April 15, 2020

Note To: Amir Afzali
Southern Company Services
Licensing and Policy Director- Next Generation Reactors
Energy Center
Birmingham, AL 35242

From: Joe Sebrosky
Senior Project Manager
Advanced Reactor Policy Branch
Office of Nuclear Reactor Regulation

Subject: Updated Draft Outline for Licensing Modernization Project Advanced Reactor License Applications

Dear Mr. Afzali,

During its interactions with stakeholders, the Nuclear Regulatory Commission (NRC) staff identified the need for additional guidance on the appropriate content of applications in terms of scope and level of detail. To this end the NRC proposed an outline for a content of application that was discussed during a December 12, 2019, public meeting. This draft outline is available in the Agencywide Documents Access and Management System (ADAMS) under Accession Number ML19325C089. The draft outline was released to support ongoing public discussions on this topic, including periodic stakeholder meetings, planned meetings on developing guidance on the content of applications, and planned meetings on a rulemaking to establish a technology-inclusive regulatory framework.

As you know there is a public meeting scheduled for April 22, 2020 on this topic. To support discussions during the meeting the draft outline discussed during the December 12, 2019, public meeting has been updated to include additional detail within the outline so that stakeholders have a better idea on what information is being sought in the various sections. The additional detail provided in the attached document results in the previous version of the draft outline going from 6 pages to 43 pages. In addition to providing detail for the various sections of the outline, minor modifications were made to reorganize the outline itself. However, the general outline provided in the attached document remains very similar to the previous outline discussed during the December 12, 2019, public meeting.

The attached document does not include revisions to address comments that you provided on April 10, 2020 (ADAMS Accession No. ML20107H849). The staff did not have enough time to review your comments prior to issuance of the attached document. The staff looks forward to the discussions on April 22, 2020, on this topic and intends to issue future revisions of the outline based on stakeholder interactions.

The proposed outline has not been subject to NRC management and legal reviews and approvals and should not be interpreted as an official agency position.

Introduction

The application should contain sufficient technical information, both in scope and depth, for the NRC staff to complete the detailed technical review and render an independent assessment with regard to applicable regulatory requirements and the protection of public health, safety, and security. The level of detail provided in each section of the FSAR is expected to be commensurate with the safety significance of the topic. The applicant may reference NRC approved topical or technical reports for technical information about included topics. Additionally, detailed supporting information, such as calculations or piping and instrumentation drawings and system description documents that provide content beyond the level of detail of the text, need not be provided with the application but should be available for audit by the NRC staff.

1. General Information

1.1. General plant description (later)

1.1.1. Specific technology

1.1.2. Power level

1.1.3. General arrangement

1.2. Other Important Plant Features

In this section, the applicant should provide a general description of the design that identifies the important plant design and operational features in sufficient detail to allow the reviewer to understand how the plant operates under normal and off-normal conditions. The description should include the important plant structures, systems, components and operating conditions that are modelled in the PRA and relied upon to maintain a low risk to public health and safety. Fundamental aspects of the design and their bases such as the choice of coolant, moderator, materials, fuel design, neutron spectra and passive/inherent design features should be described.

Also, a summary should be provided of how the design accomplishes the fundamental safety functions of:

- Controlling reactivity
- Removing heat from the reactor and spent fuel
- Limiting the release of radioactive material

In addition, a comparison of the design against the attributes for enhanced safety listed in the Commission's 2008 "Policy Statement on the Regulation of Advanced Reactors" should be provided.

1.3. General site description (later)

1.4. Overview of process used to develop safety analysis

Attachment

Applicants proposing to utilize an alternate approach to that described in NEI 18-04[1] and addressed in NRC Reg. Guide XXXXX for developing the safety analysis should provide a description of their alternate approach in this section. This description should justify why the alternative process adequately describes the safety case. Additionally, if the applicant used the LMP process but deviated from this guidance in selective areas then this section should describe and justify these deviations.

1.4.1. Selection and evaluation of licensing basis events
1.4.2. PRA development and technical adequacy
1.4.3. SSC safety classification and performance requirements
1.4.4. Evaluation of defense-in-depth adequacy
1.4.5. Role of the [Integrated Decision Panel (IDP)] or Expert Review Panel

1.5. Identification and bases for the principal design criteria (PDC) of the facility (later)

1.6. Overview of analytical codes and methods validation/verification (later)

1.7. Referenced materials (later)

1.8. Drawings and other detailed information (later)

1.9. Combined licensee action items (later)

1.10. Conformance with Regulatory Guides, TMI and USI Issues

Describe conformance with the applicable application content guidance document. As applicable to the design technology, describe how the design is in conformance with NRC Regulatory Guides, Three Mile Island requirements and USIs and GSIs. The following provides guidance regarding those evaluations:

1.10.1. Regulatory Guides (RGs)

The applicant should review and identify those NRC RGs applicable to the specific technology described in the application and those which are not applicable. The applicant should include a description of all deviations from the applicable RGs and the justification for the deviations. The applicant should also identify the FSAR sections where the applicable RGs are discussed.

1.10.2. Three Mile Island Requirements

The applicant should review and identify the Three Mile Island requirements (set forth in 10 CFR 50.34(f)) that are applicable to the specific technology described in the application. The applicant should provide a description of the approach

[1] Nuclear Energy Institute (NEI) "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Report Revision 1, August 2019.

proposed to address the applicable requirements. The applicant should also identify the FSAR section where these items are discussed.

1.10.3. Generic Safety Issues (GSIs) and Unresolved Safety Issues (USIs)
The applicant should review the medium and high priority GSIs and USIs described in NUREG-0933 (the version applicable 6 months before the docket date of the application) and identify those applicable to the specific technology described in the application and those which are not applicable. For those GSIs and USIs which are applicable, their proposed resolution should be described along with identification of the FSAR sections in which the resolutions are discussed.

1.11. Considerations for multi-unit sites (later)

2. **Site Information**

Chapter 2 of the FSAR should provide information on geological, seismological, hydrological, and meteorological characteristics of the site and the surrounding vicinity. Present and projected population distribution and land use and site activities and controls be discussed. The purpose of this information is to demonstrate that the applicant has accurately described site characteristics and appropriately used them in support of plant design and operating criteria. Applicants may provide a process for review of external hazards commensurate with the risk. Similarly, extensive data collection (e.g., meteorological, regional seismic, hydrologic) need only be provided to the extent that it is needed to support the safety case.

For a design certification (DC) application, the Chapter 2 application content should focus on site-related plant design characteristics and site parameters postulated for the design. This section should summarily address the complete set of site parameters and the subset of site parameters that are included within the certified design, i.e., top-level bounding site parameters that are used to define a site as suitable for the facility referencing the certified design. Because site parameters are used in bounding evaluations of a certified design, these parameters define requirements for the design and must be met by the described site.

For a COL application that reference a DC, this section of the application should demonstrate that the characteristics of the site address site parameters specified in the DC.

2.1. Site characteristics
This chapter should provide an overview of local and regional geological, seismological, hydrological, and meteorological characteristic information. Current and projected area population distributions relative to the plant risk envelope should be identified as well as land use and access control to surrounding areas. Applicant information should emphasize characteristic descriptions that confirm the site is appropriate for the design and can robustly satisfy operating criteria.

Regulatory requirements applicable to the site should be identified and discussed with respect to how requirements are met. Regulatory guidance used by the applicant should be identified, explained and justified as appropriate for use. Deviations from employed regulatory guidance

employed should be explained. Justification should be provided concerning methods used as an alternative to established guidance. The applicant should clearly describe data collected, analyses performed, investigative results, and conclusions of site studies. Previous studies used to explain conclusions about the site should be presented and justified in the FSAR.

2.2. Geography and Demography

2.2.1. Site Location and Description

Applicants should specify the location of each reactor at the site by latitude and longitude to the nearest second and by Universal Transverse Mercator Coordinates (zone number, northing, and easting, as found on topographical maps prepared by the U.S. Geological Survey (USGS)) to the nearest 100 meters (328 feet). They should consult the USGS map index for specific names of the 7½-minute quadrangles that bracket the site area. This section should also identify the State and county (or other political subdivision) in which the site is located, as well as the location of the site with respect to prominent natural features (such as rivers and lakes) and manmade features (such as industrial, military, and transportation facilities).

This section should include a suitably scaled map depicting the site area with explanatory text as necessary. Attributes of this map should follow guidance described in Section 2.1.1 of NUREG-0800.

2.2.2. Exclusion Area Authority and Control

This section should include descriptions of the applicant's legal rights with respect to all areas that lie within the designated exclusion area. As specified by 10 CFR 100.21(a), this description should establish that the applicant has the authority to determine all activities and include exclusion and removal of personnel and property from the area. The discussion should also address the status of mineral rights and easements within this area.

If the applicant has not obtained ownership of all land within the exclusion area, it should use a scaled map of the exclusion area to clearly describe those parcels of land not owned within the area. The applicant should also clearly describe the status of proceedings and the schedule to obtain ownership or the required authority over the land for the life of the plant. This section should give the minimum distance to and direction of the exclusion area boundary (EAB) for both present and proposed ownership. If the exclusion area extends into a body of water, the application should specifically address the bases upon which it has been determined that the applicant holds (or will hold) the authority required by 10 CFR 100.21(a).

Activities that will be permitted within the exclusion zone are that are unrelated to plant operation (aside from transit through the area) should be described. Limitations and conditions imposed to control activities unrelated to plant operations, arrangements for traffic control, and abandonment or relocation of roads, should be discussed.

2.2.3. Population Distribution

This section should present population data based on the latest census data and projected population at the year of plant approval and each decade thereafter out to 40 years, using a geographical distribution format as given in Section 2.1.3 of RG 1.70. Specific location(s) of potentially affected populations surrounding the site should be identified and described relevant to site and plant attributes and features. Discussion should include proposed exclusion area boundaries, local and surrounding area access control, and activities, traffic, and transient and permanent population densities that may be influenced by the plant. Discussions should be at a level and extent commensurate with the potential vulnerabilities and risks associated with all normal and off-normal plant operations. This section should describe information regarding:

- Population within the outer edge of the plume exposure pathway emergency planning zone
- Population information necessary to provide ingestion response planning
- Transient population
- Description of the low population zone – refer to Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations."
- The nearest boundary of the closest population center containing 25,000 or more residents
- Population density out to 20 miles

2.3. Nearby Industrial, Transportation, and Military Facilities

According to 10 CFR 100.21(e), "Potential hazards associated with nearby transportation routes, industrial and military facilities must be evaluated and site parameters established such that potential hazards from such routes and facilities will pose no undue risk to the type of facility proposed to be located at the site." 10 CFR 50.34(a)(1)(i) and 10 CFR 52.79(a)(1)(iv) require the applicant to evaluate the proposed reactor site with respect to the location and description of nearby industrial, military and transportation facilities and routes. The evaluation should cover the nature and extent of the nearby activities, including distance from the site, frequency of the activities and the potential hazard they represent. The evaluation needs to show that there is no undue risk to the applicant's facility from these activities.

Additional guidance regarding the scope of the evaluation and the content of the application can be found in the Standard Review Plan (NUREG-0800), Section 2, and in USNRC Regulatory Guide 4.7.

Applicants should discuss activities at and near the site to the extent necessary to establish:

- Whether effects of potential accidents in the vicinity of the site from present and projected industrial, transportation, and military activities and operations should be used as design-basis events (DBE), and
- The design parameters related to events selected.

