From: Sebrosky, Joseph
Sent: Thursday, June 10, 2021 9:19 AM
To: Afzali, Amir
Cc: Steven Nesbit; Michael Tschiltz; Chisholm, Brandon Michael; Jung, Ian; Uribe, Juan; Oesterle, Eric; Reckley, William; Segala, John; Sanfilippo, Nathan; Shams, Mohamed; Jim C. Kinsey Jr; 'Tom King'; Thomas Hicks; Christopher P. Chwasz; NICHOL, Marcus; AUSTGEN, Kati; Cyril Draffin; Smith - NRR, Brian
Subject: NRC Staff Detailed Comments on Industry-Developed Draft Technology Inclusive Content of Application Project Guidance Document
Attachments: NRC staff comments on Southern Draft TICAP Guidance Document RevB for NRC 04-15-21.docx

To: Amir Afzali
Southern Company Services
Licensing and Policy Director- Next Generation Reactors

The purpose of this email is to provide U.S. Nuclear Regulatory Commission (NRC) staff draft comments on an industry-developed draft technology inclusive content of application project (TICAP) guidance document dated April 15, 2021. The industry-developed TICAP guidance document can be found in the Agencywide Documents Access and Management System (ADAMS) under Accession Number ML21106A013. This email will be captured in ADAMS and made publicly available.

Background
In an April 15, 2021, email (ADAMS Accession No. ML21106A013), industry provided a draft TICAP guidance document. This guidance document was the subject of publicly noticed workshops on May 11 (see: https://www.nrc.gov/pmns/mtg?do=details&Code=20210516), May 19 (see: https://www.nrc.gov/pmns/mtg?do=details&Code=20210574) and May 26 (see: https://www.nrc.gov/pmns/mtg?do=details&Code=20210579). During these workshops the NRC staff noted that in addition to the items discussed during the workshops it was developing detailed comments on the industry developed TICAP guidance document.

Current Status
The attached word document contains the detailed NRC staff developed draft comments. It should be noted that the attached comments have not been subjected to NRC management or legal reviews. In addition, the comments were developed in parallel with the workshops and some of the comments therefore do not reflect the outcome of the issues discussed during the workshops. Nevertheless, the NRC staff believes that it is appropriate to provide the attached document for industry's consideration (along with the feedback provided during the workshops) as industry considers revisions to the April 15, 2021, TICAP guidance document.

Please let me, Eric Oesterle, or Juan Uribe, know if you have any questions regarding the attached document.

Sincerely,
Joe Sebrosky
Senior Project Manager
Advanced Reactor Policy Branch
Office of Nuclear Reactor Regulation

Hearing Identifier: NRR_DRMA
Email Number: 1226

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

Technology Inclusive Guidance for Non-Light Water Reactor Safety Analysis Report: Content for a Licensing Modernization Project-Based Affirmative Safety Case

Draft Report Revision B
Issued for Idaho National Laboratory Review and Comment

Document Number
SC-16166-104 Rev A

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

April 9, 2021

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

Disclaimer

Abstract

This guidance document describes one acceptable means of developing portions of the Safety Analysis Report content for some advanced reactors. Another Nuclear Energy Institute publication, NEI 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," describes the Licensing Modernization Project methodology for selection of Licensing Basis Events; safety classification of structures, systems, and components and associated risk-informed special treatments; and determination of defense-in-depth adequacy for non-light water reactors. The NEI 18-04 guidance was endorsed in Nuclear Regulatory Commission Regulatory Guide 1.233, "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." The guidance in this report focuses on the portions of the Safety Analysis Report that relate to the application of the NEI 18-04 methodology. The goal of the standardized content structure and formulation is to facilitate efficient preparation by the applicant, review by the regulator, and maintenance by the licensee.

Table of Contents

List of Abbreviations

AOO	Anticipated Operational Occurrence
ARRTF	Advanced Reactor Regulatory Task Force
BDBE	Beyond Design Basis Event
CDC	Complementary Design Criteria
CP	Construction permit
COL	Combined construction permit and operating license
DBA	Design Basis Accident
DBE	Design Basis Event
DC	Design certification
DID	Defense-in-Depth
DOE	Department of Energy
FSAR	Final Safety Analysis Report
FSF	Fundamental Safety Function
IBR	Incorporation by Reference
IDP	Integrated Decision-Making Panel
LBE	Licensing Basis Event
LMP	Licensing Modernization Project
LWR	Light water reactor
NEI	Nuclear Energy Institute
MHTGR	Modular High Temperature Gas-Cooled Reactor
non-LWR	Non-light water reactor
NRC	Nuclear Regulatory Commission
NSR	Non-Safety-Related
NSRST	Non-Safety-Related with Special Treatment
NST	No Special Treatment
OL	Operating license
PDC	Principal Design Criteria
PRA	Probabilistic Risk Assessment
PSF	PRA Safety Function
Ref	General references
RG	Regulatory Guide
RIPB	Risk-informed and performance-based
RSF	Required Safety Functions
SAR	Safety Analysis Report
SR	Safety-Related
ST	Special Treatment
SRDC	Safety-Related Design Criteria
SSCs	Structures, Systems, and Components
TICAP	Technology Inclusive Content of Application Project

Commented [A1]: Need to add "ARCAP"

Commented [A2]: Need to add "RFDC" to the list.

A. INTRODUCTION

Non-light water reactor (non-LWR) technologies will play a key role in meeting the world's future clean energy needs and are building on the foundation established by the current light water reactor (LWR) nuclear energy fleet. Given the long timeframe and significant financial investment required to mature and deploy these technologies, an efficient and cost-effective non-LWR-licensing framework that facilitates safe and cost-effective construction and operation is a critical element for incentivizing private sector investment. The Technology Inclusive Content of Application Project (TICAP) is an important part of the nuclear industry efforts to support the Nuclear Regulatory Commission (NRC) and Department of Energy (DOE) initiatives to establish that licensing framework. This DOE cost-shared, owner/operator-led initiative produced this guidance document for developing content for portions of the NRC license application Safety Analysis Report (SAR) for non-LWR designs related to the Licensing Modernization Project (LMP)-based affirmative safety case (see below and Section 1.3).

This guidance is applicable to applicants that utilize the Licensing Modernization Project (LMP) methodology documented in the Nuclear Energy Institute (NEI) publication NEI 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development" (ML19241A336). The NEI 18-04 guidance was endorsed in NRC Regulatory Guide 1.233, "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" (ML20091L698).

Purpose

The guidance in this report is focused on the portions of the SAR containing material produced by implementing NEI 18-04. The intent of the guidance is to help ensure completeness of information submitted to NRC while avoiding unnecessary burden on the applicant and rightsizing the content of application commensurate with the complexity of the design being reviewed.

This guidance provides a standardized content development process and application format designed to facilitate efficient preparation by the applicant, review by the regulator, and maintenance by the licensee. The content formulation should optimize the type and level of detail of information provided in the SAR, based on the complexity of the design's safety case and the nexus between elements of the design and public health and safety.

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

- Technology inclusive to be generically applicable to all non-LWR designs
- Risk-informed and performance-based (RIPB) approach to:
    - Ensure the NRC review is focused on information that directly supports the safety case of nuclear power plants.

- Provide a consistent and coherent approach for establishing portions of the SAR scope and level of detail guidelines for various advanced technologies and designs.
- Encourage innovation by focusing more on the final results ~~as opposed to~~than the pathway taken to achieve the results.

This proposed, technology inclusive RIPB license application content should advance the following:

- The goal of having a safety-focused review that minimizes the burden on developers and owner-operators of generating, supplying, and maintaining safety-insignificant information
- The NRC and industry objective of reaching agreement on information needed in a SAR to demonstrate reasonable assurance of adequate protection for non-LWRs
- NRC's stated objective and policy statement regarding the use of risk-informed decision-making to remove unnecessary regulatory burden

NEI plans to submit this guidance document to NRC for review and endorsement as one acceptable approach for the development of those portions of the Safety Analysis Report required for a combined construction and operating license (COL), a reactor construction permit (CP) followed by an operating license (OL), or design certification (DC) that employs the LMP methodology endorsed by Regulatory Guide 1.233.

Background

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

Most of the currently operating nuclear power reactors were initially licensed in the 1970s and 1980s. The regulatory framework for those plants was developed over decades and tailored specifically for thermal neutron spectrum LWRs using light water coolant and moderator, zirconium-clad uranium oxide fuel, and the Rankine power cycle. Many advanced non-LWRs are in development, with each reactor design differing significantly from the current generation of LWRs. For example, advanced reactors might employ liquid metal, gas, or molten salt as a coolant, enabling them to operate at lower pressures but higher temperatures than LWRs. Some will use a fast or epithermal rather than just a thermal neutron spectrum. A range of fuel types is under consideration, including fuel dissolved in molten salt and circulated throughout the reactor coolant system. In general, advanced reactors emphasize passive safety features that do not require operator action or rapid automatic action from powered systems to prevent or mitigate radionuclide releases. Materials may be different, particularly for the high-temperature reactors.

Commented [A3]: I'm a little worried that this gives the impression that the methodology isn't as important as the results. I hope this isn't what is intended since the methodology is extremely important to DBA and other analyses.

Commented [A4R3]: I agree. Perhaps language that conveys "focuses on documenting the final results in a SAR, and including the pathway taken to achieve the results in supporting documentation"

Commented [A5]: If it is safety-significant, then it is important to generate, supply, and maintain the information. Perhaps this bullet intends to say that it minimizes the amount of non-safety-significant information that is included in the SAR?

Commented [A6]: No ML?

