	• Technical items addressed. Others need to be included using 10 CFR 50 requirements. • Drop non-power reactor requirements.
50.55a	Codes and Standards Sets minimum standards commensurate with safety Identifies ASME Standards as minimums Sets Minimum Requirements for Specific Structural Materials	• Included
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	• Included
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	• Included
50.109	Backfitting Definition of Backfitting Conditions to Require Backfitting	• Included
Appendix B:	Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants	• Included

Table K-1 10 CFR 50 comparison.

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	• Included
50.48 Fire Protection General Description Specific Hazard Detection and Suppression Systems Administrative Controls Risk-informed Analysis Requirement	• Included
50.49 Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants	• Included
50.59 Changes, Tests, and Experiments Definitions of Changes, Tests, and Experiments Definition of Scope Reporting Requirements of Changes, Tests, and Experiments	• Included
Appendix E: Emergency Planning and Preparedness for Production and Utilization Facilities	• Included
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	• Included
50.46 Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Reactors	• Not applicable - LWR specific

Table K-1 10 CFR 50 comparison.

US 10 CFR Part 50	Technology Neutral Framework
50.46a Acceptance Criteria for Reactor Coolant System Venting System	• Not applicable - LWR specific
50.60 Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation	• Included
50.61 Fracture toughness requirements for protection against pressurized thermal shock events	• Included
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	• Included
Appendix G: Fracture Toughness Requirements	• Included
Appendix H: Reactor Vessel Material Surveillance Program Requirements	• Included
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	• 10 CFR 50 requirement needs to be included.
11. Security of Material and Facilities Requirements	
50.13 Requirement for Security Requires licensees to maintain security against foreign enemies and domestic criminals	• Included
50.37 Agreement Limiting Access to Classified Information	• 10 CFR 50 requirement needs to be included.
50.38 Foreign Corporation or Individual Restriction	• 10 CFR 50 requirement needs to be included.
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	• Included

Table K-1 10 CFR 50 comparison.

US 10 CFR Part 50	Technology Neutral Framework
50.36a Technical Specifications on Effluent from Nuclear Power Plants	• Included
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	• Included
13. Regulation Burden Mitigation	
50.52 Combining Licenses	• 10 CFR 50 requirement needs to be included.
50.56 License Conversion	• 10 CFR 50 requirement needs to be included.
50.57 Issuance of Operating License Requirements to issue an operating license	• 10 CFR 50 requirement needs to be included.
50.80 Transfer of Licenses Requires NRC to consent to license transfer to qualified licenses Defines requirements for new licensee to receive license	• 10 CFR 50 requirement needs to be included.
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	• Included

Addendum to Table K-1

<u>10 CFR 50, Appendix A - General Design Criteria (GDC)</u>

General Design Criteria	Technology-Neutral Framework
1. Quality Standards and Records	included
2. Design Bases for Protection Against Natural Phenomena	included
3. Fire Protection	included
4. Environmental and Dynamic Effects Design Bases	included
5. Sharing of Structures, Systems and Components	not included (design specific)
10. Reactor Design	included
11. Reactor Inherent Protection	included
12. Suppression of Reactor Power Oscillations	included
13. Instrumentation and Control	included
14. Reactor Coolant Pressure Boundary	included
15. Reactor Coolant System Design	included
16. Containment Design	included
17. Electric Power Systems	not included (design specific)
18. Inspection and Testing of Electric Power Systems	not included (design specific)
19. Control Room	included
20. Protection System Functions	included
21. Protection System Reliability and Testability	included
22. Protection System Independence	included
23. Protection System Failure Modes	included
24. Separation of Protection and Control Systems	not included (design specific)
25. Protection System Requirements for Reactivity Control Malfunctions	included