Applicable regulatory requirements and guidance that is followed should be identified. Discussions should focus on how applicable requirements are being met at the site and include, as applicable, the following topics:

- Locations and distances from the plant to significant industrial, transportation, and military facilities that may adversely affect the plant
- Descriptions of the industrial, transportation, and military facilities
- Evaluation of potential accidents and using methods and approaches described in NEI 18-04

2.4. Meteorology

This section should describe meteorological characteristics at the site and the surrounding area. Sufficient information ~~and~~ should be included to permit independent evaluations of site meteorology by the staff. An onsite meteorological measurement program as discussed in Section 2.3.3 of NUREG-0800 may be required to support the analysis. If it is determined an on-site meteorology program is necessary, RG 1.23 contains guidance for acceptable onsite meteorological programs; deviations from this guidance should be discussed and justified.

Sufficient information should be provided to enable estimation of: (1) Short-term atmospheric dispersion during accident releases, and (2) Long-term atmospheric dispersion of routine releases, during both normal and off-normal plant operating conditions. Guidance on calculating these estimates are contained in Sections C.I.2.3.4 and C.I.2.3.5 of RG 1.206 (Revision 0) as well as Sections 2.3.4 and 2.3.5 of NUREG-0800.

The applicant should also identify local meteorological and air quality conditions that are used for design-and operating-basis considerations with reference to the FSAR sections in which these conditions are used. The following topics would typically be addressed by the applicant. Areas to be addressed should include:

- Regional Climatology
- Local Meteorology

2.5. Hydrological Engineering

Applicants should provide sufficient information to permit an independent hydrologic engineering reviews of all hydrologically related site characteristics, performance requirements, and the bases for operation of structures, systems, and components (SSCs) important to safety, to the extent those phenomena or conditions may exist at site.

For sites located in river valleys, on flood plains, or along coastlines where there is a potential for flooding exists, this section should describe the potential for floods (refer to Regulatory Guide 4.7, “General Site Suitability Criteria for Nuclear Power Stations,” Regulatory Guide 1.59, “Design Basis Floods for Nuclear Power Plants” and Regulatory Guide RG 1.206, (Rev 0) C.I.2.4 “Floods”).

The following phenomena or conditions may be in the scope of such consideration:

(1) probable maximum precipitation, on site and on the contributing drainage area
(2) runoff floods for streams, reservoirs, adjacent drainage areas, and site drainage, and flood waves resulting from dam failures induced by runoff floods
(3) surges, seiches, and wave action
(4) tsunami
(5) nonrunoff-induced flood waves attributable to dam failures or landslides, and floods attributable to failure of onsite or near-site water control structures
(6) blockage of cooling water sources by natural events
(7) ice jam flooding
(8) combinations of flood types
(9) low water and/or drought effects (including setdown resulting from surges, seiches, frazil and anchor ice, or tsunami) on safety-related cooling water supplies and their dependability
(10) channel diversions of safety-related cooling water sources
(11) capacity requirements for safety-related cooling water sources
(12) dilution and dispersion of severe accidental releases to the hydrosphere relating to existing and potential future users of surface water and ground water resources

The level of analysis presented in this section may range from very conservative, based on simplifying assumptions, to detailed analytical estimates of each facet of the bases under study. The staff suggests conservative approaches with simplifying assumptions for evaluating phenomena that do not influence the selection of site characteristics, or where the adoption of very conservative site characteristics does not adversely affect plant design.

The site and all safety-related elevations, structures, exterior accesses, equipment, and systems should be described from the standpoint of hydrologic considerations (both surface and subsurface). A topographic map of the site should be provided showings any proposed changes to natural drainage features.

The location, size, shape, and other hydrologic characteristics of streams, lakes, shore regions, and ground water environments influencing plant siting should be described. Descriptions of existing and proposed water control structures, both upstream and downstream, that may influence conditions at the site should be discussed. For these structures, the applicant should:

(1) tabulate contributing drainage areas
(2) describe types of structures, all appurtenances, ownership, seismic design criteria, and spillway design criteria
(3) provide elevation-area-storage relationships and short-term and long-term storage allocations for pertinent reservoirs

A regional map showing major hydrologic features should be provided. The applicant should list the owner, location, and rate of use of surface water users whose intakes could be adversely affected by accidental release of contaminants.

2.6. Geology, Seismology, and Geotechnical Engineering

The applicant should provide sufficient information regarding the seismic and geologic characteristics of the site and surrounding region to permit an adequate seismic evaluation of the proposed site, support evaluations performed to estimate the site-specific ground motion response spectrum (GMRS), and to permit adequate engineering solutions to actual or potential geologic and seismic effects at the proposed site. A summary of studies that include a brief description of the site, investigations performed, results of investigations, conclusions, and identification of who did the work, should be provided.

2.6.1. Basic Geologic and Seismic Information

Basic geologic and seismic information should be provided that adequately provides a basis for evaluation. The applicant can reference information obtained from published reports, maps, private communications, or other sources. Information from surveys, geophysical investigations, borings, trenches, or other investigations should be documented and include descriptions of techniques, graphic logs, photographs, laboratory results, identification of principal investigators, and other data necessary to assess the adequacy of the information.

The review should consider:

(1) Regional Geology - Discuss all geologic, seismic, tectonic, nontectonic, and manmade hazards within the site region. A review of the regional tectonics, with emphasis on the quaternary period, structural geology, seismology, paleoseismology, physiography, geomorphology, stratigraphy, and geologic history within a distance of 200 miles (320 km) from the site (site region) should be provided.

(2) Site Geology - A description of the site-related geologic features, seismic conditions, and conditions caused by human activities, at appropriate levels of detail within areas approximately defined by radii of 25 miles (40 km), 5 miles (8 km), and 0.6 miles (1 km) around the site.

2.6.2. Vibratory Ground Motion

The applicant should present the criteria and describe the methodology used to establish the GMRS. Guidance regarding the characterization of seismicity, geologic and tectonic characteristics, correlation of earthquake activity, probabilistic seismic hazard analysis, seismic wave transmission characteristics, and ground motion response spectrum development can be found in Section C.I.2.5.2 of RG 1.206.

2.6.3. Surface Faulting

The applicant should provide information describing whether a potential exists for surface deformation that could affect the site. The detailed surface and subsurface

geological, seismological, and geophysical investigations performed around the site to compile this information should be described.

2.6.4. Stability of Subsurface Materials and Foundations

The applicant should present information concerning properties and stability of all soils and rock that may affect nuclear power plant facilities, under both static and dynamic conditions, including the vibratory ground motions associated with the GMRS. The applicant should demonstrate the stability of these materials as they influence the safety of seismic Category I facilities and present an evaluation of the site conditions and geologic features that may affect nuclear power plant structures or their foundations.

2.6.5. Stability of Slopes

The applicant should present information concerning the static and dynamic stability of all natural and manmade earth or rock slopes (such as cuts, fills, embankments, and dams) for which failure, under any of the conditions to which they could be exposed during the life of the plant, could adversely affect the safety of the nuclear power plant facilities. A thorough evaluation of site conditions, geologic features, and the engineering properties of the materials comprising the slope and its foundation should be included. The results of slope stability evaluations using classic and contemporary methods of analyses: should be presented. Whenever possible, comparative field performance of similar slopes should be included. For the stability evaluation of manmade slopes, summary data and a discussion of construction procedures, record testing, and instrumentation monitoring to ensure high-quality earthwork should be included

2.6.6. Design Basis External Hazard Level (DBEHL)

A set of DBEHLs will be selected to form an important part of the design and licensing basis. This will determine the design basis seismic events and other external events that the SR SSCs will be required to withstand. When supported by available methods, data, design, site information, and supporting guides and standards, these DBEHLs will be informed by a probabilistic external hazards analysis and will be included in the PRA after the design features that are incorporated to withstand these hazards are defined. Other external hazards not supported by a probabilistic hazard analysis will be covered by DBEHLs that are determined using traditional deterministic methods.

In many cases, it is expected that the initial selection of SR SSCs and selection of the DBAs will be based on a PRA that includes internal events but has not yet been expanded to address external hazards. With the understanding that SR SSCs are required to be capable of performing their RSFs in response to external events within the DBEHL, there will be no new DBAs introduced by external hazards.

3. **Licensing Basis Event Analysis**

3.1. LBE analysis process description

The purpose of this section is to describe how the guidance provided in Draft Regulatory Guide (DG)-1353, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors" was utilized to complete the analysis and evaluation of the design and performance of structures, systems, and components (SSCs) of the facility. It is recognized that this is an iterative process and that the results are highly integrated within various section of the application (SSC Classification, etc.) The results of the LBE analysis must be shown to address the license application content requirements reflected in associated regulatory requirements, including 10 CFR 50.34 and other applicable regulatory requirements as they relate to the assessment of the risk to public health and safety resulting from operation of the facility. In particular, the identification of a complete set of LBEs is a key for assuring the comprehensive analysis and evaluation of the major structures, systems and components of the facility which bear significantly on the acceptability of the site (excerpt from 10 CFR 50.34). The event sequences considered in this analysis include internal events, internal plant hazards, and external events. The modes of operation to be considered include full power, low power & shutdown and refueling. The LBE analysis must include a determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

It is noted that the remaining subsections in this area reflect the outcomes from the application of the LBE analysis process as they relate to the primary topic of licensing basis event identification, assessment, and determination of margins of safety.

3.2. Mechanistic source term

Mechanistic source terms reflect the results of the quantitative determination of the types, quantities, and timing of the potential release of radioactive material from facility sources for the various LBE categories. These quantitative determinations play a critical role in the facility's design and evaluation of its capabilities in meeting the associated regulatory requirements for protecting the public against radiation hazards. The purpose of this section is to provide a description of the methods, analytical tools, and referenced inputs from research or test results that were utilized in performing the quantitative analyses of the event sequence families described in the following sections. The description should include a discussion of uncertainties and how they were addressed.

Additional background regarding Mechanistic Source Terms is provided in SECY-16-0012, "Accident Source Terms and Siting For Small Modular Reactors And Non-Light Water Reactors."

3.3. Frequency – Consequence Criteria

The purpose of this section is for the applicant to summarize the overall design goals for the facility as they relate to establishing margin to the Frequency-Consequence Target reflected in NEI 18-04 and discussed in DG-1353. The applicant should also include a frequency-

consequence plot that depicts the analyzed risk for each of the LBE sequences identified and described in Sections 3.4, 3.5, and 3.6 below.

3.4. AOOs

3.4.1. Event Sequence Families
Provide a listing and summary description of the identified AOO event sequence families. Individual event sequences within these AOO families are generally referred to as transients in associated regulatory requirements. They are anticipated to occur one or more times during the life of a nuclear power plant, which may include one or more reactor modules. The description of each family should summarize the plant response and end state, the mechanistic source term for sequence families with a release, and the offsite radiological consequences, if applicable.

3.4.2. Provide a summary description of SSC response and end state for each AOO and the associated mechanistic source term for event sequences resulting in a radiological release. The description should include a comparison assessment of the results of the response against the design objective and associated margin to the F-C target that was established for this event sequence type as reflected in Section 3.3.

3.5. DBEs

3.5.1. Event Sequence Families
Provide a listing and summary description of the identified DBE event sequence families. These families are those that include infrequent event sequences that are not expected to occur in the life of a nuclear power plant, which may include one or more reactor modules, but are less likely than AOOs. The description of each family should summarize the plant response and end state, the mechanistic source term for sequence families with a release, and the offsite radiological consequences, if applicable.