Advanced reactors may produce energy for applications other than electricity generation, and they may be coupled to energy storage systems. Given these technical differences, applying the current regulatory framework to advanced reactor designs would be difficult and inefficient. Changes to the current regulatory framework are needed to allow for a risk-informed safety evaluation and timely, efficient deployment of advanced reactor designs.

The DOE cost-shared TICAP, a utility-led project, was initiated to interact with NRC in support of the objective of modernizing the regulatory framework to improve the effectiveness and efficiency of NRC reviews. The project team included reactor owner-operators, reactor designers, and consultants as well as a senior advisory group consisting of several former NRC commissioners. This guidance document reflects feedback received from stakeholders as part of several reviews and interactions. In addition, the team worked with four reactor designers to perform tabletop exercises that applied portions of preliminary TICAP guidance to develop notional SAR content. The final guidance in this document benefits from the lessons learned from those tabletop exercises.

TICAP built on the foundation that was successfully established in NEI 18-04 Rev 1, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development". That document presented a technology inclusive, RIPB process for selection of Licensing Basis Events (LBEs); safety classification of Structures, Systems, and Components (SSCs); and evaluation of defense-in-depth (DID) adequacy for non-LWRs based on a systematic evaluation of the safety case. NRC endorsed the NEI 18-04 guidance with the publication of Regulatory Guide 1.233. Although NRC staff expectations were delineated, this regulatory guide took no significant exceptions to the LMP methodology as described in NEI 18-04. The TICAP guidance contained herein focuses on the portion of the application related to the outcomes of applying the LMP approach~~LMP~~ and the documentation of the applicant's safety case. Ultimately, the information included in an application must demonstrate that the safety case of a particular design provides reasonable assurance of adequate protection of public health and safety.

This guidance document was developed as part of a two-year effort by Southern Company-led team composed of reactor owner-operators, reactor designers, and consultants. A senior advisory group consisting of several former NRC commissioners oversaw the effort. As part of the development process, the team interacted extensively with the NEI Advanced Reactor Regulatory Task Force (ARRTF), other industry stakeholders, and NRC. The team issued intermediate products covering key aspects of the guidance and provided them for ARRTF and NRC review and comment. This guidance document reflects feedback received from stakeholders as part of these reviews and interactions. In addition, the team worked with four reactor designers to perform tabletop exercises that applied portions of preliminary TICAP guidance to develop notional SAR content. The final guidance in this document benefits from the lessons learned from those tabletop exercises.

Scope

The baseline guidance presented in this document assumes an applicant is applying for a Combined License (COL) under 10 CFR Part 52, Licenses, Certifications, and Approvals for Nuclear Power Plants. The guidance further assumes that the applicant is not referencing an existing Design Certification (DC). Since the level of detail and design finality is the same for an Operating License (OL) application as a COL applicant not referencing a DC, the scope of this guidance also applies to an OL applicant under 10 CFR Part 50. Supplemental guidance also addresses two other ~~additional~~ licensing approaches:

- Construction Permit ~~Two-step license~~ (CP) application~~and OL)~~ under 10 CFR Part 50
- Design certification under 10 CFR Part 52

This document provides guidance on the following:

- Scope of content to be included in an application (specifically, portions of the SAR)
- Level of detail for the content
- Structure to be used for providing the content

The guidance on the SAR content scope and level of detail prescribes an appropriate level of design-specific information that should be provided to demonstrate that the design's safety case meets the regulatory standards for reasonable assurance of adequate protection of public health and safety. To accommodate an effective and efficient technology inclusive content guidance while ensuring the underlying intent of the current content requirements is met, this guidance is formulated to describe an LMP-based affirmative safety case, defined as follows:

> An affirmative safety case is a collection of technical and programmatic evidence which documents the basis that the performance objectives of the technology inclusive Fundamental Safety Functions (FSFs) are met by a design during design specific Anticipated Operational Occurrences (AOOs), Design Basis Events (DBEs), Beyond Design Basis Events (BDBEs), and Design Basis Accidents (DBAs). This is accomplished by the following:
>
> - Identifying design-specific safety functions that are adequately performed by design-specific SSCs
>
> - Establishing design-specific features (programmatic, e.g., inspections, or physical, e.g., diversity) to provide reasonable assurance that credited SSC functions are reliably performed and to demonstrate the adequacy of defense-in-depth

The term safety case is a collection of statements that, if confirmed to be true by supporting technical information, establishes reasonable assurance of adequate protection for operation of the nuclear power plant described in the application. An affirmative safety case is a holistic

Commented [A7]: Isn't the level of design detail for an OL/COL and a DC similar?

Commented [A8]: Since the level of detail and design finality for an OL and COL (not referencing a DCD) is the same, perhaps the scope of the guidance should apply to OL applicants and COL (w/o DCD) applicants. This would make the supplemental guidance specific to CP applicants and DC applicants.

Commented [A9]: Why not a ML also?

Commented [A10]: Regulatory compliance?

Commented [A11]: Comment would seem to question the accuracy of this statement; i.e., reasonable assurance is met when all of the regulations are met. Additionally, there are ARCAP application components that are not part of this safety case that are also needed for reasonable assurance.

approach that focuses on demonstrating that a set of fundamental safety functions will be accomplished. It may be contrasted with a traditional compliance-oriented safety case that demonstrates the satisfaction of pre-established requirements using a prescribed set of processes or equipment.

The use of the LMP-based affirmative safety case to formulate the application content will optimize the following:

- The scope of information to be included based on relevance to the design-specific safety case
- The type of information to be provided based on the LMP-based affirmative safety case elements that are structured to be consistent with the current application content requirements for LWRs
- The level of detail formulation based on the importance of the functions and SSCs to the LMP-based affirmative safety case (RIPB details) and the relevance to the safety determination

The content structure facilitates efficient (i) preparation by an applicant, (ii) review by the regulator, (iii) maintenance by the licensee, and (iv) ease of use by stakeholders, including the public.

This guidance addresses only the portion of an advanced reactor SAR related directly to the application of the NEI 18-04 methodology. Concurrent with the development of this document, the NRC is developing guidance for other parts of an advanced reactor license application (including part of the SAR) in its Advanced Reactor Content of Application Project (ARCAP).[1] With respect to the SAR portion of an advanced reactor application, this TICAP guidance pertains to most of the content in Chapters 1 through 8, while ARCAP provides guidance for Chapters 9 through 12 as well as portions of some of the earlier chapters. This SAR organization is significantly different from the approach that has evolved for large light water reactors. ARCAP also provides guidance for non-SAR parts of an application. The relationship this TICAP guidance to ARCAP guidance is shown pictorially in Figure 1.

[1] Slides from the February 25, 2021 NRC Advanced Reactor Stakeholder Meeting provide information on the ARCAP project and its relationship with the TICAP project. See ML21055A541 pp. 91-105.

Commented [A12]: It may be helpful to expand/clarify this thought. If I am interpreting correctly, the traditional compliance-oriented safety case means that there is a presumption of safety if you comply with the regulations. The shift in focus for an LMP based safety case is on ensuring that regulatory limits on radiation exposure are not exceeded and recognizes inherently that not all regulations have a nexus with safety. It should be noted, however, that compliance with the regulations using LMP is still going to be required but that the safety case is focused on complying with radiation exposure limits. This is more eloquently stated later in this document as…" i.e., how the characteristics of the plant and its operation provide reasonable assurance of adequate protection of public health and safety from a radiological consequence perspective."

Commented [A13]: The connections to regulatory compliance resulting from the implementation of this holistic and optimized approach could be strengthened if the document made a more direct connection to related language in 50.34 regarding application content. For instance:

A description and safety assessment of the site and a safety assessment of the facility. It is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products.

The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents.