General Design Criteria	Technology-Neutral Framework
26. Reactivity Control System Redundancy and Capability	included
27. Combined Reactivity Control System Capability	included
28. Reactivity Limits	included
29. Protection Against AOOs	included
30. Quality of Reactor Coolant Pressure Boundary	included
31. Fracture Prevention of Reactor Coolant Pressure Boundary	included
32. Inspection of Reactor Coolant Pressure Boundary	included
33. Reactor Coolant Makeup	not included - LWR specific
34. Residual Heat Removal	included
35. Emergency Core Cooling	not included - LWR specific
36. Inspection of Emergency Core Cooling System	not included - LWR specific
37. Testing of Emergency Core Cooling System	not included - LWR specific
38. Containment Heat Removal	not included (design specific)
39. Inspection of Containment Heat Removal System	not included (design specific)
40. Testing of Containment Heat Removal System	not included (design specific)
41. Containment Atmosphere Cleanup	not included (design specific)
42. Inspection of Containment Atmosphere Cleanup System	not included (design specific)
43. Testing of Containment Atmosphere Cleanup System	not included (design specific)
44. Cooling Water	not included (design specific)
45. Inspection of Cooling Water System	not included - LWR specific
46. Testing of Cooling Water System	not included - LWR specific
50. Containment Design Basis	included

General Design Criteria	Technology-Neutral Framework
51. Fracture Prevention of Containment Pressure Boundary	not included - LWR specific
52. Capability for Containment Leakrate Testing	included
53. Provisions for Containment Testing and Inspection	included
54. Piping Systems Penetrating Containment	not included - LWR specific
55. Reactor Coolant Pressure Boundary Penetrating Containment	not included - LWR specific
56. Primary Containment Isolation	included
57. Closed System Isolation Valves	not included - LWR specific
60. Control of Releases of Radioactive Materials to the Environment	included
61. Fuel Storage and Handling and Radioactivity Control	to be added later
62. Prevention of Criticality in Fuel Storage and Handling	included
63. Monitoring Fuel and Waste Storage	included
64. Monitoring Radioactivity Releases	included

K.3 Comparison Against IAEA NS-R-1

Table K-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 K-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.

Table K-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	• Included 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.	• Included in principle
Defense in Depth Level 1: defense to prevent deviations from normal operation, and to prevent system failures Level 2: defense to detect and intercept deviations from normal operational states in order to prevent anticipated operational occurrences from escalating to accident conditions Level 3: Anticipate unlikely escalations in the design basis for the plant and to achieve stable and acceptable plant states following such events Level 4: defense to address severe accidents in which the design basis may be exceeded and to ensure that radioactive releases are kept as low as practical Level 5: mitigation of the radiological consequences of potential releases of radioactive materials that may result from accident conditions	• DID discussed in framework. DID applied in process to identify needed requirements and DID provisions are included in the requirements.
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.	• Included in principle through protective strategies
2. Oversight/Enforcement	

Table K-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 included • Procedures are included • Safety culture is not included
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.	• Included in principle • Included in principle • Included in principle • Not included
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.	• Included in principle • Included in principle • Not included
5. Safety Objectives	
Independent Verification of the Safety Assessment	

Table K-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. Postulated Initiating Events (PIE) produce no significant safety related effect or produce only a change in the plant towards a safe condition by inherent characteristics. Following a PIE, the plant is rendered safe by passive safety features or by the action of safety systems that are continuously operating in the state necessary to control the PIE. Following a PIE, the plant is rendered safe by the action of safety systems that need to be brought into service in response to a PIE. Following a PIE, the plant is rendered safe by specified procedural actions.	• Included in principle • Not included • Not included • Included in principle • Included 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.	• Included 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.	• Included in principle
6. Owner/Management Competency and Fitness Requirements	

Table K-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.	• Included in principle
Operational Experience and Safety Research Design shall take into account relevant operational experience.	• Included 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.	• Included 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.	• Included in principle

Table K-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.	• Included in principle
Emergency Control Center An on-site emergency control center separated from the plant control room shall be provided for use by emergency staff.	• Included in principle
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.	• Included 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.	• Included 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.	• Included in principle

Table K-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.	• Included in principle
Auxiliary Services Auxiliary services supporting safety systems shall be considered part of the safety systems and shall be classified accordingly.	• Included 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.	• Included 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.	• Included in principle
Ageing Appropriate margin shall be provided to incorporate ageing into SSCs designs throughout the design life.	• Included in principle
Interactions of Systems When there is a significant probability that it will be necessary for safety systems to operate simultaneously, possible interaction whether direct or indirectly shall be evaluated.	• Included 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.	• Included 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.	• Included in principle

Table K-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.	• Included in principle
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 is included in the 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.	• Included in principle

Table K-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.	• Included 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 included
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.	• Included 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.	• Included in principle