3.5.2. Required Safety Functions
Provide a description of the Required Safety Functions that are necessary and sufficient to meet the F-C Target for all DBEs, and to conservatively ensure that 10 CFR 50.34 dose requirements can be met using realistic assumptions.

3.5.3. Safety-Significant Functions
Provide a description of the Safety-Significant Functions that meet the associated risk significance and defense in depth criteria for the identified DBE event sequence families within the proposed design. These functions are in addition to the Required Safety Functions identified and described in Section 3.5.3 above.

3.5.4. Required Functional Design Criteria (RFDCs) for Safety-related SSCs

The purpose of this section is for the applicant to provide a description of the top-down logical development that was used to define the functional requirements that must be fulfilled for the reactor design to meet each RSF identified in Section 3.5.3. (The RFDCs can be viewed as functional criteria that are defined in the context of the specific reactor design features that are necessary and sufficient to provide the RSF.) In addition to the results from the top-down approach, applicants should consider the guidance provided in RG 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors" for additional insights in establishing the RFDCs.

The corresponding SRDCs are then developed from the RFDCs.

3.5.5. Adequacy of plant response to DBEs
Provide a summary description of SSC response and end state for each DBE, and the associated mechanistic source term for event sequences resulting in a radiological release. The description should include a comparison assessment of the results of the response against the design objective and associated margin to the F-C target that was established for this event sequence type as reflected in Section 3.3.

3.6. BDBEs

3.6.1. Event Sequence Families
Provide a listing and summary description of the identified BDBE event sequence families. These families are those that include rare event sequences that are not expected to occur in the life of a nuclear power plant, which may include one or more reactor modules, and are less likely than a DBE. The description of each family should summarize the plant response and end state, the mechanistic source term for sequence families with a release, and the offsite radiological consequences, if applicable.

3.6.2. Required Safety Functions (RSFs)
Provide a description of the Required Safety Functions associated with the high consequence BDBEs (those with consequences that exceed 10 CFR 50.34 dose criteria). Include the reliability performance requirements for the identified RSFs for preventing any of the high-consequence BDBEs from increasing in frequency into the DBE region and outside the F-C Target.

3.6.3. Safety-Significant Functions
Provide a description of the Safety-Significant Functions that meet the associated risk significance and defense in depth criteria for the identified BDBE event sequence families for the proposed design. These functions are in addition to the Required Safety Functions identified and described in Section 3.6.3 above.

3.6.4. Required Functional Design Criteria for Safety Related SSCs
The purpose of this section is for the applicant to provide a description of the top-down logical development that was used to define the functional requirements that

must be fulfilled for the reactor design to meet each RSF identified in Section 3.6.3. (The RFDCs can be viewed as functional criteria that are defined in the context of the specific reactor design features that are necessary and sufficient to provide the RSF.) In addition to the results from the top-down approach, applicants should consider the guidance provided in RG 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors" for additional insights in establishing the RFDCs.

The corresponding SRDCs are then developed from the RFDCs.

3.6.5. Adequacy of plant response to BDBEs
Provide a summary description of SSC response and end state for each BDBE, and the associated mechanistic source term for event sequences resulting in a radiological release. The description should include a comparison assessment of the results of the response against the design objective and associated margin to the F-C target that was established for this event sequence type as reflected in Section 3.3.

3.7. Risk Significance Evaluations
Describe the absolute and relative risk significance of individual LBEs and SSCs, which include radionuclide barriers, including the application of the LBEs the integrated risk evaluations. These evaluations include the use of PRA. risk importance metrics, where applicable, and the examination of the effectiveness of each of the layers of defense in retaining radionuclides. LBEs should be classified as risk-significant if the LBE site boundary dose exceeds 2.5 mrem over 30 days and the frequency of the dose is within 1% of the F-C Target. SSCs should be classified as risk-significant if the SSC function is necessary to keep any LBEs inside the F-C Target, or if the total frequency of LBEs with the SSCs failed is within 1% of any of the three cumulative risk targets. Describe how this information is used to provide risk insights, to identify safety-significant SSCs, and to support the RIPB evaluation of defense-in-depth.

3.8. Aircraft Impact Analysis
The purpose of this section is to provide the results of the applicant's assessment of the effects on the facility from the impact of a large, commercial aircraft in accordance with the requirements of 10 CFR 50.150. The analysis should include consideration of the guidance provided in Reg. Guide 1.217, "GUIDANCE FOR THE ASSESSMENT OF BEYOND-DESIGN-BASIS AIRCRAFT IMPACTS", and in NEI 07-13, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs". The analysis of this deterministically defined and beyond-design-basis event should provide a realistic analysis, and be able to show that, with reduced use of operator actions:

(i) The reactor core remains cooled, or the functional containment remains intact; and
(ii) Spent fuel cooling or spent fuel pool integrity is maintained.

4. **Integrated Plant Analysis**
In this section the applicant should describe the methods, assumptions and results of integrated plant analysis that shows (1) the overall risk to public health and safety from plant operation is low; (2) the plant releases are within the NRC safety goals Quantitative Health Objectives (QHOs); and (3)

the requirements in 10 CFR 20 and 10 CFR 50 regarding limiting the dose to members of the public from normal operation will be met. The following sub-sections provide additional guidance regarding the information that should be provided.

4.1. NRC Safety Goal QHOs for Early and Latent Fatalities

The risk to members of the public should be calculated using risk metrics that assess, in an integrated fashion, the overall risk to the public from the plant design and operation. NEI 18-04 and DG-1353 utilize the QHOs as the chosen risk metrics.

As stated in the Commission's 1986 Safety Goal Policy Statement, the QHOs are defined in terms of average individual risk. Specifically, the following quantitative values for the QHOs, expressed as mean values, have been adopted for use:

- Early Fatality Individual Risk less than 5x10-7/yr for individuals located within 1 mile of the EAB
- Latent Fatality Individual Risk less than 2x10-6/yr for individuals located within 10 miles of the EAB

The applicant should describe the analysis done to determine the integrated annual Early and Latent Fatality Individual Risk from plant operation. The description should include:

- The site parameters (e.g. meteorology, off-site population distribution, EAB size) used in the analysis.
- The analysis method used.
- Key assumptions (e.g. emergency preparedness measures, source terms, timing and duration of release, credit for medical treatment, early and latent fatality risk coefficients) used in the analysis.
- Modes of operation (full power, low power & shutdown, refueling) considered in the analysis.
- How multiple units on the site were considered.
- Uncertainty/sensitivity analysis performed
- Results, including comparison to the QHOs.

4.2. Analysis to show compliance with Part 20 requirements

As stated in 10 CFR 20.1301(a) (1) and (2) the effective dose equivalent to individual members of the public should not exceed 0.1 rem (1 mSv) per year and the dose in any unrestricted area should not exceed 0.002 rem (0.02 millisievert) in any one hour, respectively. To demonstrate compliance with these requirements, the applicant should describe the analysis done, including:

- Assumptions on location of individual members of the public.
- The analysis method used.
- Doses from normal operation, including AOOs, considered in the analysis.
- Annual average meteorology used in the analysis.
- How multiple units on the site were considered.
- Uncertainty analysis.
- Results, including comparison to the requirements specified in 10 CFR 20.1301.

As stated in 10 CFR 50.34a(a) the levels of radioactive materials in gaseous and liquid effluents released to unrestricted areas should be As Low As Reasonably Achievable (ALARA). Non-LWRs should justify ALARA criteria (e.g., numerical guidelines for meeting the ALARA criterion for LWRs are provided in 10 CFR 50, Appendix I). The applicant should describe the technology specific numerical guidelines for gaseous and liquid effluents, and their bases, to be applied in the design to comply with the ALARA criterion.

5. **Description and Classification of SSCs**

The purpose of this section is to describe SSC safety classification and to identify the derivation of requirements necessary to support SSC performance of safety functions in the prevention and mitigation of LBEs that are modeled in the PRA. Such requirements include those to provide the necessary capabilities to perform their mitigation functions and those to meet their reliability targets to prevent LBEs with more severe consequences.

5.1. SSC classification process description

The applicant should describe in this section the process used to classify SSCs and describe the approach used to identify performance and special treatment requirements. NEI 18-04 describes an acceptable approach to classifying SSCs. The SSC performance and special treatment requirements identified by this process should provide reasonable confidence in the SSC capabilities and reliabilities in performing the necessary safety functions

5.2. Overview of Primary Safety Functions

As described in NEI-18-04, Fundamental Safety Functions (FSFs) are a set of high-level functions that, when demonstrated, satisfy the public safety objective of the Atomic Energy Act. The application of the reactor-specific safety design approach leads to a set of reactor-specific safety functions, i.e., required safety functions, that achieve the FSFs. Required safety functions are reactor design specific SSC functions modeled in a PRA that serve to prevent and/or mitigate a release of radioactive material or to protect one or more barriers to release such that the consequence of one or more design basis events (DBEs) or the frequency of one or more high-consequence beyond DBEs (BDBEs) is maintained inside the F-C Target. The applicant should summarize in this section how the specific design achieves each of the following fundamental safety functions.

5.2.1. Reactivity control

The function of controlling reactivity is defined as the active, passive, or inherent means provided (1) to control the nuclear chain reaction consistent with the intended plant operating conditions, (2) to terminate the nuclear chain reaction when transient or accident conditions dictate that the facility must be shut down, and (3) to prevent inadvertent criticality in the reactor core, primary system, or other areas of the plant where inadvertent criticality is an adverse condition that could result in unacceptable radiological consequences.

5.2.2. Remove core heat

The function of removing heat from the reactor and waste stores is defined as the active, passive, or inherent means provided (1) to remove the heat generated from

the nuclear chain reaction during normal plant operating modes so that the nuclear fuel and primary system retain their integrity, (2) to remove the decay or residual heat from the reactor and primary system when the nuclear chain reaction is terminated and when the facility is shut down, and (3) to remove the residual heat from material that is being stored in waste handling and fuel handling areas so that unplanned releases of radioactive materials from the plant do not occur.

5.2.3. Maintain control of radionuclide release

The function of limiting the release of radioactive materials is defined as the active, passive, or inherent means provided to prevent or mitigate the release of radioactive materials from the plant to the public and the environment.

5.3. Safety-Related SSCs

The purpose of this section is to describe safety-related (SR) SSCs and related requirements necessary to support SSC performance of required safety functions in the prevention and mitigation of LBEs that are modeled in the PRA. This section should address the application content requirements specified in §52.47(a)(2) or §50.34(a)(4) that require a description and analysis of the SSCs of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. For SSCs classified as SR, required functional design criteria (RFDC) and lower-level design criteria should be defined to capture design-specific criteria that may supplement or may not be captured by the principal design criteria (PDC) for a reactor design developed using the guidance in RG 1.232. These criteria are used within the methodology to frame specific design requirements as well as special treatment requirements for SR SSCs. Requirements to be described include those to provide the necessary capabilities to perform SSC mitigation functions and those to meet reliability targets to prevent LBEs with more severe consequences. For each SR SSC, information regarding the following topics should be described:

- Design requirements, applicable codes, and relationship to PDCs
- External hazard levels
- Reliability and capability performance requirements
- Design features
- Required supporting functions
- Evaluation of adequacy of special treatment
- Associated testing/validation

SR SSCs to be described in this section include:

- SSCs selected by the designer from the SSCs that are available to perform the required safety functions (RSFs) to mitigate the consequences of DBEs to within the LBE F-C Target, and to mitigate DBAs that only rely on the SR SSCs to meet the dose limits of 10 CFR 50.34 using conservative assumptions
- SSCs selected by the designer and relied on to perform RSFs to prevent the frequency of BDBE with consequences greater than the 10 CFR 50.34 dose limits from increasing into the DBE region and beyond the F-C Target

5.3.1. Design requirements, applicable codes, and relationship to PDCs

Design Requirements: The required safety functions (RSFs) should be used to define a set of reactor-specific required functional design criteria (RFDC) from which safety-related design criteria (SRDC) may be derived. Because the RFDCs are derived from a specific reactor technology and design, supported by a design-specific PRA, and related to a set of design specific RSFs, each non-LWR design would require the development of a unique set of RFDCs. One purpose of the RFDCs is to form a bridge between the safety classification of SSCs and the derivation of SSC performance, special treatment requirements, and safety-related design criteria (SRDC). The RFDCs may be viewed as functional criteria that are defined in the context of the specific reactor design features that are necessary and sufficient to meet the RSF. The corresponding SRDCs are then developed from the RFDCs. The design requirements are performance-based and tied to RSFs, should be derived from the LBEs, and used to systematically select the SR SSCs. For each of the RFDCs, the applicant should identify a set of SRDC appropriate to the SR SSCs assigned to perform the RSFs.