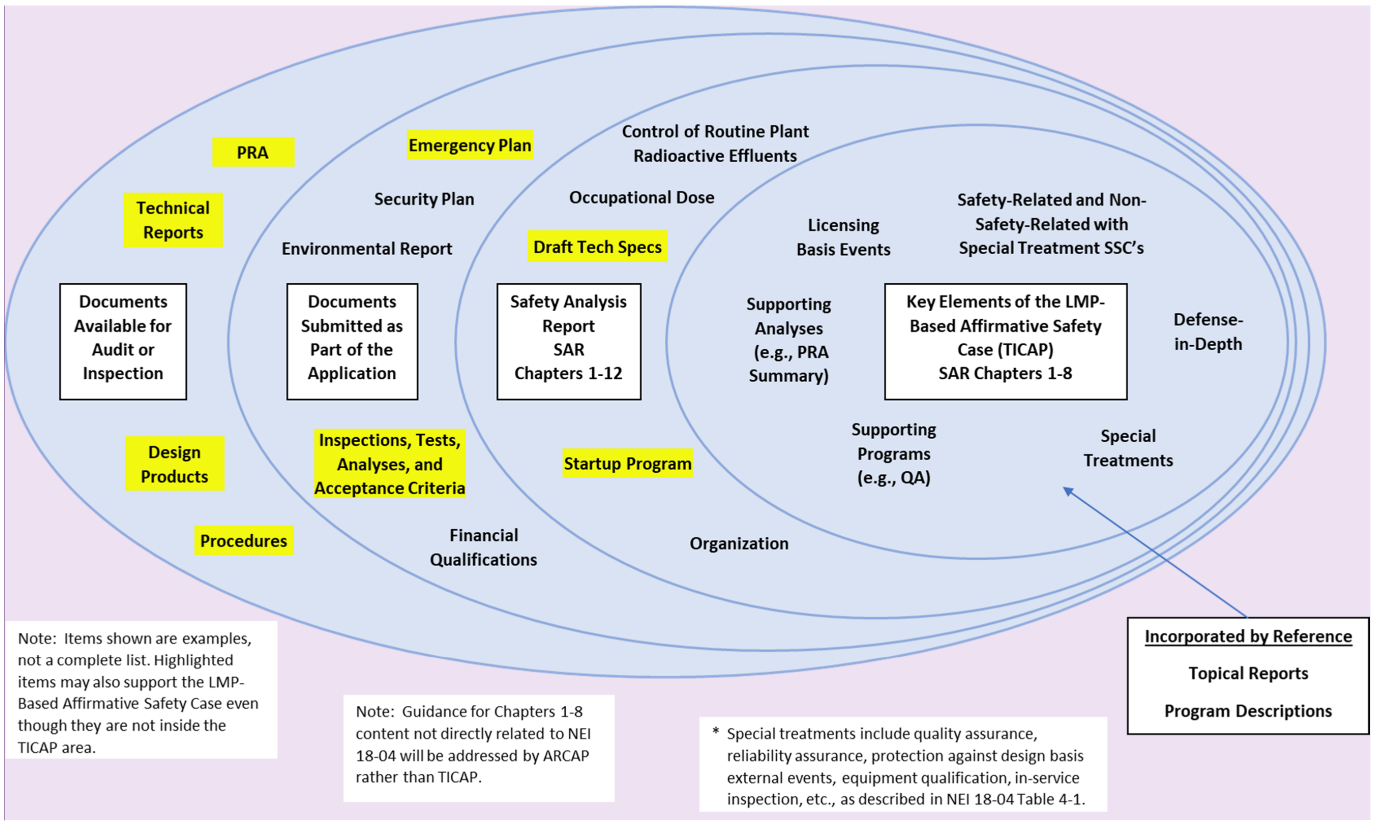

Figure 1. Relationship of TICAP to an Advanced Reactor License Application

Commented [A14]: I thought TS were supposed to be final at the COL stage… correct or not?

Commented [A15R14]: Delete 'Draft'?

Commented [A16R14]: As long as that's what was intended. If not, then we need a discussion.

Commented [A17R14]: We can clarify with Southern. There might not be an issue with the term "draft" since the applicant provides the tech specs and it is up to the NRC to review them and attach them to the license. That is, tech specs are attached to the license and the "draft" tech specs provided in the application cease to have a regulatory footprint when the license is issued

Organization of this Report

Section A of this report provides information on the purpose, background, and scope, as well as a road map for the content of this guidance document.

Section B provides information on the development of the guidance and general instructions for its use.

Section C is the chapter-by-chapter detailed guidance for the development of content at the appropriate level of detail in the sections of a SAR relating to the implementation of the NEI 18-04 methodology. The baseline guidance assumes the license applicant is requesting a COL for an advanced reactor under 10 CFR Part 52 and is not referencing a design certification.

Section C also contains adjustments to the baseline guidance if the applicant is following one of two different licensing approaches instead of the COL:

- Two-step license (CP and OL) under 10 CFR Part 50
- Design certification under 10 CFR Part 52

Commented [A18]: see previous comment on scope of applicability of TICAP guidance and making this specific to CP applicants

The adjustments are provided after the pertinent chapter or section of the baseline guidance.

Section D summarizes the results of the project.

B. DEVELOPMENT OF GUIDANCE

Overview

This document describes the necessary information provided in portions of an applicant's SAR to describe and support the LMP-based affirmative safety case for the reactor design, i.e., how the characteristics of the plant and its operation provide reasonable assurance of adequate protection of public health and safety from a radiological consequence perspective. The document presents an organization of the affirmative safety case material. It is important to recognize that this organizational approach is not the only way to present a safety case, so it should not be construed as a requirement for an advanced reactor applicant. However, for an applicant employing the LMP methodology, the following guidance provides a structure in which key technical information is provided in a clear and logical manner.

This content structure for the SAR should enable the following:

- Efficient preparation by an applicant
- Efficient review by the regulator
- Efficient maintenance by the licensee
- Ease of use by all stakeholders, including the public

The information provided in the SAR should be relevant to the design-specific affirmative safety case. The level of detail of the information should be based on the importance of the safety functions, the SSCs, and the programs to the safety case.

SAR Outline

Figure 2 provides a high-level outline of the portions of the SAR addressed by this guidance, and the following sections describe the content that applicants should provide therein. The outline is intended to present the overall safety case first and then provide the specific supporting design and operating details in subsequent chapters.

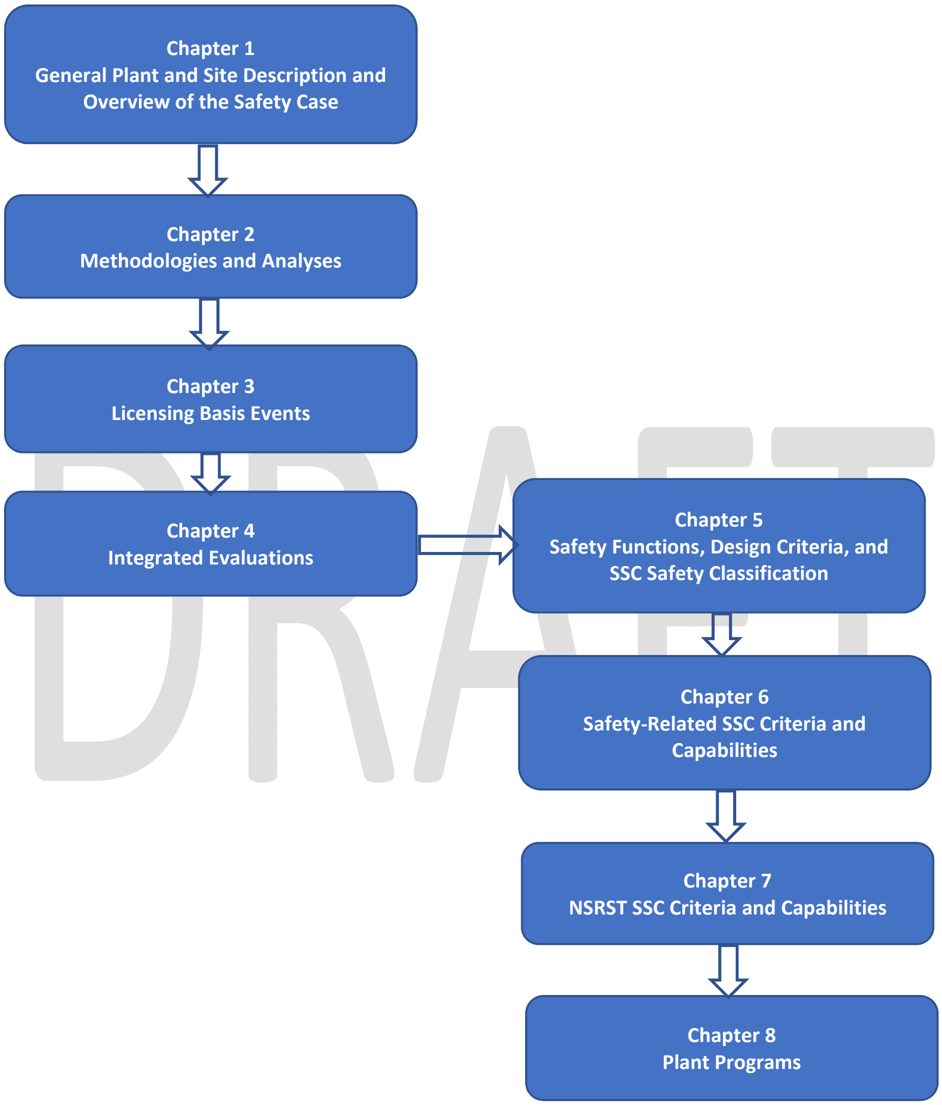

Figure 2. SAR Outline

General Instructions for Use of the Guidance

Major divisions are referred to as chapters (e.g., Chapter 2 – Methodologies and Analyses). Subdivisions of any level are referred to as sections (e.g., Section 2.1 – Probabilistic Risk Assessment).

Regular text provides instructions for the applicant under each chapter and section.

Italicized text provides additional information for context and perspective.

In addition to providing information in the SAR itself, applicants can provide material through Incorporation by Reference (IBR) or through General References (Ref).

- IBR addresses material in design-specific topical reports, application-specific program control documents, industry standards, etc. When material is IBR, the scope must be clear, i.e., if only parts of the reference are applicable to the application, the SAR should state which parts are applicable. In addition, the applicant should identify any departures from the IBR material. Applicable portions of IBR material are considered part of the application and the licensing basis.
- Ref identifies internal design or program documents or other sources of information that contain additional detail. Unlike IBR, citing material as Ref in the SAR does not make it part of the Licensing Basis except as specifically committed to by the applicant.

NEI 98-03 (Rev.1), "Guidelines for Updating Final Safety Analysis Reports," Appendix A, Nuclear Energy Institute, June 1999 (ML003779028),[1] provides additional discussion of IBR and Ref documents.

Alternative Licensing Paths

NRC regulations provide applicants with options for obtaining an Operating License (OL) for a nuclear power reactor. The baseline guidance presented in this section of the document assumes an applicant is applying for a Combined License (COL) under 10 CFR Part 52, Licenses, Certifications, and Approvals for Nuclear Power Plants. The guidance further assumes that the applicant is not referencing an existing Design Certification (DC). In this scenario, the applicant would need to provide the maximum amount of information compared to other approaches that allow for more incremental provision of equivalent information.