Table K-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.	• Included 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. Framework includes DID principles and requirements reflect DID provisions.
Categories of Plant States Plant states shall be identified and grouped into a limited number of categories according to their probability of occurrence.	• Included
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.	• Included
Internal Events All those internal events which could affect plant safety shall be identified including: Fires and explosion, and Other internal hazards.	• Included

Table K-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.	• Included in principle
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.	• Included
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.	• Included 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.	• Included in principle
Severe Accidents Certain very low probability events arising due to failure of multiple safety systems which lead to significant core degradation and jeopardize the integrity of many or 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.	• Included 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 included, except in a few key areas (i.e., reactor shutdown, decoy heat removal, barriers). Framework uses PRA
Systems containing fissile and radioactive materials shall be designed to be adequate in operational and design basis accidents.	• Included in principle

Table K-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 included
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 included
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 included - LWR specific
Control and Cleanup of the Containment Atmosphere Systems to control fission products and other substances that may be released into the containment atmosphere. Systems for cleaning up the containment atmosphere shall have suitable redundancy in components and features. Consideration shall be given to the clean up of containment atmosphere during severe accidents.	• Not included - 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.	• Included 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 included

Table K-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.	• Included 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.	• Included 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.	• Included in principle
Specified fuel design limits shall not be exceeded in normal operation and significant occurrences shall not cause further deterioration.	• Included in principle
Fuel assemblies shall be designed to permit adequate inspection of their structure and component parts after irradiation.	• Not included
Requirements shall be maintained in the event fuel management strategy is changed.	• Included 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.	• Included in principle

Table K-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.	• Included
There shall be at least two different systems available to shutdown reactor.	• Included
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.	• Included 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.	• Included 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.	• Included 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.	• Included in principle
In the design of reactivity control devices, account shall be taken of wear-out, and the effects of radiation.	• Included in principle
Reactor Coolant System	
Reactor coolant systems and associated auxiliary systems, controls and protection systems shall be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded in operational states.	• Included in principle
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.	
Inventory Control	
Provisions shall be made for controlling the inventory and pressure of coolant to prevent exceeding specified design limits.	• Included in principle

Table K-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.	• Included 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.	• Included 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.	• Included 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.	• Included 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.	• Included in principle

Table K-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.	• Included in principle
Capability for Containment Pressure Tests Containment shall be designed to allow for pressure testing.	• Included in principle
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.	• Included in principle
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.	• Not included - design specific
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.	• Not included - design specific
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.	• Not included - design specific

Table K-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.	• Not included - design specific
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 included
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.	• Included
Separation of Protection and Control Systems Interface between the protected system and the control systems shall be prevented.	• Included 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.	• Included 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.	• Included 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 prescribed 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.	• Included in principle

Table K-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 included
Removal of Radioactive Substance Adequate facilities shall be provided for the removal of radioactive substances from the reactor coolant, including corrosion and fission products.	• Included in principle
Waste Treatment and Control Systems Adequate systems 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.	• Included 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.	• Included 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.	• Included 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.	• Included 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.	• Included in principle

Table K-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).	• To be added later
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.	• Included in principle

Table K-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.	• Included 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.	• Included 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 included

K.4 Comparison Against IAEA NS-R-2

Table K-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 K-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.

Table K-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

Table K-3 NS-R-2 comparison

IAEA Safety Standards	Technology-Neutral Framework
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
Plant Modifications	
- regulatory approval	• included
- work control	• included
- update documentation	• included
Radiation Protection and Waste Management	
- radiation protection program	• included
- waste management program	• included
- ALARA	• included
- effluent monitoring	• included
Records and Reports	
- document control	• included
Periodic Safety Review	
- update safety analysis	• included (living PRA)
- impact of operator experience	• included
- use of PSA	• included

Table K-3 NS-R-2 comparison

IAEA Safety Standards	Technology-Neutral Framework
Decommissioning	
- funding arrangements	• included
- preparation for decommissioning	• not included

K.5 Comparison Against NEI 02-02

In 2002, the Nuclear Energy Institute (NEI) prepared an submitted to the NRC for information a document (NEI 02-02) describing a way to risk-inform the NRC licensing process. NEI 02-02 was written to suggest a risk-informed, performance-based alternative to 10 CFR 50, which NEI called Part 53.