SR SSCs that provide functions that support the retention of radioactive material within barriers have associated regulatory design requirements that are derived from the evaluation of the LBE against the F-C Target and the RFDCs. These functions include barrier functions in which the SSC serves as a physical or functional barrier to the transport of radionuclides and indirect functions in which performance of an SSC function serves to protect one or more other SSCs that may be classified as barriers. If a SR SSC has such a function then the barrier design requirements should be described.

Performance criteria for a barrier, or set of barriers taken together, to the transport of radionuclides are discussed in SECY-18-0096, “Functional Containment Performance Criteria for Non-Light-Water-Reactors,” dated September 28, 2018 (Ref. 30), which describes an approach to “functional containment” for non-LWRs that may not rely on traditional containment structures to limit the physical transport and release of radioactive material to the environment. The Commission’s SRM dated December 4, 2018, approved the methodology described in SECY-18-0096 for establishing functional containment performance criteria for non-LWRs.

Applicable Codes: The initial quality of the design can be support through the application of proven practices and application of available and applicable industry codes and standards. The codes applicable to each SR SSC should be described. At the individual SSC level, properly designing SSCs to available and applicable codes and standards provides an appropriate level of assurance that the SSC will perform reliably at its design conditions and normally include reserve margin for more demanding conditions.

PDCs: As required by 10 CFR 50.34(a) and 52.47, and described in Regulatory Guide 1.232, Guidance for Developing Principal Design Criteria for Non-Light-Water

Reactors, a construction permit, design certification, combined license, standard design approval, or manufacturing license must include the principal design criteria (PDC) for the facility. These PDCs should be described in Section 1.5 of the application. This section (5.3.1) should describe how of each of the SR SSCs meets the applicable PDC.

5.3.2. External hazard levels

As part of the LBE analysis the Design Basis External Hazard Levels (DBEHLs) are defined. A DBEHL is a design specification of the level of severity or intensity of an external hazard for which the Safety-Related SSCs are designed to withstand with no adverse impact on their capability to perform their RSFs. These DBEHLs should be described in Chapter 2 of the application. This section should describe how SR SSCs that are credited in the fulfillment of RSFs are capable to perform their RSFs with a high degree of confidence in response to any DBEHL.

5.3.3. Reliability and capability performance requirements

Information from the PRA is used as input to the selection of reliability targets and performance requirements for SSCs that set the stage for the selection of special treatment requirements. For SSCs classified as SR, which are considered safety-significant SSCs, these requirements are used to develop specific design and special treatment requirements.

In order to meet the risk targets (F-C Target and cumulative risk targets), SR SSCs will need to meet strict reliability performance targets and will need to demonstrate DID adequacy. Strategies to achieve design reliability targets include use of passive and inherent design features redundancy, diversity, and defenses against common-cause failures. Programmatic actions would be used to maintain performance within the design reliability targets.

5.3.4. Design features

The applicant should identify specific design features of each SR SSC that are responsible for meeting the SRDC. This description should include features that demonstrate system capability and reliability for both prevention and mitigation of LBEs, as applicable.

5.3.5. Required supporting functions

Describe important system interdependencies, including failure modes and effects of nonsafety-related SSCs (e.g., support systems) that could directly affect safety-related functions including the following as applicable:

5.3.5.1. Instrumentation for control and monitoring
5.3.5.2. Structural
5.3.5.3. Power

5.3.6. Evaluation of adequacy of special treatment

The term “special treatment” is used in a manner consistent with NRC regulations and Nuclear Energy Institute (NEI) guidelines in the implementation of 10 CFR 50.69. In Regulatory Guide 1.201, the following definition of special treatment is provided:

“…special treatment refers to those requirements that provide increased assurance beyond normal industrial practices that structures, systems, and components (SSCs) perform their design-basis functions.”

All safety-significant SSCs are subject to special treatment requirements. This section should describe special treatment requirements applicable to each SR SSC. These requirements should include specific performance requirements to provide adequate assurance that the SSCs will be capable of performing their RSFs with significant margins and with appropriate degrees of reliability. These include numerical targets for SSC reliability and availability, design margins for performance of the RSFs, and monitoring of performance against these targets with appropriate corrective actions when targets are not fully realized. Another consideration in the setting of SSC performance requirements is the need to assure that the results of the plant capability DID evaluation described in Chapter 6 of the application are achieved not just in the design, but in the as-built and as-operated and maintained plant throughout the life of the plant.

5.3.7. Associated testing/validation

Special treatment requirements for SR SSCs may include the performance of routine testing and validation of SSC performance capability. Describe, as applicable, the special treatment requirements from NEI 18-04, Table 4-1, on a case-by-case basis and in the context of the SSC functions in the prevention and mitigation of applicable LBEs. These special treatment items for SR SSC may include the following:

- Equipment qualification - Essentially the same as for existing reactors for SR SSCs, 10 CFR 50.49
- Seismic qualification - Essentially the same as for existing reactors for SR SSCs, 10 CFR 100 Appendix A, Regulatory Guide 1.100
- Materials qualification
- Pre-service and risk-informed in-service inspections - See Regulatory Guide 1.178
- Pre-service and in-service testing - In–service testing needs to be integrated with Reliability Assurance Program
- Surveillance testing - Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met (i.e., demonstrate the ability to perform the safety function).

5.4. Non-Safety-Related SSCs with Special Treatment

5.4.1. Design requirements, applicable codes, and relationship to PDCs

Design Requirements: The required safety functions (RSFs) should be used to define a set of reactor-specific required functional design criteria (RFDC) from which safety-related design criteria (SRDC) may be derived. Because the RFDCs are derived from a specific reactor technology and design, supported by a design-specific PRA, and related to a set of design specific RSFs, each non-LWR design would require the development of a unique set of RFDCs. One purpose of the RFDCs is to form a bridge between the safety classification of SSCs and the derivation of SSC performance, special treatment requirements, and safety-related design criteria (SRDC). The RFDCs may be viewed as functional criteria that are defined in the context of the specific reactor design features that are necessary and sufficient to meet the RSF. The corresponding SRDCs are then developed from the RFDCs. For each of the RFDCs, the applicant should identify a set of SRDC applicable to NSRST SSCs assigned to perform the RSFs.

SSCs that provide functions that support the retention of radioactive material within barriers have associated regulatory design requirements that are derived from the evaluation of the LBE against the F-C Target and the RFDCs. These functions include barrier functions in which the SSC serves as a physical or functional barrier to the transport of radionuclides and indirect functions in which performance of an SSC function serves to protect one or more other SSCs that may be classified as barriers. If a NSRST SSC has such a function then the barrier design requirements should be described.

Applicable Codes: The initial quality of the design can be support through the application of proven practices and application of available and applicable industry codes and standards. The codes applicable to each NSRST SSC should be described. At the individual SSC level, properly designing SSCs to available and applicable codes and standards provides an appropriate level of assurance that the SSC will perform reliably at its design conditions and normally include reserve margin for more demanding conditions.

PDCs: As required by 10 CFR 50.34(a) and 52.47, and described in Regulatory Guide Regulatory Guide 1.232, Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors, a construction permit, design certification, combined license, standard design approval, or manufacturing license must include the principal design criteria (PDC) for the facility. These PDCs should be described in Section 1.5 of the application. This section should describe how of each of the NSRST SSCs meets the applicable PDC.

5.4.2. External hazard levels

As part of the LBE analysis the Design Basis External Hazard Levels (DBEHLs) are defined. A DBEHL is a design specification of the level of severity or intensity of an external hazard for which the NSRST SSCs are designed to withstand with no adverse impact on their capability to perform their RSFs. These DBEHLs should be described in

Chapter 2 of the application. This section should describe how NSRST SSCs that are credited in the fulfillment of RSFs are capable to perform their RSFs with a high degree of confidence in response to any DBEHL.

5.4.3. Performance requirements

Information from the PRA is used as input to the selection of reliability targets and performance requirements for SSCs that set the stage for the selection of special treatment requirements. For SSCs classified as NSRST, which are considered safety-significant SSCs, these requirements are used to develop specific design and special treatment requirements.

In order to meet the risk targets (F-C Target and cumulative risk targets), SSCs that are relied upon will need to meet strict reliability performance targets and will need to demonstrate DID adequacy. Strategies to achieve design reliability targets include use of passive and inherent design features redundancy, diversity, and defenses against common-cause failures. Programmatic actions would be used to maintain performance within the design reliability targets.

5.4.4. Design features

The applicant should identify specific design features of each NSRST SSC that are responsible for meeting the SRDC. This description should include features that demonstrate system capability and reliability for both prevention and mitigation of LBEs, as applicable.

5.4.5. Required supporting functions

Describe important system interdependencies, including failure modes and effects of nonsafety-related SSCs (e.g., support systems) that could directly affect NSRST functions including the following as applicable:

5.4.5.1. Instrumentation for control and monitoring
5.4.5.2. Structural
5.4.5.3. Power

5.4.6. Evaluation of adequacy of special treatment

All NSRST SSCs are subject to special treatment requirements. This section should describe special treatment requirements applicable to each NSRST SSC. These requirements should include specific performance requirements to provide adequate assurance that the SSCs will be capable of performing their RSFs with significant margins and with appropriate degrees of reliability. These include numerical targets for SSC reliability and availability, design margins for performance of the RSFs, and monitoring of performance against these targets with appropriate corrective actions when targets are not fully realized. Another consideration in the setting of SSC performance requirements is the need to assure that the results of the plant capability DID evaluation described in Chapter 6 of the application are achieved not

just in the design, but in the as-built and as-operated and maintained plant throughout the life of the plant.

5.4.7. Associated testing/validation

Special treatment requirements for NSRST SSCs may include the performance of routine testing and validation of SSC performance capability. Describe, as applicable, the special treatment requirements from NEI 18-04, Table 4-1, on a case-by-case basis and in the context of the SSC functions in the prevention and mitigation of applicable LBEs. These special treatment items for NSRST SSCs may include the following

- Reliability assurance targets
- Seismic qualification
- Pre-service and risk-informed in-service inspections - See Regulatory Guide 1.178
- Pre-service and in-service testing - In–service testing needs to be integrated with Reliability Assurance Program
- Surveillance testing - Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met (i.e., demonstrate the ability to perform the safety function).