Advanced reactor applicants may choose alternative licensing pathways. This section also provides guidance for two alternative pathways deemed to be reasonably likely. Those pathways are:

Commented [A19]: This is using the term topical reports differently than we've used it for Part 52. I believe they are describing "Technical Reports" which are design specific. Topical Reports do not have to be design specific (e.g. a systems code methodology topical report) but they *do* have stand alone staff review and approval as opposed to technical reports.

Commented [A20]: From NEI 98-03.

Commented [A21]: See previous comment about scope of guidance document

Commented [A22]: please clarify that this refers to COL w/o reference to DCD and consider that this scope is same as OL (see above comment)

[1] NEI 98-03 Rev. 1 was endorsed by Regulatory Guide 1.181, Content of the Updated Final Safety Analysis Report in Accordance With 10 CFR 50.71(e), Nuclear Regulatory Commission, September 1999 (ML992930009).

- Two-step licensing—The applicant first applies for and obtains a Construction Permit (CP) under 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities, and subsequently applies for and obtains an Operating License (OL) also under 10 CFR Part 50.
- Design certification (DC)—The applicant is a reactor vendor that applies for a standard DC under 10 CFR Part 52, Subpart B. It does not contain site-specific information. A future applicant would reference the DC along with site-specific and owner-specific information as part of a COL application.

Note that this guidance is not comprehensive—there are other potential licensing pathways involving Part 50 and Part 52 besides the ones addressed in this guidance document. At the time of this writing, these two alternative approaches plus the COL baseline approach were considered the most likely to be employed. Each is discussed in summary fashion below. With respect to the guidance, where there are adjustments to the baseline COL guidance, those adjustments are provided at the end of the applicable section or chapter in Section C.

Two-Step Licensing

With this alternative, the applicant obtains a construction permit (CP), constructs the plant, and obtains an operating license. Issuing a CP does not constitute approval to operate a facility and it does not provide finality for the preliminary design unless specifically requested by the CP applicant. Accordingly, the NRC expects the preliminary design information in a CP application to be supplemented, ~~and~~ updated, and finalized in the OL application. CP applicants provide less information initially than in the OL application (e.g., operational programs are not described in a CP application). The SAR submitted as part of a CP application is referred to as a Preliminary Safety Analysis Report or PSAR.

The application content for all licensing paths will be impacted by the overall licensing strategy. This impact is particularly pronounced for the CP licensing path because the degree of information which is needed in an application is highly dependent on the finality of the decision requested from the NRC at the CP stage. Therefore, to optimize the applicability of the CP guidance provided in this document, it is assumed that the applicant will seek the minimum possible level of decision finality when applying for the CP.

The scope and level of detail of an FSAR for a COL submitted under Part 52 (the baseline process) ~~are~~ is expected to be commensurate with the FSAR for an OL application submitted under Part 50 (i.e., the second step of two-step licensing). One major difference between the approach taken in this guidance document and the large LWR Part 50 and 52 processes is that the approach used herein, building on the LMP based safety case, shifts from the compliance-based set of licensing requirements to a performance-based affirmative safety case. It should be noted, however, that compliance with the regulations using LMP is still going to be required and exemptions may be required. The difference is that instead of there being a presumption of safety based on compliance with the regulations the LMP affirmative safety case is based how the characteristics of the plant and its operation provide reasonable assurance of adequate protection of public health and safety from a radiological consequence perspective. The integral

Commented [A23]: NRC would seek complete guidance for the range of licensing options to implement LMP for part 50 and 52. Would a near-term or long-term revision to this document "fill in the holes" on addressing MLs?

Commented [A24]: This statement is the basis for making this guidance document applicable to OL applicants as well despite the LMP based safety case shifting away from compliance based approach. Compliance with regulations will still be required, or exemptions sought, if using Part 50 or Part 52 and there should be a reference to the applicability of regulations document (see also comment above about compliance-based approach). LMP approach focuses on functional safety requirements but the legal requirements of compliance still need to be met.

Commented [A25]: Another major difference that should be noted is that there is a requirement for COL and DC applicants to provide ITAAC whereas there is not such requirement in Part 50 for OL applicants.

Commented [A26]: I didn't think that LMP is used (alone) to show compliance with regulations. Or at the very least there are a subset of regulations that LMP does not address, but still require compliance.

addition of risk insights to the LMP based approach creates a synergy that can dramatically reduce the level of content and enable a safety-focused review instead of a compliance review.

The chapter and section designations for the PSAR are the same as those used in Section C of this report. Adjustments to the guidance for a CP application ~~two-step licensing~~ are provided at the end of the applicable section or chapter.

Design Certification

With this alternative, the applicant submits a SAR as part of a 10 CFR Part 52 Subpart B DC application.

There is significant similarity between FSAR requirements for a COL application and FSAR requirements for a DC application. The exception is that a DC does not address a specific site. Thus, the content of application discussions in the COL guidance are directly applicable to a DC application with the exception of any site-specific information. Note that a DC will require the incorporation of a plant parameter envelope to permit the evaluation of the proposed design certification to meet established regulatory criteria. Such plant parameter envelope information may be placed in Chapter 2.

Commented [A27]: DC's are specifically focused on designers and, as such, the designers were not expected to include information on operational programs as this was responsibility of the COL applicant.

In accordance with the regulations for a DC, site parameters are postulated for the design. This is typically accomplished using a plant-parameter envelope which would specify appropriately bounding parameters for a site that might be chosen by an applicant. However, in implementing the PRA standard ASME/ANS RA-S-1.4-2021 referenced in the COL guidance, the concept of a "bounding site" is introduced for external hazard assessments. The bounding site is a hypothetical site defined as having a set of site characteristics that will be used to establish values for evaluation of the performance capabilities of the design. The site characteristics may be selected from site parameters from actual sites and may reflect hazards from different sites for different external hazards. For the bounding site, site-related parameters are defined using a set of external hazard conditions that are chosen to provide appropriately high external hazard design parameter values and the most adverse metrological conditions and population data for assessing off-site radiological impact. These considerations are reflected in the selection of the DBEHLs for the standard plant design. It should be noted that the bounding site is consistent with the plant parameter envelope used in the DC application.

Commented [A28]: For a COL referencing a DC, actual site characteristics should be confirmed to be within the plant parameter envelope. In addition, actual site characteristic may be used to inform and potentially modify the design in the DC, adjust as necessary any LBEs, safety classifications of SSCs, and DiD adequacy assessments.

The chapter and section designations for the DC SAR are the same as those used in Section C of this report. Adjustments to the guidance for a DC are provided at the end of the applicable section or chapter.

C. SAR CONTENT GUIDANCE

Section C of this report contains detailed guidance for the entire TICAP portion of the SAR, Chapters 1 through 8. A table of contents and a list of tables for the SAR Content Guidance are provided below.

Table of Contents for the SAR Content Guidance

List of Tables for the SAR Content Guidance

1 GENERAL PLANT AND SITE DESCRIPTION AND OVERVIEW OF THE SAFETY CASE

The applicant should provide general descriptive information about the plant and the site to provide context for the NRC safety review. In addition, the descriptive information should be useful for stakeholders such as nearby residents who want to understand how the plant and its operations will impact the surrounding communities.

The descriptive information is divided into four sections as follows:

- Section 1.1 provides descriptive information about the reactor and supporting systems that provides a framework and context for the information in subsequent chapters.
- Section 1.2 provides descriptive information about the site that provides a framework and context for the information in subsequent chapters.
- Section 1.3 provides a high-level overview of the LMP-based affirmative safety case in terms that can be understood by non-subject matter experts. This section is intended to summarize and provide context to the information in Chapters 3 through 7.
- Section 1.4 provides a listing of reference information and a “road map” to the Chapters in which the information is explicitly used to support the licensing basis.

While important for providing an understanding of the plant and the safety case, to establish review context, Sections 1.1 through 1.4 are not intended to be bases for regulatory conclusions by the NRC and are therefore outside the licensing basis that is provided in Chapters 2 through 8.

Applicants who use this guidance are expected to employ a variety of technologies differing in numerous aspects, including size, physical characteristics, materials, reactor power level, fuel type, coolant type, and power conversion system. Rather than prescribe a specific organization for the information, this guidance specifies elements that should be included in an adequate description.

With respect to the level of detail, the information should accomplish the following goals:

1. Provide a stakeholder who is not an expert on nuclear technology with sufficient information to understand the purposes of the facility and the general means by which each purpose is accomplished
2. Provide a high-level summary of specific site information with a focus on the information that is relevant to the LMP-based affirmative safety case. Detailed site information supporting the development of design basis external hazards should be provided in Chapter 2.
3. Provide a summary of the LMP-based affirmative safety case that is demonstrated in Chapters 2 through 8 in a concise and understandable manner for all reviewers and stakeholders

Commented [A29]: A regulatory gap analysis (i.e., regulation applicability) summary that include exemptions requested may also be useful.

Commented [A30R29]: I believe this is outside the scope of industry’s TICAP guidance document – we may need to pick it up in the draft TICAP RG

Commented [A31]: This should be more specific in terms of perhaps radiation exposure, land use, water use, EPZs, etc to distinguish from economic impact.

Commented [A32R31]: I agree that the proposed language sounds like it is going down the path of an environmental assessment. For the Part 50 review the focus (in my opinion) is on the radiological hazard. Can it be reworded to state: “nearby residents who want to understand the key features of the plant that protect the

Commented [A33]: It may be helpful to provide a reference to the definition of “licensing basis” or “current licensing basis” in 10 CFR 54.3. Question for discussion is whether or not the definition needs to modified for the purposes of this guidance document or other advanced reactor guidance documents?