The NEI document is a high-level document describing a concept, structure, approach and content for the proposed Part 53, including examples of how to develop risk-informed alternatives to 10 CFR 50. The examples provided focused on LWR technology but acknowledged that other technologies could also be addressed if a technology-neutral approach were taken. Very little technical basis was provided for the examples and there were many technical areas that were incomplete. Nevertheless, it is useful to compare the framework topics identified in Chapter 8 against the content of NEI 02-02. This comparison is shown in Table K-4 below.

As can be seen from Table K-4, many technical items are not included in NEI 02-02. NEI 02-02 does, however, include a thorough listing of the administrative items which should be included in the proposed Part 53. It does list one item which is not included in the framework and that is in the area of selective implementation.

Table K-4 NEI 02-02 Comparison.

Framework Topic	NEI 02-02
(A) Topics Common to Design, Construction and Operation	
1) QA/QC	Included
2) PRA scope and technical acceptability	Minimally included
(B) Physical Protection	
1) General (10 CFR 73)	Included
2) Perform security assessment integral with design	Not included
3) Security performance standards	Not included
(C) Good Design Practices	
1) Plant Risk: - Frequency-Consequence curve - QHOs (including integrated risk)	Not included

Table K-4 NEI 02-02 Comparison.

Framework Topic	NEI 02-02
2) Criteria for selection of LBEs	Included
3) LBE deterministic acceptance criteria: • frequent events (dose, plant damage) • infrequent events (dose, plant damage) • rare events (dose) • link to siting	Partially included
4) Keep initiating events with potential to defeat two or more protective strategies <10^{-7}/plant year	Not included
5) Criteria for safety classification and special treatment	Partially included
6) Equipment Qualification	Included
7) Analysis guidelines • realistic analysis, including failure assumptions • source term	Partially included
8) Siting and site-specific considerations	Partially included
9) Use consensus design codes and standards	Not included
10) Materials qualification	Not included
11) Provide 2 redundant, diverse, independent means for reactor shutdown and decay heat removal	Partially included
12) Minimum - 2 barriers to FP release	Partially included
13) Containment functional capability	Partially included
14) No key safety function dependent upon a single human action	Not included
15) Need to consider degradation and aging mechanisms in design	Not included
16) Reactor inherent protection (i.e., no positive power coefficient, limit control rod worth, stability, etc.)	Partially included
17) Human factors considerations	Not included
18) Fire protection	Included
19) Control room design	Partially included
20) Alternate shutdown location	Not included
21) Flow blockage prevention	Not included
22) Specify reliability and availability goals consistent with PRA: - establish Reliability Assurance Program - specify goals on initiating even frequency	Not included

Table K-4 NEI 02-02 Comparison.

Framework Topic	NEI 02-02
23) Use of prototype testing	Not included
24) Research and Development	Not included
25) Combustible gas control	Not included
26) Coolant/water/fuel reaction control	Not included
27) Prevention of brittle fracture	Not included
28) Leak before break	Not included
29) I and C System • analog • digital • HMI	Not included
30) Criticality prevention	Not included
31) Protection of operating staff during accidents	Not included
32) Qualified analysis tools	Partially included
(D) Good Construction Practices	
1) Use accepted codes, standards, practices	Not included
2) Security	Included
3) NDE	Not included
4) Inspection	Not included
5) Testing	Not included
(E) Good Operating Practices	
1) Radiation protection during routine operation	Included
2) Maintenance program	Not included
3) Personnel qualification	Not included
4) Training	Included
5) Use of procedures	Not included
6) Use of simulators	Not included
7) Staffing	Included
8) Aging management program	Included
9) Surveillance (including materials surveillance program)	Included
10) ISI	Not included

Table K-4 NEI 02-02 Comparison.

Framework Topic	NEI 02-02
11) Testing	Included
12) Technical specifications, including environmental	Included
13) Develop EOP and AM procedures integral with design	Not included
14) Develop EP integral with design	EP included
15) Monitoring and feedback	Included
16) Work and configuration control	Included
17) Living PRA	Not included
18) Maintain fuel and replacement part quality	Not included
19) Security	Included
(F) Administrative	
1) Standard format and content of applications	Included
2) Change control process	Included
3) Record keeping	Included
4) Documentation control	Included
5) Reporting	Included
6) Monitoring and Feedback: - plant performance - environmental releases - testing results	Included
7) Corrective action program	Not included
8) Backfitting	Included
9) License amendments	Included
10) Exemptions	Included
11) Other legal, financial and process items	Included

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.]

[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.

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

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