5.5. Non-Safety-Related SSCs with No Special Treatment (General Info only)

The level of detail for ancillary plant systems in non-LWR designs could be significantly less than that provided for LWRs because of the expected use of passive safety systems and increased thermal capacities of reactor systems, which reduce sensitivities to plant upsets. A description of ancillary plant systems or the interface between the ancillary and primary plant systems should focus on any safety functions being supported and possible contributions to initiating events. Where SSCs do not play a meaningful role in preventing or mitigating LBEs, minimal information on those SSCs should be provided within this application.

For those SSCs classified as no special treatment, the reliability and capability targets are part of the non-regulatory owner design requirements.

6. **Design Basis Accidents Analysis (10 CFR 50.34**)

6.1. DBA analysis process description

Discuss the applicable accident analyses and justify its conformance to applicable requirements including 10 CFR 50.34, “Contents of Applications; Technical Information,” 10 CFR 52.47(a), “Contents of applications; technical information.” Describe the code validation done to ensure that the analysis results properly reflect the plant design and operating conditions.

6.2. Deterministic accident analysis

Event sequences included should encompass all DBAs, which are derived from the design basis events (DBEs) (refer to Chapter 3) by prescriptively assuming that only Safety Related SSCs are

available to mitigate postulated event sequence consequences to within the 10 CFR 50.34 dose limits. For each DBE identified, a deterministic DBA is defined that includes the required safety function (RSF) challenges represented in the DBE but assumes that the RSFs are performed exclusively by SR SSCs, and all non-safety-related SSCs that perform these same functions are assumed to be unavailable.

NRC Regulatory Guide 1.203, "Transient and Accident Analysis Methods," provides additional discussion of developing appropriate evaluation models for analyzing DBAs. The selection of conservative assumptions to be used in the DBA analysis will be informed by the quantitative uncertainty analysis of consequences that will be performed for the corresponding DBEs. In view of the fact that advanced non-LWRs will employ a diverse combination of inherent, passive, and active design features to perform the RSFs across layers of defense, and, taking into account the fact that the reactor safety design approach will be subjected to an evaluation of DID adequacy, the application of a single failure criterion is not deemed to be necessary.

Some design basis external events such as external floods or seismic events may impact multiple reactor modules concurrently; therefore, the analyses should explain the effects of such events (i.e., the site boundary dose when plotted and evaluated on the F-C Target with the LBE frequency would not result in a risk-significant LBE).

The codes and models used in DBA analysis are expected to satisfy Regulatory Guide 1.203 requirements for evaluation models. The applicant should summarize the assumptions, parameters, and calculational methods used to determine the doses that result from accidents. Provide sufficient information to allow an independent analysis to be performed. Include all pertinent plant parameters that are required to calculate doses for the EAB and LPZ, as well as those locations within the EAB where significant site-related activities may occur (e.g., the control room).

6.2.1. Acceptance criteria for postulated accidents

Provide information describing the acceptance criteria used for the DBA analysis. Describe how the plant meets the offsite dose requirements in 10 CFR 50.34.

6.2.2. Event X evaluation

6.2.2.1. Plant characteristics considered/assumed

The applicant should summarize the plant parameters considered in the safety evaluation (e.g., core power, core inlet temperature, reactor system pressure, core flow, axial and radial power distribution, fuel and moderator temperature coefficient, void coefficient, reactor kinetics parameters, available shutdown rod worth, and control rod insertion characteristics). Specify the range of values for plant parameters that vary with fuel exposure or core reload. Ensure that the range is sufficiently broad to cover expected changes predicted for the fuel cycles to the extent practicable based on the fuel design and acceptable analytical methodology at the time of the design

certification or COL application. Specify the permitted operating band (permitted fluctuations in a given parameter and associated uncertainties) on reactor system parameters. Use the most adverse conditions within the operating band as initial conditions for transient analysis.

6.2.2.2. Assumed protection and safety system actions

The applicant should list the settings of all protection system functions that are used in the safety evaluation. Typical protection system functions may include reactor trips, isolation valve closures, and emergency cooling system initiation. List the expected limiting delay time for each protection system function, and describe the acceptable methodology for determining uncertainties (e.g., from combined effect of calibration error, drift, instrumentation error) to be included in the establishment of the trip setpoints and allowable values specified in the plant TS.

6.2.2.3. Identification of causes and frequency classification

Refer to Chapter 3 for a description of event causes and frequencies.

6.2.2.4. Sequence of events and Systems Operation

The applicant should discuss the following considerations for each DBA:

(1) step-by-step sequence of events from event initiation to the final stabilized condition; identify each significant occurrence on a time scale (e.g., for LWRS events may include flux monitor trips, insertion of control rods begins, primary coolant pressure reaches safety valve setpoint, safety valves open, safety valves close, containment isolation signal is initiated, and containment is isolated); identify all operator actions credited in the transient and accident analyses for consequence mitigation
(2) extent to which normally operating plant instrumentation and controls are assumed to function
(3) extent to which plant and reactor protection systems are required to function
(4) credit taken for the functioning of normally operating plant systems
(5) operation of safety systems that are required

6.2.2.5. Core, system, and barrier performance

<u>Evaluation Model</u>

The applicant should discuss the evaluation model used and any simplifications or approximations introduced to perform the analyses. Identify digital computer codes used in the analysis. If a set of codes is used, describe the method used to combine these codes. Present and discuss the important output of the codes under “Results.” Emphasize the input data and the extent or range of variables investigated. The detailed descriptions of evaluation models and digital computer codes or listings should be included by

referencing documents that are available to the NRC, if possible, and providing only summaries in the text of the application itself.

The applicant should provide a table listing the titles of topical reports that describe models or computer codes used in transient and accident analyses and listing the associated NRC safety evaluation reports approving those topical reports. The applicant should ensure that these referenced topical reports are also included in the table of material referenced that should be provided in FSAR Section 1.7.

Demonstrate that the use of the NRC-approved models or codes is within the applicable range and conditions of the models or codes. Provide a discussion to address compliance with each of the conditions and limitations in the NRC safety evaluation reports approving the topical reports that document the models or codes used.

Input Parameters and Initial Conditions

The applicant should identify the major input parameters and initial conditions used in the analyses. Include the initial values of other variables and parameters in the application if they are used in the analyses of the particular event under study. Ensure that the parameters and initial conditions used in the analyses are suitably conservative for the event under study. Discuss the bases (including the degree of conservatism) used to select the numerical values of the input parameters.

6.2.2.6. Results

The applicant should present the results of the analyses, including key parameters as a function of time during the course of the transient or accident. Describe how the upper bound consequences for each DBA, defined as the 95th percentile of the uncertainty distribution, meet the 10 CFR 50.34 dose limit at the EAB. Sources of uncertainty in both frequencies and consequences of DBAs should be described.

6.2.3. Event Y Evaluation [same as DBA X above]
- 6.2.3.1. Plant characteristics considered/assumed
- 6.2.3.2. Assumed protection and safety systems actions
- 6.2.3.3. Identification of causes and frequency classification
- 6.2.3.4. Sequence of events and systems operation
- 6.2.3.5. Core, system, and barrier performance
- 6.2.3.6. Results

7. **Defense in Depth**

7.1. DID process description

The philosophy of DID, multiple independent but complimentary methods for protecting the public from potential harm from nuclear reactor operation, is an important part of the design,

licensing, and operation of nuclear power plants. A key to implementing this philosophy is for the inclusion of multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon.

The purpose of this section is to provide a summary description of how the process reflected in NEI 18-04 was implemented to assess and establish DID by utilizing a combination of plant capabilities and programmatic controls. Evaluations are performed to assess the reactor design and operation and to determine if additional measures are appropriate to address an over-reliance on specific SSCs or programmatic controls and to address uncertainties.

The framework for evaluating DID described in NEI 18-04 has the following two elements: (1) programmatic DID and (2) plant capability DID. Each of these should be described in the application with respect to how they have been applied in the design process to achieve an acceptable level of DID. The evaluation of DID should be performed by a risk-informed integrated decision-making process (IDP), as described in NEI 18-04. It is recognized that the process is iterative in that if the evaluation of DID identifies a weakness, the design or programmatic measures are adjusted and DID reevaluated until the reevaluation indicates the IDP decision guidelines are met. A summary of the DID iterations should be provided. The outcome of the IDP also establishes the DID baseline for managing risk throughout the plant life.

In addition to the above, the application should describe the key elements of the programmatic and plant capability DID discussed in the following sections.

7.2. Programmatic Defense-in-Depth

The purpose of this section is to provide a description of the programs and associated administrative controls that have been established and will be implemented to align with and assure that the performance targets discussed in Chapter 5 are being met. See Tables 5-5, 5-6 and 5-7 in NEI 18-04 for additional background regarding this topic. Programmatic DID is used to address uncertainties when evaluating plant capability DID as well as uncertainties in programmatic measures. It provides the basis for defining special treatment requirements to ensure there is reasonable assurance that the predicted performance of SSCs and programmatic measures can be achieved throughout the life of the plant. The uncertainties addressed by programmatic DID should be described.

7.2.1. Performance targets for SSC reliability and capability

In developing the design and in performing the PRA, values are chosen for SSC reliability, availability and performance. These values are used to develop plant level and system level functional requirements and represent the expected performance of those SSCs considered in the DID evaluation. These performance requirements should be described. Refer to Sections 7.3.3 and 7.3.4 below.

7.2.2. Design, testing, manufacturing, construction, operations, and maintenance programs to meet performance targets

To help ensure the SSC reliability, availability and performance targets are met the engineering process employs design standards, defines testing programs, manufacturing processes, construction methods, operational limits and required maintenance to provide a product that meets the designer's expectations. In some cases additional measures are employed to help ensure the desired performance. These are the special treatment measures and should be described.

7.2.3. Tests, inspections, and monitoring of SSC performance and corrective actions
Over the life of the plant periodic testing and inspections will be performed on those SSCs important to DID. In addition, the performance of those SSCs will be monitored for consistency with values used in the PRA and safety analysis. The application should describe the test, inspection and monitoring planned for the SSCs along with a summary of how it will be determined when corrective actions are needed.

7.2.4. Operational procedures and training to compensate for human errors, equipment failures, and uncertainties
To prevent human errors, procedures and training are used along with human factors engineering in the design of control stations and man-machine interfaces. The application should describe how these have been employed to reduce human error.

7.2.5. Technical specifications to bound uncertainties
Technical specifications can be used to ensure that the plant operating envelope remains within the bounds of the PRA and safety analysis. This can include setting limiting conditions for operation and allowable outage times for SSCs consistent with the assumed reliability for risk-significant LBEs. The application should describe the technical specification measures established to maintain an acceptable level of DID.

7.2.6. Capabilities for emergency plan protective action
Emergency preparedness (EP) has traditionally been considered an element of DID. The application should describe how EP has been factored into the DID evaluation.

7.3. Plant Capability Defense-in-Depth
The purpose of this section is to provide a description of the SSCs and the layers of defense they represent in the overall achievement of an acceptable level of DID. See Table 5-3 in NEI 18-04 for additional background regarding this topic. Plant capability DID can include redundancy, diversity and independence in SSCs to ensure that plant safety is not dependent on a single feature in the plant design. As with programmatic DID, plant capability DID is implemented using an iterative process that could result in plant design changes until the IDP determines that the guidelines for an acceptable level of DID are met.

7.3.1. Reactor and site characteristics
The application should describe the reactor design and site characteristics that contribute to DID. This could include inherent or passive design features; redundant, diverse and independent design features; remote siting; siting in low seismic hazard areas or other important characteristics.