Commented [A34]: Should there perhaps be a statement made regarding the significance of this info being outside the licensing basis (i.e., that the information in these sections are not subject to the change control process in 50.59 or whichever change control process applies). Also, perhaps some discussion about FSAR updating requirements. I would expect that the primary focus of FSAR updates would be on the other chapters to ensure the licensing basis is maintained current but that updates to Chapter 1 should also be considered when updating the other chapters.

1.1 Plant Description

The intent of this section is to describe at an overview level the plant and the plant systems. The focus of this section is on those systems that are relevant to the LMP-based affirmative safety case; however, this section is expected to be a reasonably complete plant description that should enable the reader to understand the fundamental concepts of the plant and how it operates. Elements of the plant description are listed below.

Commented [A35]: Section 1.1 should also describe the plant operating parameters (power level, temperatures, flow rates, etc.). I didn't see these mentioned.

1.1.1 Reactor Supplier and Model

Describe the reactor supplier and the model of the reactor. This should be a very brief description that allows the reviewer and stakeholder to identify and obtain background information (pre-application engagement information, publicly available vendor information, etc.) on the vendor and the design.

1.1.2 Intended Use of the Reactor

Describe the intended purpose of the reactor and the end uses. This could include descriptions of electricity generation, heat generation and use, industrial facilities served, micro-grids served, etc. This description is intended to inform reviewers and stakeholders but is not intended to provide justification for licensing the reactor.

1.1.3 Overall Configuration

This section provides information on the overall layout of the plant and summarizes any features or the plant layout that are significant from the perspective of the LMP-based affirmative safety case. This is intended to be a summary discussion accompanied by illustrative drawings, site plans, etc., that support the discussion. A system-level plant block diagram should be considered for this section, possibly color-coded to identify systems that are relevant to fundamental safety functions. The block diagram should align with the construct of Sections 1.1.4.1 through 1.1.4.4.

If the plant includes more than one reactor module, the relationship of the reactors should be described.

1.1.4 Description of Plant Structures, Systems, and Components

This section provides an overview of the plant Structures, Systems, and Components (SSCs). Given that this guidance is technology inclusive, the systems will vary among designs and technologies. The balance of this section provides examples of the information that should be provided. A detailed description of the plant SSCs is not expected in this section but rather a brief description of the SSCs such that the discussion of FSFs in Section 1.3.2 can be put into context with the overall plant design and SSCs. This section should reference the location of more detailed SSC-specific information to the extent it is provided elsewhere in the SAR (i.e., subsections of Chapters 6 or 7).

This section should provide a high-level summary description including figures and diagrams when the text description is not sufficient for general understanding. The NEI 18-04 methodology categories SSCs as Safety-Related (SR), Non-Safety-Related with Special Treatment (NSRST), and No Special Treatment (NST). SR and NSRST SSCs will be addressed in greater detail in Chapters 6 and 7, respectively.

Note: Light water reactor SARs contain detailed descriptions of some SSCs (e.g., reactivity control and control rod drive mechanisms, fuel, emergency cooling, etc.) in LWR SAR Chapters 4 and 5. Chapter 1 should not be the primary source of detailed SSC information. Care must be taken to limit this section consistent with the objective of minimizing any redundancy with subsequent sections.

Note: Sections 1.1.4.1 through 1.1.4.4 provide examples of how the information could be organized, recognizing that different designs and technologies will likely be organized differently, based on the systems in the design and relative importance of the SSCs. This section is organized in a traditional systems-centric manner in order to facilitate an overall understanding of the plant by a broad group of stakeholders. It could be organized differently for a given technology as long as the overall plant functionality can be understood.

1.1.4.1 Reactor Systems and Components

1. Nuclear design (e.g., spectrum, reactor control, multi-module reactor control)
2. Fuel
3. Reactor cooling
4. Reactivity control

1.1.4.2 Secondary Systems and Components

1. Heat transfer and cooling system
2. Power conversion system
3. Power transmission (e.g., switchyard)

1.1.4.3 Significant Support Systems and Components

1. Fuel handling
2. Fuel management
3. Control room
4. Electrical power
5. Radioactive waste

Commented [A36]: Add spent fuel storage.

1.1.4.4 Major Structures

1. Reactor building
2. Auxiliary, secondary, and support buildings
3. Cooling towers/systems
4. Co-located facilities (e.g., cogeneration, fuel processing, and buildings)

Two-Step Licensing

Section 1.1 would provide the general description of the plant and plant systems. The discussion of plant systems would be preliminary but sufficient to permit the reader to understand fundamental concepts of the plant and how it operates. The descriptions of the overall configuration in Section 1.1.3 also would be preliminary but with sufficient detail to support reader understanding of the design and how the LMP-based affirmative safety case will be developed. Discussion of systems and components in Sections 1.1.4.1 through 1.1.4.4 will be preliminary but sufficiently clear for the reader to understand the initial plant functionality.

1.2 Site Description

This section provides a high-level overview of the site and the general vicinity of the licensed activities. Specific site attributes directly relevant to the affirmative safety case are included in Chapter 2 and are only briefly summarized in this section. This section should include a site layout and maps of the general vicinity showing the site exclusion area, low population zone boundaries, nearby industrial facilities, and population centers sufficient to provide the reader an overview understanding of the plant and the site. Discussion of site features (flood plains, access roads, etc.) can be included here to the extent that it facilitates an overall understanding of the safety case overview in Section 1.3.

Design Certification

This section is not applicable because a design certification is not associated with a specific site. *Note that information on the assumed plant parameter envelope should be provided in Section 6.1.1 Design Basis External Hazard Levels.*

1.3 Safety Case

This section provides a high-level overview of the safety case methodology and the outcome of executing the methodology. It focuses on the fundamental safety functions and how they are accomplished by the plant design described in Section 1.1.

Commented [A37]: see previous comment on making this specific to CP applicants...typical throughout

Commented [A38]: This section should explain what content differences there are for a DC application.

Commented [A39]: Clarify to say preliminary only for CP application. OL application content should be final and complete. Similar comment for all uses of "preliminary" in this paragraph and similar paragraphs throughout the document.

Commented [A40R39]: I agree. It should be specific to CP.

Commented [A41]: Add another sentence: "The R & D planned to support the technical basis for the design should be described."

Commented [A42]: Need to add site info to Chapter 2 or reference Chapter 2 ISG here or in the TICAP RG.

Commented [A43]: Are site characteristics such as meteorology (and others if needed, such as hydrology) and the atmospheric dispersion factors both for accidents and for normal releases/effluents and their method for determination included in section 1.2 or 1.3 or elsewhere in the FSAR (would the information be in Section 2.4 or Section 3 description of analyses)?

Assumptions on atmospheric transport and meteorological data are necessary to calculate LBE and DBA consequences for comparison to the F-C target using a code such as MACCS.

Commented [A44R43]: Issue discussed with commenter. Noted that Chapter 1 info is not part of licensing basis. Commenter was going to see if they have has the same issues in other Chapters.

Commented [A45]: Where are the "assumed" site parameter atmospheric dispersion factors and the basis for their values located for the DC? Both for the accidents (short-term dispersion) and for routine releases/effluents (long-term dispersion and diffusion). These are not external hazard, per se (since they don't affect the plant). But they are site characteristics that are needed to evaluate the radiological consequences offsite for the DC.

Commented [A46R45]: See above

Commented [A47R45]: Additional comment noted that for DCs this section could at least indicate that site parameters for a DC application are typically assumed and established to be bounding parameters.

1.3.1 Safety Case Methodology

This section should refer to NEI 18-04 and RG 1.233. If the applicant conforms to the guidance in its entirety, then a brief statement of conformance, as demonstrated in Chapters 2 through 8, is adequate. An example statement is provided below.

> The selection of Licensing Basis Events (LBEs); safety classification of structures, systems, and components (SSCs) and associated risk-informed special treatments; and determination of defense-in-depth (DID) adequacy were done in accordance with the methodology of Nuclear Energy Institute report NEI 18-04, Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development, Report Revision 1 (August 2019), as endorsed by Nuclear Regulatory Commission Regulatory Guide 1.233, 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 Revision 0 (June 2020). This is demonstrated in Chapters 2 through 8. There were no ~~departures~~deviations from the endorsed methodology.

Commented [A48]: please use "deviation" throughout document and avoid use of departure as it has a specific definition under Part 52

If the applicant deviated from the methodology, either by not using parts of it or by using an alternate method, a brief statement to that effect should be included here, and a discussion and justification should be provided in the relevant chapters of the SAR to support and clarify the licensing basis.

1.3.2 Fundamental Safety Functions

The section should begin by establishing the overall performance objectives—the regulatory dose ~~limits~~ criteria and quantitative health objectives (ref. NEI 18-04 Figure 3-1). The discussion should go through each FSF and summarize how it is satisfied. This section is not intended to be complete and exhaustive but is a high-level summary for general consumption. A systematic and thorough discussion of the safety case is provided in subsequent chapters. LBEs and event sequences relevant to each FSF are discussed in detail in Chapter 3, integrated evaluations and overall risk are discussed in Chapter 4, design criteria are discussed in detail in Chapters 5 through 7, and plant programs supporting reliability and availability are discussed in Chapter 8.

Commented [A49]: Part 20 includes dose limits, 50.34 and 52.79 include dose criteria. The use in the F-C target is more like criteria. Recommend changing the word here to "criteria."