7.3.2. Radionuclide physical and functional barriers
The application should describe the physical and functional barriers to the release of radioactive material and their design capability.

7.3.3. SSC reliability
This section should describe the key SSC reliability and availability goals assumed in the PRA and how the Section 13.4 Reliability Assurance Program results are used to assure the continued adequacy of the SSC reliability and availability.

7.3.4. SSC capability
This section should describe the capability of those SSCs relied upon for DID. This should include how that capability is ensured through testing, maintenance, inspection and performance monitoring.

7.3.5. Defenses against common cause failures
An important aspect of DID is the prevention of common cause failures that can affect more than one line of defense at a time. The application should describe how the process identifies the potential for common cause failures and how those vulnerabilities were eliminated.

7.3.6. Conservative design margins in SSC performance
The application should describe how design margins are used to ensure the adequate performance of SSCs. The key design margins should be identified.

7.4. Evaluation of Overall Defense in Depth Adequacy
The evaluation of the adequacy of DID should be done by an IDP. The DID approved as part of the initial operating license will be considered the baseline DID which will be used in determining the acceptability of future plant changes. The following sections describe the information that should be included in the application on each of these topics.

7.4.1. Implementation of an Integrated Decision-Making Process
This section of the application should describe the IDP that was applied for evaluating the overall adequacy of DID. The process implemented must reflect that comprehensive consideration was applied and implemented for each of the defined DID attributes, incorporating insights from deterministic analyses, probabilistic insights, operating experience, engineering judgment, etc. The description should address each of the decision guidelines described in Section 5.9.3 of NEI 18-04, including the basis for concluding the guideline has been met. For those guidelines where a quantitative measure can be provided (e.g. balance between accident prevention and mitigation, risk margins), the quantitative measures used in the decision-making should be provided.

This section should also include a description of how the results of the process are documented, and how those results will be incorporated into future IDP decision-

making during the facility's operation, including operations, maintenance program implementation, design activities, and the ongoing assessment of risk insights.

7.4.2. Baseline Evaluation of Defense in Depth
This section should describe the "baseline" level of defense in depth provided by the proposed facility. This baseline is established when the recurring evaluation of plant capability and programmatic capability associated with design and PRA update cycles no longer identifies risk-significant vulnerabilities where potential compensatory actions can make a practical, significant improvement to the LBE risk profiles or risk-significant reductions in the level of uncertainty in characterizing the LBE frequencies and consequences. This baseline DID evaluation and its outcome are to be documented in sufficient detail to assure that future changes to physical, functional, operational, or programmatic features of the facility can be effectively evaluated for their potential for reduction of DID before proceeding.

7.4.3. Evaluation and Incorporation of Changes to Defense in Depth
The change control process should be described addressing how the baseline DID evaluation will be re-evaluated, based on proposed changes, to determine which programmatic or plant capability attributes have been affected for each layer of defense. Changes that impact the definition and evaluation of LBEs, safety classification of SSCs, or risk significance of LBEs or SSCs must be assessed via the IDP described in Section 7.4.1. This section should also describe how any changes to the baseline DID evaluation will be documented and implemented.

8. **Control of Routine Plant Radioactive Effluents and Solid Waste**

The routine operation of a NPP generates liquid, gaseous and solid waste which must be contained and properly stored or disposed of. 10 CFR 20 sets limits on the activity of liquid and gaseous waste which can be released into the environment. 10 CFR 50 contains guidelines for LWR releases of liquid and gaseous waste to the environment that are consistent with the policy of ALARA. 10 CFR 61 describes the classes of low level waste and acceptable packaging for its disposal, as a function of its composition and activity level. Accordingly, each reactor design must have waste management systems that ensure the requirements of 10 CFR 20, 50 and 61 are met, or propose alternative requirements consistent with the technology of the proposed design.

These waste management systems are to be described in the application in sufficient detail to allow the NRC staff to conclude that the requirements are met. and that there is reasonable assurance public health and safety is protected. The level of detail provided in the application should be commensurate with the safety significance of the associated classification of the SSCs.

The guidance provided below summarizes, at a high level, the type of information that should be provided in the application

8.1. Liquid effluents
The application should include a description of the design's Liquid Waste Management System (LWMS) that includes the items listed below. For additional detail on the information that may

be appropriate for the application, see the LWR guidance provided in NUREG-0800, Section 11.2, "Liquid Waste Management System".

- The design features that are necessary for collecting, handling, processing, treating, storing and releasing or disposing of liquid waste, including P&IDs.
- Confirmation that failure of the LWMS will not prevent safe shutdown of the reactor or compromise any safety significant SSC.
- The LWMS design conditions (e. g. temperature, pressure, materials of construction, transient conditions, design codes or standards used).
- The equipment and system design capacities, flow rates, volumes and radionuclide removal efficiencies and the technical basis that indicates they are sufficient to support plant operation, including AOOs.
- The Quality Group and Seismic Classification applied in the design.
- The design provisions to control and collect any spillage or uncontrolled releases to the environment.
- The control features of the LWMS, including the location and types of sensors, the location of isolation valves and set points for automatic actions.
- A description of any shared features with other units.
- A summary of how the design (1) conforms with the release limits specified in 10 CFR 20, Appendix B, Table 2, Column 2, including Note 4 and (2) conforms with the ALARA provisions for liquid effluents specified in 10 CFR 50, Appendix I.

NOTE: alternate release criteria to those specified in 10 CFR 20 and 50 may be proposed if appropriate for the technology of the proposed design.

8.2. Gaseous effluents

The application should include a description of the design's Gaseous Waste Management System (GWMS) that includes the items listed below. For additional detail on the information that may be appropriate for the application, see the LWR guidance provided in NUREG-0800, Section 11.3, "Gaseous Waste Management System".

- The design features that are necessary for collecting, handling, processing, treating, storing and releasing or disposing of gaseous waste, including P&IDs.
- Confirmation that failure of the GWMS will not prevent safe shutdown of the reactor or compromise any safety significant SSC.
- The GWMS design conditions (e.g. temperature, pressure, materials of construction, transient conditions, design codes or standards used).
- The system design capacity relative to the expected input flow rates and volumes and the times the system is required to be in service to process the gaseous flow rates and volumes from normal operation, including AOOs.
- The Quality Group and Seismic Classification applied in the design.
- The control features of the GWMS, including the location and types of sensors, the location of isolation valves and set points for automatic actions.
- Design features to vent radioactive gases and vapors from tanks, vessels and processing equipment to appropriate exhaust ventilation and filtration systems.
- Design features to prevent internal explosions or detonations.

- The types and characteristics (e.g. decontamination factors, filtration efficiencies, hold up times) of filtration and absorbent material used in the design.
- A description of any shared features with other units.
- A summary of how the design (1) conforms with the release limits specified in 10 CFR 20, Appendix B, Column 2, including Note 4 and (2) conforms with the ALARA provisions for gaseous effluents specified in 10 CFR 50, Appendix I.

NOTE: alternate release criteria to those specified in 10 CFR 20 and 50 may be proposed if appropriate for the technology of the proposed design.

8.3. Solid radioactive waste

The application should provide a description of the design's Solid Waste Management System (SWMS) that includes the items listed below. For additional detail on the information that may be appropriate for the application, see the LWR guidance provided in NUREG-0800, Section 11.4, "Solid Waste Management System".

- The design features that are necessary for collecting, handling, processing, packaging and storing solid wastes. Including P&IDs.
- Confirmation that failure of the SWMS will not prevent safe shutdown of the reactor or compromise any safety significant SSC.
- Expected sources of waste (e.g. resins, sludge, filters, charcoal), waste composition (e.g. mixed waste), chemical make up, dry or wet and other important factors.
- The equipment design capacities for expected waste volumes and radioactivity inventories of Class A, B and C waste associated with normal operation, including AOOs.
- The Quality Group and Safety Classification applied in the design.
- The design provisions to control and collect any solid waste spillage from equipment malfunction or puncture of waste containers.
- Provisions for on-site waste storage, including expected volumes and radioactivity inventory.
- A description of any shared features with other units.
- A summary of how the solid waste packaging process conforms with the provisions of 10 CFR 61.55 and 56 regarding near surface disposal of low level waste.

9. **Control of Occupational Dose**

The application should describe the plant radiation protection design in sufficient detail to demonstrate that individual doses and total person-Sievert (person-rem) doses to plant workers, including construction workers, are maintained within the limits of 10 CFR Part 20. The application should also describe how security of licensed radioactive materials that are stored in controlled or unrestricted areas are maintained.

The application should describe the facility in sufficient detail to demonstrate that it meets the requirements of 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1203, 10 CFR 20.1204, 10 CFR 20.1206, and 10 CFR 20.1207.

9.1. Maintaining ALARA

9.1.1. Radiation Protection Design Features

The applicant should describe equipment and facility design features used to ensure that occupational radiation exposures are ALARA. Indicate whether and, if so, how the applicant has followed the design feature guidance given in Section C.2 of RG 8.8, "Information Relevant To Ensuring That Occupational Radiation Exposures At Nuclear Power Stations Will Be As Low As Is Reasonably Achievable." Conversely, if the applicant has not followed that guidance, describe the specific alternative approaches used. Also describe the design features provided to control access to radiologically restricted areas (including potentially very high radiation areas). Describe each very high radiation area and refer to its location on plant layout diagrams. Provide detailed drawings showing isometric views of each very high radiation area and indicate physical access controls and radiation monitor locations for each of these areas.

The applicant should include those features that reduce the need for maintenance and other operations in radiation fields, reduce radiation sources in areas where operations may be performed, allow quick entry and easy access, provide remote operation capability, or reduce the time spent working in radiation fields, as well as any other features that reduce radiation exposure of personnel. Also, include descriptions of methods for reducing the production, distribution, and retention of activation products through design, material selection, water chemistry, decontamination procedures, and so forth. Describe design features intended to minimize personnel dose during operation or maintenance. Describe how sampling ports, instrumentation, and control panels are located to facilitate access and minimize personnel exposure.

9.1.2. Operational Considerations

For COL applications, the applicant should describe the methods to be used to develop the detailed operational plans, procedures, and policies for ensuring that occupational radiation exposures are ALARA. Describe how these operational plans, procedures, and policies impact the design of the facility, and how such planning has incorporated information from operating plant experience, other designs, and so forth. Indicate the extent to which the plant follows the guidance on operational considerations given in RGs 8.8 and 8.10. Conversely, if the plant does not follow this guidance, describe the specific alternative approaches to be used.

9.2. Radiation Sources

The applicant should provide scaled layout and arrangement drawings of the facility. On these drawings, show the locations of all sources described in this section of the FSAR and identify those sources in a manner that can easily be related to tables containing the pertinent and necessary quantitative source parameters. Accurately locate positions, indicating the approximate size and shape of each source.

Descriptions should be provided for all radiation sources that require (1) shielding, (2) special ventilation systems, (3) special storage locations and conditions, (4) traffic or access control, (5) special plans or procedures, or (6) monitoring equipment. The source descriptions should include all pertinent information required for (1) input to shielding codes used in the design

process, (2) establishment of related facility design features, (3) development of plans and procedures, and (4) assessment of occupational exposure.

9.2.1. Contained Sources

The applicant should describe the sources of radiation, during normal plant operations and accident conditions, that are the bases for the radiation protection design. These sources should be described in the manner needed for input to the shield design calculation. This description should include isotopic composition, source strength and source geometry, and the bases for all values. The application should also describe those sources that are contained in equipment of the radioactive waste management systems. In this section, provide descriptions of other sources, such as the reactor core, spent fuel storage pool, and various auxiliary systems, equipment, and piping containing activation product sources. For the reactor core, describe the source as it is used to determine radiation levels external to the biological shield at locations where occupancy may be necessary.

Describe the contribution of neutron and gamma streaming to radiation levels in these potentially occupied areas of containment. Relevant experience from operating reactors may be used. For other sources, tabulate sources by isotopic composition or gamma ray energy groups, strength (becquerel or curie content), and geometry, and provide the bases for all values. For all sources identified above, including activation product sources, provide the models and parameters used to calculate the source magnitudes. Indicate whether and, if so, how the applicant has followed the applicable guidance provided in ANSI/ANS 18.1-1999. Conversely, if the applicant has not followed this guidance, describe the specific alternative methods used. Describe any required radiation sources containing byproduct, source, and special nuclear material that may warrant shielding design consideration. Provide a listing of isotope, quantity, form, and use of all sources in this latter category that exceed 3.7 E+9 Bq (100 millicuries). Describe any additional contained radiation sources that are not identified above, including radiation sources used for instrument calibration or radiography.

9.2.2 Airborne Radioactive Material Sources

The applicant should describe, and identify by location and magnitude, those airborne radioactive material sources in the plant that are considered in designing the ventilation systems and specifying appropriate monitoring systems. These sources should be described in the manner required for design of personal protective measures and for dose assessment. This description should include those sources of airborne radioactive material that are created by leakage in equipment cubicles, corridors, and operating areas that are normally occupied by operating personnel. Those airborne radioactivity sources that have to be considered for their contributions to the plant's effluent releases through the radioactive waste management system or the plant's ventilation systems should be described in Chapter 8 of the FSAR.

Describe how airborne radioactivity concentrations in frequently occupied areas are a small fraction of the concentrations related to 10 CFR 20.1203, 10 CFR 20.1204, and Appendix B to 10 CFR Part 20. The assumptions made in arriving at quantitative values for these various sources should be specified in this section.

Tabulate the calculated concentrations of airborne radioactive material by nuclides expected during normal operation, AOOs, and accident conditions for equipment cubicles, corridors, and operating areas normally occupied by operating personnel. Provide the models and parameters used for calculating airborne radioactivity concentrations.

10. **Human Factors Analysis**

As discussed in the Commission's Policy Statement on the Regulation of Advanced Reactors, one of the characteristics the Commission expects to see in advanced reactor designs is reduced reliance on operator actions and simplified systems that facilitate operator comprehension. Accordingly, human factors considerations are an important part of the design process. Therefore, the application should describe the human factors engineering (HFE) program applied in the design and operation, including how the HFE Program has been coordinated with the plant risk analysis and the features of the design and operation that result from the HFE program and contribute to reduced reliance on operator actions. The elements of the HFE program that need to be described in the application are summarized below. For more detailed guidance on the information which should be included in the application, refer to NUREG-0711, Rev. 3, "Human Factors Engineering Review Model".

10.1. Overview of the HFE Program

The HFE program goals and scope (e.g. control room, local control stations, etc.) should be described. In addition, the process used in conducting the HFE program should be described along with information on the team of HFE experts conducting the program. This should include items such as:

- Assumptions (e.g. staffing levels, types of I&C used in the design, etc.) used in conducting the program.
- Coordination of the HFE program with the overall plant design activities, including coordination with the plant risk analysis.
- Facilities (e.g. mockups) used in support of the program.
- Qualifications of the HFE personnel.
- HFE team organization and responsibilities.
- Process and procedures followed.
- Documentation developed.
- Summary of how the results of the HFE analysis have been incorporated into the design, operation and risk analysis.

10.2. Operating Experience Review

Operating experience can provide information useful to the HFE program. Therefore, a review of operating experience should be part of the HFE program. The results of the review of

operating experience applicable to the design should be described. This should include items such as:

- The criteria used to identify the operating experience applicable to the design.
- The operating experience reviewed, including recognized industry HFE issues (e.g. TMI issues, operating experience reports – NUREG-1275 series, etc.).
- The process used in conducting the review (e.g. documentation reviews, interviews, etc.)
- The results of the operating experience reviews, including how the design is addressing the human factors issues identified by the operating experience reviews.

10.3. Safety Function Review

The safety functions that must be performed to maintain the plant within the safety envelope described in the PRA and the Chapter 6 (Design Basis Accidents) safety analysis should be identified, analyzed and described in the application. This analysis should include items such as:

- Definition of the safety functions and the SSCs used to accomplish them.
- Identification of the most important human actions resulting from the plant risk analysis.
- Allocation of the safety functions to human actions, automation or a combination of the two.
- Methodology used in making the allocations.
- Technical bases for the allocations made.
- Comparison to human vs. automation allocations made on previous predecessor plants of similar design.

10.4. Human Action Task Analysis

For those tasks that require human action to ensure the accomplishment of safety functions, a detailed analysis is necessary to determine the elements that are involved in the human actions. This analysis should be described in the application and include items such as:

- Identification of the most important human actions and the methodology used to identify them (e. g. risk analysis, deterministic considerations).
- Identification of the information (e.g. instrumentation, alarms, procedures, etc.) needed to perform the actions.
- Identification of the physical parameters (time available for the action, physical movement needed for the action, number of people needed to perform the action, knowledge required, etc.) involved in the actions, including overall workload involved.
- Identification of personnel qualifications needed to perform the actions.
- Identification of how the HFE Program was used to reduce the likelihood of human error in performing these tasks.

10.5. Human-System Interface Design

It is likely that all of the human actions will involve some interface with instrumentation, procedures, system controls and communications equipment. It is important that the human-system interface (HSI) with that equipment be designed to minimize the likelihood of errors in comprehension and actions. Accordingly, the application should describe items such as:

- The approach used in conducting the HIS design, including the guidance used in the design process.

- The HSI features incorporated into the design of the Main Control Room, the Technical Support Center, the Emergency Operations Facility, the Remote Shutdown Facility and local control stations.
- The results of tests and evaluations undertaken to support the HSI design.

10.6. Procedures, Training and V & V

Procedures, training and verification and validation (V & V) of the HSI features should be developed and performed to ensure the HSI design features are integrated into the overall plant operations and result in human actions consistent with assumptions in the PRA and deterministic safety analysis. The application should describe items such as:

- The process used for procedure development, training program development and V & V.
- A description of the complete set of scenarios used for V & V.
- The results of the V & V performed.

10.7. Human Performance Monitoring

Human performance should be monitored over the life of the plant for degradation and for consistency with assumptions in the PRA and deterministic safety analysis. The application should describe the human performance monitoring program planned for the facility, including the criteria to be used to determine human performance degradation, the process to be used to collect human performance data and the frequency of evaluating the human performance data.

11. **Physical Security**

For design certification (DC) applications, physical security systems, components, and measures identified to be within the scope of an applicant's design should be described in this section. The information should encompass the material intended to meet the general performance objective as described in Title 10 of the Code of Federal Regulations (CFR), Paragraph 73.55(b).

The security design may utilize a performance-based approach in determining the applicability of alternatives to the prescribed minimum number of armed responders currently defined in 10 CFR 73.55(k)(5)(ii) and the prescriptive requirements defined in 10 CFR 73.55(i)(4)(iii) for an onsite secondary alarm station. However, this section should describe the analysis that concludes the design meets any performance criteria that are established by [future] rulemaking to determine the applicability of the alternative physical security requirements. The analysis description should include the radiological consequences from a hypothetical, unmitigated event involving the loss of engineered systems for decay heat removal and possible breaches in physical structures surrounding the reactor, spent fuel, and other inventories of radioactive materials result in offsite doses below the reference values defined in 10 CFR 50.34 and 52.79 (e.g., no definable target sets of equipment or operator actions that if prevented from performing their intended safety function or prevented from being accomplished, would likely result in offsite doses exceeding the cited reference values).

The description of the performance-based analysis also needs to include either:

- the plant features necessary to mitigate an event and maintain offsite doses below the reference values in 10 CFR 50.34 and 52.79 and demonstrate that the facility cannot reasonably be compromised by the design basis threat (DBT) for radiological sabotage (e.g., no achievable target set resulting in offsite doses exceeding the cited reference values given the design features and security features incorporated into a specific advanced reactor facility); or
- demonstrate that plant features include inherent reactor characteristics combined with engineered safety and security features that allow for facility recovery and mitigation strategy implementation if a target set is compromised, destroyed, or rendered nonfunctional, such that offsite radiological consequences are maintained below the reference values defined in 10 CFR 50.34 and 52.79 (e.g., a reactor design with a large heat capacity and slow progression from loss of safety equipment to degradation of fission product barriers and release of radionuclides from the facility). Facility recovery and mitigation strategies may, where feasible, include support from offsite resources.

As necessary to support the above analysis, the application should include text and figures that identify and depict vital areas that are within the physical design of the power reactor with adequate visual clarity to allow for review. Scale diagrams, figures, drawings, etc. should provide a clear visual depiction of the facility and site physical layout, to include, but not limited to the following: physical structures and their location on the site, owner-controlled area, protected area(s), vital areas and vital equipment, physical barriers, alarm stations, access control points, parking lots relative to the areas adjacent to physical barriers surrounding protected areas, and any special features of the terrain which may present special vulnerability problems. The site and facility layout should also include a visual depiction of the site in relation to nearby towns, roads, and other environmental features important to the effective coordination of response operations.

This section of a COL application should include a discussion indicating that a security plan has been prepared and submitted separately to the NRC. The details of a plant's overall security plan should include a description of the elements of the individual security plans (e.g., physical security, training and qualification, and safeguards contingency) proposed by a COL applicant, as required by 10 CFR 73.55, "Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage." In addition, the security plan for a COL applicant should describe the proposed site security provisions implemented during construction of a new plant that is either inside an existing protected area, owner-controlled area, or is a greenfield site. The COL applicant should identify the schedule implementation requirements associated with the elements of its security plan and security assessment. In addition, the COL applicant should address, in this section, any COL action items or information items applicable to the security plan and security assessment that may have been established for ESPs and/or certified designs that are referenced in the COL application.

The COL applicant should also submit the following information:

- a proposed schedule for implementing the site's operational security programs, security systems and equipment, and physical barriers
- proposed ITAAC for physical security hardware

The security design attributes listed below should be described in sufficient detail to demonstrate compliance with the general performance objective as described in 10 CFR 73.55(b) or support the performance-based analysis described above.

11.1. Physical barriers

Provide a design description of the physical barriers that protect vital areas (e.g., walls, ceilings, floors, doors, and gratings that comply with the definition of physical barrier provided in 10 CFR 73.2, "Definitions"), and the personnel/vehicle/material portal(s) into the owner controlled areas and describe how these barriers meet the applicable requirements of 10 CFR 73.55(e). 10 CFR 73.2, 10 CFR 73.55(e).

Provide a design description (consistent with the stated function of the barrier) that confirms that openings in any barrier or barrier system established to meet the requirements of 10 CFR 73.55, "Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors against Radiological Sabotage," are secured and monitored to prevent the exploitation of the opening 10 CFR 73.55(e)(4).

The physical barriers at the perimeter shall be separated from any other barrier designated as physical barrier for a vital area within the protected area. Isolation zones in outdoor areas adjacent to the physical barrier at the perimeter of the protected area permit observation. Intrusion detection system detects penetration or attempted penetration of the protected area (PA) barrier. The external walls, doors, ceiling and floors in the main control room are bullet resistant. Vehicle control measures which include vehicle barrier systems protect against the use of land vehicle.