Note: The concept of FSFs goes back to International Atomic Energy Agency TECDOC-1570[1]. NEI 18-04 Section 3.3.4 discusses the FSFs as used in the LMP methodology. RG 1.233 presents the FSFs in a slightly different form from NEI 18-04, but the differences are more stylistic than substantive. The applicant may choose to express its FSFs consistent with either NEI 18-04 or RG 1.233. If the applicant adds one or more FSFs or modifies its FSFs substantially from those

[1] International Atomic Energy Agency, "Proposal for a Technology-Neutral Safety Approach for New Reactor Designs," Technical Report IAEA-TECDOC-1570, 2007.

documented in NEI 18-04 or RG 1.233, the applicant should discuss the basis for the selection of its FSFs. The form of the FSFs provided below is taken from NEI 18-04.

1.3.2.1 Retaining Radionuclides

This section should provide an overview of how the plant design accomplishes the FSF. This should include a high-level discussion of location and types of radiological inventory and the various SSCs that are available to prevent or mitigate releases through various modes of operation, including response to off-normal events or accidents.

1.3.2.2 Controlling Heat Generation

Commented [A50]: might it be useful to state that this is a surrogate for reactivity control?

This section should provide an overview of how the plant design accomplishes the FSF. This should include a high-level discussion of SSCs that are utilized to control heat generation through various modes of operation, including response to off-normal events or accidents.

1.3.2.3 Controlling Heat Removal

This section should provide an overview of how the plant design accomplishes the FSF. This should include a high-level discussion of passive and active heat removal SSCs and their roles through various modes of operation, including response to off-normal events or accidents.

1.3.3 Defense-in-Depth

This section should provide an overview of the DID aspects of the design. In this overview, the applicant should cite key examples of design features and programmatic elements included in the DID baseline described in detail in Chapter 4. DID is a key element of the LMP-based affirmative safety case and the demonstration of reasonable assurance of adequate protection of public health and safety.

1.4 Summary of Reference or Source Materials

This section lists information that is incorporated by reference (IBR) in Chapters 2-8 to support the LMP-based affirmative safety case or referenced (Ref) to identify additional information. Information in this section is intended solely to assist in reviewer and user efficiency by "road-mapping" the reference information to the chapter in which the licensing basis resides. Specific applicability and commitments to the use of the information listed below must be identified in Chapters 2 through 8 as part of the licensing basis.

Commented [A51]: This list of IBRs, as well as other lists or RGs, SRPs, codes and standards is a convenient list to include within the licensing basis information sections and would be expected to be maintained coincident with required FSAR updates. I would recommend that some version of this list also included as licensing basis information.

See "General Instructions for Use of the Guidance" (in Section B of this document) for additional discussion of IBR and Ref.

1.4.1 Reference Designs, Licenses, or Certifications

Table 1-1. Example Table of Design, License, and Certification References

Reference	Use
Identify the explicit reference	Describe briefly how the reference is used and why it is relevant

1.4.2 Topical Reports

Table 1-2. Example Table of Topical Report References

Reference	Use	SAR Chapter/Section
Identify the explicit reference	IBR vs. Ref	Identify SAR chapter/section where the information is cited and used

1.4.3 Other Technical Reports (e.g., Environmental Report, Submitted Technical Reports, and Test Data Reports)

Table 1-3. Example Table of Other Technical Report References

Reference	Use	SAR Chapter/Section
Identify the explicit reference	IBR vs. Ref	Identify SAR chapter/section where the information is cited and used

1.4.4 Industry Codes, Standards, Guidance (e.g., ASME, ANS, ACI, and NEI)

Table 1-4. Example Table of Industry Codes, Standards, and Guidance References

Reference	Use	SAR Chapter/section
Identify the explicit reference	IBR vs. Ref	Identify SAR chapter/section where the information is cited and used. Any exceptions or conditions of use should be described in that SAR chapter/section.

2 METHODOLOGIES AND ANALYSES

Certain analyses and analytical tools (methodologies) are used in the identification of licensing basis events, the evaluation of the consequences of such events, or assessing the performance of safety-related and NSRST SSCs. This chapter of the SAR presents information on some of those analyses and analytical tools. It is intended primarily for cross-cutting information or evaluations that support multiple LBEs or SSCs. Providing that information or evaluation upfront in one place is intended to make the documentation that follows in subsequent chapters more efficient and concise.

The amount of information provided in this chapter will depend in large part on the degree of pre-licensing engagement activities and associated NRC technical reviews and approvals. If limited pre-licensing engagement activities occurred, the level of detail provided in this chapter would need to be of sufficient detail that NRC can perform its independent review. If pre-application submittals (e.g., topical reports) were made and NRC approvals were obtained for some or all of these topics, then the detailed review of these topics will have been documented through a separate licensing process. If the review is occurring through separate documents, then only a high-level summary of the topic is required with appropriate references to the separate licensing documents. Applicants are encouraged to make maximum use of the topical report process.

Commented [A52]: Not sure if this discussion refers to Topical Reports or something else (e.g., methodologies endorsed by NRC in RGs?). I would expect the appropriate references also include NRC safety evaluation reports. Perhaps this can be clarified.

The number and scope of these methodologies and analyses will vary depending on the technology and the safety case. Several of the methodologies and analyses are expected to be common to all applications and are set forth below. Others may be included in this chapter, depending on the specific details of the application.

2.1 Probabilistic Risk Assessment

The Probabilistic Risk Assessment (PRA) is the plant model that provides an integrated assessment of risk to the public from the nuclear power plant. A technically sound PRA is essential for implementing the NEI 18-04 methodology. The purpose of this section is to summarize elements of the PRA that are essential to the NEI 18-04 affirmative safety case without duplicating PRA products described in other chapters. The PRA information included in the SAR should be at a summary level only as described below. It should address the requirement in 10 CFR Part 52 that the SAR includes a description of the design-specific PRA and its results. It is included near the beginning of the SAR because of the PRA's prominent role in exercising the NEI 18-04 methodology.

Commented [A53]: Define "technically sound"? This term is not used in NEI 18-04, RG 1.233, PRA standard, or SRP 19.0. Verify whether the interpretation in SRP 19.1 is taken into consideration?

Commented [A54R53]: Suggest change to "technically acceptable" consistent with RG 1.200 and ASME/ANS-RA-S-1.4-2021

Key products of the PRA are reflected in other parts of the application. Chapter 3 presents the LBEs supported by the PRA (Anticipated Operational Occurrences [AOOs], Design Basis Events [DBEs], and Beyond Design Basis Events [BDBEs] and includes the LBE descriptions, frequencies and uncertainties, consequences and uncertainties, and evaluation of risk significance against the LMP Frequency-Consequence Target. Chapter 4 shows the integrated risks across all the LBEs and compares them to NEI 18-04 cumulative risk metrics.

Uncertainties in the PRA results are considered as part of the DID evaluation described in Chapter 4. The purpose of this section is to summarize results and insights from the PRA that are essential to the NEI 18-04 affirmative safety case without duplicating PRA products described in other chapters. The applicant maintains complete PRA documentation in its plant records and are available for audit by the NRC staff.

2.1.1 Overview of PRA

This section summarizes the scope, methodology, and pedigree of the PRA. The pedigree is intended to be (i) a statement of conformance (with any ~~departures~~deviations) with the advanced non-LWR PRA standard,[1] ASME/ANS RA-S-1.4-2021, the manner in which the standard was applied, and PRA peer review findings, or (ii) an alternative means of demonstrating PRA technical adequacy that may be proposed by the applicant.

The discussion should include the following items:

- A statement that the PRA conforms to the non-LWR PRA Standard ASME/ANS-RA-S-1.4-2021
- A statement that a peer review was completed following the guidance in NEI 20-09, Rev. 0, "Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR PRA Standard"[2]
- Discussion of how the NRC regulatory guide that endorses the PRA standard was implemented
- Identification of the sources of radionuclides addressed and the sources of radionuclides that were screened out
- Discussion of how multi-reactor scenarios were addressed, if applicable
- Identification of the internal and external hazards that were included and the ones that were screened out
- Identification of the plant operating states that were included and those that were screened out
- Identification of the software and analytical tools that were used to perform the event sequence modeling and quantification, determine the mechanistic source terms, and perform radiological consequence evaluations (with appropriate references to technical and/or topical reports provided as applicable)

Note: If the applicant chooses to use a PRA that is not compliant with the non-LWR PRA standard, the requirements for PRA documentation, either in the SAR or other documentation, are expected to be significantly greater than the guidance provided herein. This guidance document does not address SAR content for a non-compliant PRA.

[1] ANSI/ASME/ANS RA-S-1.4-2021, "Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants," American Society of Mechanical Engineers and American Nuclear Society, approved January 28, 2021

[2] NEI 20-09 "Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR PRA Standard," Rev. 0, Nuclear Energy Institute, August 2020. (https://adamswebsearch2.nrc.gov/webSearch2/main.jsp?AccessionNumber=ML20339A485).

Commented [A55]: Modify this paragraph to include 1) PRA safety functions and PRA success criteria in Chapter 5, 2) PRA EE/DBEHL in Chapter 6, and 3) the uses of PRA for RAP and Maintenance Rule in Chapter 8.

Commented [A56]: Change all "PRA technical adequacy" to "PRA acceptability." Otherwise, insert "scope of the PRA," "Level
of detail," "plant representation," and "PRA configuration control."

Commented [A57]: Key assumptions made in the PRA can be important in understanding the PRA and its limitations. The identified key assumptions should be confirmed as the detailed design is implemented and the PRA is updated/upgraded. Thus, we suggest the key assumptions to be listed below.