11.2. Vital areas

Identify vital areas within the physical design and provide a list of vital equipment, by component and location, in vital areas. 10 CFR 73.55(e)(9). List and describe all vital equipment and locations. Describe the design features to be used for protecting all potential access points into vital areas against authorized intrusion. Such features should include locking devices and intrusion detection devices.

Vital equipment should be listed by component and identified by location within vital areas. Describe how vital equipment is only located within a vital area, which in turn, shall be located within a protected area such that access to vital equipment requires passage through at least two physical barriers as defined in 10 CFR 73.2.

Descriptive text and figures should identify and depict the control room and spent fuel pool as located within a vital area. A design description of the control room must identify that the materials for the physical barriers (e.g., walls, floors, ceiling, and doors) that protect the control room will be bullet-resisting. Vital area access controls should meet the guidance in Regulatory Guide 5.65, "Vital Area Access Controls, Protection of Physical Security Equipment and Key and Lock Controls."

11.3. Detections aids

Describe intrusion alarms, emergency exit alarms, duress alarms, alarm systems, and line supervisory tamper indicating systems, and communication systems. Provide a design description that confirms that all vital area portals and emergency exits are equipped with intrusion detection equipment and locking devices that allow for rapid egress during an emergency and that satisfy the vital area access requirements of 10 CFR 73.55. Describe how all alarms required pursuant to this part shall annunciate in a continuously manned central alarm station located within the protected area and in at least one other continuously manned station, not necessarily onsite, such that a single act cannot remove the capabilities of calling for assistance or otherwise responding to an alarm. The central alarm station shall be considered a vital area, shall be bullet-resisting, the interior will not be visible from the protected area perimeter, and associated onsite secondary power supplies for alarm annunciators and non-portable communication equipment must be located within vital areas. Alarm devices and transmission lines should be tamper indicating and self checking. Alarm annunciation shall indicate type of alarm and location. All emergency exits from protected and vital areas shall be alarmed. Describe how the design meets the guidance in Regulatory Guide 5.44, 'Perimeter Intrusion Alarm Systems."

11.4. Communication

Describe how communications systems meet Section (f) of 10 CFR 73.55 - Communication Requirements. Describe how each security officer, watchman or armed response individual is capable of maintaining continuous communications with an individual in each continuously manned alarm stations. Describe how conventional telephone and radio or microwave transmitted two-way voice communications is provided with local law enforcement authorities.

Describe the provisions to be utilized in the design for the protection of security system service panels, conduits and wiring devices integral to security communication systems, intrusion detection systems, security power systems, security digital systems and other related physical security features.

11.5. Access controls

Describe how the design meets Section (d) of 10 CFR 73.55 - Access Requirements, including how the applicant plans to control all points of personnel and vehicle access into a protected area, to include detection equipment capable of detecting firearms, explosives and incendiary devices. Describe how unoccupied vital areas are locked and alarmed with activated intrusion detection systems that annunciate in both the central and secondary alarm stations upon intrusion into a vital area. Describe how the design meets the guidance in Regulatory Guide 5.12, "General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials" and Regulatory Guide 5.7, "Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas."

11.6. Security lighting

Describe how lighting is credited in intrusion detection and assessment and how that equipment remains operable from a UPS during the loss of normal power. Describe how all exterior areas within the protected area are illuminated.

11.7 Safety-Security Interface

Describe how the design includes considerations for safety and security requirements together in the design process such that security issues (e.g., newly identified threats of terrorist attacks) can be effectively resolved through facility design and engineered security features, and formulation of mitigation measures, with reduced reliance on human actions. This section should describe how the guidance in Regulatory Guide 5.74 is met.

12. **Overview of PRA** (later)

13. **Administrative Control Programs (* COLA only)**

The purpose of this chapter is to describe the Administrative Control Programs the applicant intends to use to ensure that the plant management organization is properly structured with clear lines of responsibility, that the operating staff is properly trained, that plant operations are conducted using verified procedures, that maintenance is conducted on plant SSCs to maintain their reliability and performance, that equipment reliability is monitored for consistency with the PRA and that there is a process in place to control changes to the plant to ensure consistency with the design basis. It is important to note that many of these administrative controls have a connection to the PRA in that the basis for much of their content will come from the plant risk analysis and assumptions.

13.1. Organization*

The operating organization should be described in the application. This should include:

- A description of each management position in the operating organization, including the job responsibilities. If the application is for multiple units, the description must address any multi-unit responsibilities and interfaces.
- The educational, training and experience required for each management position, including a comparison to the guidance in USNRC RG 1.8, “Qualification and Training of Personnel for Nuclear Power Plants”.
- The interfaces with support groups (e.g. Technical Support Center, Corporate).
- The lines of authority and communication among the organizational units and support groups.
- The number of operating shift crews, their staffing and responsibilities.
- The basis for the operating shift crew staffing levels and structure

13.2. Training of personnel*

The licensed operator training program should be described in the application. This should include the plans for initial operator qualification as well as requalification. It is recognized that the program may not be fully developed at the time of the application submittal, in which case any incomplete items can be verified during the Construction Inspection Program. The description should include:

- A commitment to USNRC RG 1.8, including a description and bases for any deviations.
- A summary of the training program, including the industry standards used, the material covered for each licensed operator position, the examination methods to be used and when the program will be in place.

- A description of any plant simulator facility used in the training and how it has been verified to accurately simulate the plant response to normal and off normal conditions.
- A description of any outside accreditation to be sought for the program and the periodic evaluation to be conducted on the program to ensure it remains effective and in compliance with the standards used, including who will conduct the evaluation.

13.3. Conduct of operations*

A description of the controls and procedures for the conduct of operations should be provided. This description should include:

- How procedures are to be used to guide the conduct of actions taken for all plant activities (e.g. startup, shutdown, response to off-normal conditions, refueling, maintenance, surveillance, plant modifications, record keeping).
- Approvals needed to initiate a plant activity or to change from one operating mode to another.
- The process for development and verification of the procedures and for modifying procedures, both in advance of an activity and while an activity is underway.
- Communications required among the various areas of the plant organization (e.g. operations, maintenance, engineering) during the conduct of the various plant activities.
- How risk considerations are factored into decisions regarding the conduct of operations.
- The process for evaluating operating events and inspection results, including how risk information is used in the evaluation to determine safety significance and the process for determining the appropriate corrective actions

13.4. Reliability Assurance Program

A Reliability Assurance Program (RAP) should be designed to confirm whether or not the plant is operating in a manner consistent with the risk insights and assumptions used in the PRA. NUREG-0800, Section 17.4 provides a discussion on the RAP required by 10 CFR 52 and should be referred to for additional detail. In summary, the application should describe:

- Who in the plant organization is responsible for implementation and evaluation of the RAP.
- A description of how the equipment to be monitored was selected and its relation to the plant risk analysis (e.g. selected by importance measures).
- A listing of the equipment included in the RAP, the parameters being monitored and the frequency of monitoring.
- A description of how the data collected is evaluated to determine the equipment reliability and its uncertainty.
- A description of how the reliability and uncertainty information is used (e.g. how is it determined that equipment reliability is consistent with the PRA, what is done if inconsistences are found). This should include assessing the impact of the reliability information on the PRA and its risk insights (e.g. LBE selection, SSC safety classification, DBAs).

13.5. Maintenance Program*

10 CFR 50.65 requires that a maintenance program be established, equipment performance be monitored and corrective action be taken when degraded equipment performance is

detected. USNRC RG 1.160 provides additional guidance on one way to meet the requirements of 10 CFR 50.65. The maintenance program and the RAP interface with each other and, in fact, the RAP may be implemented by the maintenance program. The information that should be included in the application regarding the maintenance program is essentially the same as for the RAP described in Section 13.4 above.

13.6. Change control process (to monitor performance and manage SSC categorization changes)
The application should describe how changes to the plant configuration (either temporary or permanent) are approved and controlled so that all plant personnel affected by the changes are aware of the changes. Also, the process should include provisions for assessing those changes against the PRA and its risk insights (e.g. SSC safety significance) for any impact on LBE selection or definition, SSC safety classification or DBAs.

14. **Initial Startup Programs**

This Chapter should discuss the applicant's plan to address preoperational and initial startup/operations testing as required under 10 CFR 50.34(b)(6)(iii). Discussions should also consider Section XI of Appendix B to 10 CFR Part 50, which requires a test program to assure SSCs will perform satisfactorily in service. To the extent the requirements apply, discussions should also address Section III.A.4 of Appendix J to 10 CFR Part 50, relating to preoperational leakage rate testing of the primary reactor containment and related systems and components penetrating the primary containment pressure boundary; testing related to "functional containment" as an alternative to a fixed containment structure should also be identified and discussed as appropriate.

Additional discussions should address requirements under 10 CFR 30.53, "Tests," as it relates to testing radiation detection equipment and monitoring instruments.

The initial testing program should clearly describe objectives and offer descriptions for each major phase of the test program as is noted in Chapter 14-2 of NUREG-0800. Because discussions will include key testing of SSCs, extensive interface with discussions contained in Section 5, Description and Classification of SSCs, and Section 12, Overview of PRA, of this guidance document will be required.

14.1. As-built verification program (ITAAC)
This section will address Section XI of Appendix B to 10 CFR 50 and (for COL applicants) 10 CFR 52.47(b)(1). Discussions should center on DC applications that contain proposed inspections, tests, analyses, and acceptance criteria (ITAAC) necessary and sufficient to provide adequate assurance that a plant meeting said ITAAC is built and will operate in accordance with the design certification.

Additional discussions should also address proposed inspections, tests, and analyses requirements for emergency planning, as is required under 10 CFR 52.80(a).

14.2. Preoperational testing program
This section should discuss the need for COL applicants to provide plans under 10 CFR 50.34(b)(6)(iii) and (for COL applicants) 10 CFR 52.79(a)(28) concerning for preoperational

testing and initial operations. Preoperational tests are tests done following completion of construction and construction-related inspections and tests, but before fuel loading. These tests are intended to demonstrate, to the extent practicable, the capability of SSCs to meet specified performance requirements and design criteria.

14.3. Initial startup testing/operations program
This section should discuss initial startup tests as outlined in NUREG-0800, Chapter 14.2. . These test activities are scheduled for performance during and following fuel activities and include fuel loading, precritical tests, initial criticality, low-power tests, and power ascension tests that confirm the design bases and demonstrate, to the extent practicable, that the plant will operate in accordance with its design and is capable of responding as designed to anticipated transients and postulated accidents. Adequate control over the evaluation and approval of test prerequisites, protocols, and results should be identified.

Separate Licensing Documents

DC and COL Application (if not referencing a DC)

- Technical Specifications
- Technical Requirements Manual
- Quality Assurance Plan (design)
- Fire Protection Program (design)
- PRA
- Fuel qualification report
- Exemptions
- Environmental Report

COL Application only

- Quality Assurance Plan (construction and operations)
- Emergency Plan
- Physical Security Plan
- SNM (special nuclear materials) physical protection program
- SNM material control and accounting plan
- Cyber Security Plan
- New fuel shipping plan
- Fire Protection Program (operational)
- Radiation Protection Program
- Offsite Dose Calculation Manual
- ISI/IST Program
- Environmental Report
- Site Redress Plan
- Exemptions, Departures, and Variances