Commented [A58]: Include a statement that the PRA reasonably reflects the as-designed, as-built, and as-operated plant.

Commented [A59R58]: Question on this as to whether an OL or COL applicant can say "as-operated" yet? Maybe the right terminology is "intended to be operated"? Perhaps a clarification on this would help.

Commented [A60R58]: You are right. The term should be "as-to-be-built and as-to-be-operated." I used "as-built and as-operated" for consistency with ASME/ANS RA-S-1.4-2021, which is cited in the document. They know what to say.

Commented [A61]: The RG that endorses NEI 20-09 should be added here for completeness.

Commented [A62]: A good starting point for developing AIA guidance.

Commented [A63]: perhaps a clarification on what it means to be "screened out" would help here and in other places it is used (e.g., some may be based on radionuclide inventory, others may be based on frequency of occurrence, etc)

Commented [A64]: How are deterministically selected external events factored into the PRA or the uncertainty evaluation?

Commented [A65]: Change to "and justification for the ones that were screened out"

Commented [A66]: Change to "and justification for those that were screened out"

Commented [A67]: please clarify if "quantification" addresses both radionuclide source inventory and frequency of occurrence of events

Commented [A68]: Uncertainty analysis should be discussed here.

Two-Step Licensing

At the construction permit stage, neither the plant design nor the PRA is expected to have the level of maturity that will be necessary to support an operating license application. At the CP application stage, the applicant should describe its ultimate intended approach for qualifying the PRA. If conformance to ASME/ANS RA-S-1.4-2021 is planned, a simple statement to that effect should be sufficient. If the applicant intends to use another PRA methodology, that planned approach for establishing PRA technical adequacy should be described. In either case, the applicant should address the last five items in the Section 2.1.1 list, consistent with the state of the plant design and the PRA at the time of CP application. To be clear, no PRA peer review should be required at the construction permit application stage.

Design Certification

Section 2.1.1 should describe adjustments made to the PRA and uncertainty assessments to address the bounding site characterizations and SSC fragilities based on the DBEHLs described in Section 6.1.1. The degree to which the use of the bounding site characterizations could affect analyses performed in other chapters and sections would be addressed in the descriptions of those analyses and results (determination of LBE's, SSC classification, etc.).

2.1.2 Summary of Key PRA Results

Because NEI 18-04 is a risk-informed methodology, key PRA results are incorporated in the descriptions of the outputs of the methodology provided in the SAR. Those results are not repeated here, but this section provides pointers to those PRA results.

The applicant should provide a statement such as the following, identifying those parts of the SAR that include key PRA results:

> Key PRA results are provided in subsequent chapters of the SAR.

- Chapter 3 presents LBEs that are supported by event sequences in the PRA. It includes a plot of the frequencies, consequences, and uncertainties of these LBEs with a comparison against the Frequency-Consequence Target in NEI 18-04 Figure 3-1.
- Chapter 4 presents the integrated risks across all of the LBEs and compares them to the NEI 18-04 cumulative risk metrics. It also describes the DID evaluation, which is informed by uncertainties in the PRA results.

Two-Step Licensing

With respect to results, the COL guidance is applicable, with the understanding that the Chapter 3 and Chapter 4 results will be preliminary relative to those to be presented in support of an OL application.

Design Cetification section?

Commented [A69]: CP PRA is an area of ongoing internal discussions along with external interactions on CP guidance development, Part 50/52 rulemaking etc. This TICAP guidance may need to be updated based on eventual NRC position.

Commented [A70]: Change to "PRA acceptability." See previous comment.

Commented [A71]: Inconsistent with the staff's position on PRA peer review. Although PRA peer review plays an important role in demonstrating the PRA acceptability, however, it is NOT REQUIRED (per regulation) at any stages of the licensing process. The staff highly recommends that applicants to perform PRA peer review to help reduce the need for an in-depth review by NRC reviewers. Clarify the requirement for a PRA peer review for all licensing applications.

Commented [A72]: This section should also provide Identification and assumptions regarding plant-specific SSCs

Commented [A73R72]: It's also not clear as to what adjustments are contemplated since the DC assumes site parameters. The only adjustments that I can think are adjustments to those assumed parameters after a specific site is chosen but that might only occur for a COL referencing a DC which is not in scope for this guidance. Please clarify.

Commented [A74R72]: During DC stage, in addition to the site parameters, PRA may not explicitly model all aspects of the design. e.g., balance of plant. Some SSCs are conceptual designs, e.g., component cooling water, essential service water, switchyard, grid configuration, etc. The assumptions for these SSCs should be clarified.

Commented [A75]: The PRA results should include risk-significant SSCs and human actions , which plays a significant role in SSC classification and special treatments. We suggest considering pointers to additional chapters where risk-significant SSCs (e.g., Chapter 5) are discussed.

Commented [A76]: SRP Chapter 19.0 defines PRA results as the PRA quantitative and qualitative results, including CDF, LRF, the identification of key PRA assumptions, the identification of PRA-based insights, and discussion of the results and insights from importance, sensitivity, and uncertainty analyses. It is unclear what "key PRA results" would refer to?

Commented [A77]: It is expected that an OL application will have the same level of design finality as a COL, therefore I believe this statement should be referring to a CP applicant having only preliminary results.

2.2 Source Term

Source term refers to the type and~~,~~ quantity~~, and timing of the release~~ of radioactive material analyzed for potential release from a facility, including timing, during a postulated event. The source term varies with the reactor, plant design, operating characteristics, and the nature of the event. A designer may elect to use a conservative, enveloping source term or a mechanistic source term that is based on a more realistic evaluation of reactor operation and event progression. To the extent that source term information is generic to some or all the events considered for the reactor, that information may be provided in this section rather than with each event.

The applicant should quantify the radionuclide inventory prior to the beginning of the event sequence. For light water reactors with a defined solid core region, this is typically done with computer codes such as the SCALE package. The applicant should describe and justify the key inputs used, such as the quantity of fissile material, core operating history, and core operating characteristics. The applicant should justify the applicability of the analytical methodology to the characteristics of the reactor, including a discussion of the underlying experimental or analytical basis. The applicant should assess the uncertainty associated with the calculation and make appropriate allowances for it. For sources other than a defined solid core region, the applicant should describe the basis for the quantity and activity of the material present.

The applicant should address the transport of the radioactive material from its point of origin to the accessible environment. For light water reactors, this is typically done with computer codes such as LOCADOSE for design basis events or MAAP for beyond design basis events. The applicant should provide the key inputs and assumptions used and justify the applicability of the analytical methodology to the characteristics of the plant, including a discussion of the underlying experimental or analytical basis. The applicant should describe the available pathways for transport. The applicant should assess the uncertainty associated with the calculation and make appropriate allowances for it. Mechanistic source terms employed in the PRA are subject to the technical requirements in the non-LWR PRA Standard ASME/ANS RA-S-1.4-2021. Guidance for source term development can be found in "Risk-Informed, Performance-Based, Technology-Inclusive Regulatory Infrastructure; Technology-Inclusive Determination of Mechanistic Source Terms for Offsite Dose-Related Assessments for Advanced Nuclear Reactor Facilities," INL/EXT-20-68717, June 2020.

Two-Step Licensing

Section 2.2 will mirror the discussions above but will reflect the preliminary nature of the design information as appropriate.

Design Certification section?

Commented [A78]: Good starting point for AIA guidance.

Commented [A79]: What about radiological source terms for routine effluents, radwaste system design, shielding design and equipment qualification?

Commented [A80]: Need to summarize the fuel qualification program or reference a Topical Report.

Commented [A81]: Inconsistent with the PRA acceptability. The use of bounding source term is only justified as PRA Capability Category I. For LMP application, in most cases, PRA CC II is preferred.

Commented [A82]: Suggest adding a statement here like at the end of section 2.4: "*Details of the analyses should be in design records and available for regulatory audits.*"

Commented [A83R82]: Seems like a good suggestion for clarity

Commented [A84]: Identify and quantify available radionuclide inventories – there is more than one source of radionuclides that are subject to release (e.g., core, coolant, tanks, spent fuel, etc.), including potentially multiple sources in a single event scenario.

Commented [A85R84]: I agree

Commented [A86]: Because this guidance is for non-LWRs, why give examples for LWRs? The NRC recently completed SCALE demonstration calculations for heat pipe and high-temperature gas cooled reactors. The NRC ha ... [1]

Commented [A87R86]: Suggest deleting it?

Commented [A88R86]: LWR-SMRs may use LMP so ... [2]

Commented [A89]: The attenuation mechanisms ... [3]

Commented [A90]: Because this guidance is for nor ... [4]

Commented [A91R90]: See above

Commented [A92]: Suggested edit for clarity.

Commented [A93]: It is not clear if transport also ... [5]

Commented [A94R93]: Seems important to add wh ... [6]

Commented [A95]: Should also include "time-depe ... [7]

Commented [A96]: It is not clear what this means

Commented [A97R96]: Should be clarified

Commented [A98]: Specifically the MS element. Sta ... [8]

Commented [A99R98]: Noted

Commented [A100]: I suggest referencing INL/EXT- ... [9]

Commented [A101]: For CP application only. OL ... [10]

Commented [A102R101]: Same as earlier comment.

2.3 DBA Analytical Methods

Deterministic calculations of Design Basis Accident (DBA) sequences are typically performed using one or more computer codes that constitute an analytical model of the plant response. Examples for light water reactors include the RETRAN and RELAP computer codes. If there is a release of radionuclides, the source term discussed above would also be involved. The applicant may elect to describe the analytical methods associated with multiple DBAs in this section of Chapter 2. Multiple subsections (2.3.1, 2.3.2, etc.) can be used to describe multiple methods.

The applicant should describe the overall analytical methodology and identify and describe the significant computer codes used to model the plant response. The applicant should justify the applicability of the analytical methodology to the characteristics of the plant, including a discussion of the underlying experimental or analytical basis. Typically, this is done through topical reports that are incorporated by reference in the SAR or through technical reports that are summarized in the SAR and available for regulatory audits.

Two-Step Licensing

Section 2.3 should mirror the COL guidance but will reflect the preliminary nature of the design information as appropriate. The applicant should describe the technical areas that require research and development to confirm the assumptions and methodologies used to present the adequacy of the design.

Design Certification section?

2.4 Other Methodologies and Analyses

Sections 2.4, 2.5, et al.: Descriptions and results of other generic analyses and methodologies may reside in additional sections in this chapter. The efficiency of presenting additional generic analyses and methodologies will be driven by the nature of the facility and the LMP-based affirmative safety case. These sections are optional and up to the discretion of the applicant. They may be subdivided as appropriate for the topic. Potential examples include:

- Civil and structural analysis
- Piping analysis
- Electrical load analysis
- Stress analysis
- Criticality analysis
- Thermal-hydraulic analysis
- Environmental qualification analysis

These analyses should be pertinent to the LMP-based affirmative safety case (i.e., to safety-related SSCs and/or associated special treatments). The applicant should describe the analytical

Commented [A103]: I believe this should discuss the treatment of uncertainty, whether a conservative or BEPU approach.

Commented [A104R103]: Seems like a good comment

Commented [A105]: I think the guidance is reasonable if topical reports (which are reviewed and approved separately) are used. However, if topicals are not used, then the guidance provided here wouldn't necessarily lead an applicant to provide enough information for the staff to begin a review. For example, there is no discussion about uncertainty and bias determination, presentation of data to support calculation models, benchmark cases, etc.

Commented [A106R105]: See above

Commented [A107]: I generally agree with the contents of this section and have no significant comments. ... [11]

Commented [A108R107]: Suggest to the industry to add the info Tim mentioned?

Commented [A109]: I generally agree. I took the "The applicant should justify the applicability of the analyt ... [12]

Commented [A110]: Because this guidance is for non-LWRs, why give examples for LWRs?

Commented [A111R110]: See similar comments earlier e

Commented [A112]: This statement seems a little vague as currently written. I would recommend something ... [13]

Commented [A113R112]: Seem like a good suggestion

Commented [A114]: I think a high-level discussion of the methodology should include any major assumptions ... [14]

Commented [A115R114]: Seems like a good suggestion

Commented [A116]: I've typically seen technical reports submitted on the docket and incorporated by referer ... [15]

Commented [A117R116]: I agree

Commented [A118]: If this section is intended to include actual methodologies as opposed to referencing a ... [16]

Commented [A119R118]: It seems details are missing in this section.

Commented [A120]: The applicant should provide a description of the assumptions used in the conseque ... [17]

Commented [A121R120]: See above

Commented [A122]: There should be more structure to this than making this optional. For example, if there ... [18]

Commented [A123]: If these analysis descriptions are not provided here then they need to be provided somew ... [19]

methodology and the key inputs and assumptions used. The applicant should justify the applicability of the analytical methodology to the specific analysis. Details of the analyses should be in design records and available for regulatory audits.

3 LICENSING BASIS EVENTS

This chapter documents the selection and evaluation of LBEs that serve as the foundation for the safety case. Because the NEI 18-04 methodology has been approved for use by the NRC in RG 1.233, the scope and content of the Final Safety Analysis Report (FSAR) are focused on presenting the results and not presenting the details of the process except where assumptions and process details are key to the results.

The method for identification and evaluation of the LBEs is described in NEI 18-04, Section 3.2 and in the text that accompanies Figure 3-2. The LBEs evolve through design and licensing, as discussed in NEI 18-04, Section 3.2.3. At the time the SAR is submitted to NRC for review as part of the advanced reactor combined license application, the process will have been completed. The SAR documents the results, not the process, except where assumptions and process details are key to the results.

3.1 Licensing Basis Event Selection Methodology

This guidance assumes that the applicant followed the NEI 18-04 methodology for the selection and evaluation of LBEs. If the applicant did not use parts of the methodology or used an alternative method, the following statement should be added:

> The following ~~departures~~deviations or exceptions from the approved methodology were taken.

The applicant should go on to list each ~~departure~~deviation or exception, describe any alternative method, and provide justification for the approach employed.

The NEI 18-04 methodology affords some flexibility in implementation, so the specific manner in which the methodology was applied should be described as necessary to provide an adequate description of the grouping of event sequence families that are used to define the AOOs, DBEs, and BDBEs. The role of the PRA and resulting risk insights that were used to confirm the completeness and classification of the LBEs should be summarized as needed to gain an understanding of how the LBEs are defined. It is not necessary to repeat aspects of the methodology already covered in NEI 18-04, but rather to point out the specifics of how the methodology was applied within the range of options specified in NEI 18-04. Details of the analyses should be contained in design calculations and retained in the design records.

Safe, stable end states are a key element of the reactor safety case and should be covered in this section. In LWR safety analysis reports, it is generally understood how safe, stable end states are defined in such terms as preventing core damage, maintaining containment integrity, achieving cold shutdown, etc. However, for advanced non-LWRs, the safe, stable end states, including success criteria that are needed to achieve them, need to be defined for the specific technology and design. The plant parameters used to define the end states, core reactivity, reactor power, fuel temperatures, etc., should be identified.

3.2 LBE Summary

3.2.1 Summary Evaluation of AOOs, DBEs, and BDBEs

In this section, a summary of the evaluation of LBEs is presented. This summary should include:

- Tables with brief word descriptions of the AOOs, DBEs, and BDBEs
- Identification of the radionuclide sources associated with each of the LBEs
- A plot of the frequencies, consequences, and uncertainties of these LBEs with comparison against the NEI 18-04 Frequency-Consequence Target in Figure 3-1 of NEI 18-04
- Identification of all risk significant LBEs as defined in NEI 18-04
- Identification of any high consequence BDBEs as defined in NEI 18-04, i.e., those BDBEs with site boundary doses greater than 25 rem
- Definition of the reactor-specific safe, stable end states, described previously and used to establish the success criteria for the safety functions modeled in the PRA, referred to in NEI 18-04 as PRA Safety Functions (PSFs) and reflected in the LBE descriptions

Commented [A124]: Description of consequence assessment assumptions for these events.

Commented [A125]: Somewhere in the document the applicant should describe whether the EAB and site boundary are the same or not. They have not necessarily been congruent for previous licensed reactors.

The word descriptions of the LBE should be in sufficient detail to indicate the PSFs involved in the prevention and mitigation of the LBEs. These PSFs are performed by specific SSCs and are used to determine the SSC safety classifications in Chapter 5. See Table 5-1 of the LMP LBE report[1] and Sections 5.1 and 5.2 as a general reference.

Commented [A126]: Better to include table in guidance document as example format

The plots of the LBE frequencies and consequences should be made with points corresponding to the mean estimates of frequency and consequence with uncertainty bars indicating the 5^{th} and 95^{th} percentiles of the quantified uncertainty distributions in both frequency and dose for each LBE. LBEs with no release and hence no dose should be plotted on the Y-axis.

Identification of risk significant LBEs is based on the criteria in NEI 18-04. The plot discussed above should identify the risk-significant zone for LBEs.

Note: If any part of the plotted uncertainty bands for frequency or dose falls inside the risk significant zone in Figure 3-4, the LBE is regarded as risk significant.

3.2.2 Summary Evaluation of DBAs

In this section, a summary of the evaluation of DBAs is presented. This summary should include a reference to Section 3.6 for the details of the evaluation of all the postulated DBAs as well as the following::

- A table that shows the mapping of DBEs into DBAs with brief word descriptions of the DBEs and DBAs (see Table 5-4 of the LBE report)

Commented [A127]: Better to include in guidance document as an example.

[1] Idaho National Laboratory, "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors, Selection and Evaluation of Licensing Basis Events," Revision 0, August 2019.

- Identification of the Required Safety Functions (RSFs) and SR SSCs selected to perform those functions (with appropriate cross-reference to Chapter 5 where these are derived)
- Identification of the radionuclide sources associated with each DBA
- Justification for the conservatism of the estimated doses for DBAs
  - NRC Regulatory Guide 1.203, "Transient and Accident Analysis Methods,"[1] provides additional discussion of developing appropriate evaluation models for analyzing DBAs
  - Another option is to use the 95th percentile dose from the corresponding limiting DBE mapped into the DBA
- A table that shows the dose consequences of the DBAs for comparison against the 25 rem criterion ~~limit~~ derived from 10 CFR 50.34 (Some DBAs may have no releases and, therefore, no doses.)

Commented [A128]: Criterion, not limit.

3.3 Anticipated Operational Occurrences

This section identifies and describes the plant AOOs that are informed by the PRA event sequence families. AOOs are anticipated event sequences expected to occur one or more times during the life of a nuclear power plant, which may include one or more reactor modules. Event sequences with mean frequencies of $1{\times}10^{-2}$/plant-year and greater are classified as AOOs. AOOs take into account the expected response of all SSCs within the plant, regardless of safety classification.

3.3.1 AOO-1

For each AOO, the following information should be provided:

Commented [A129]: For consistency with the LWR safety philosophy, AOOs are not to lead to any plant damage (e.g., hence the use of SAFDLs for the fuel). There is no equivalent criterion in TICAP. In the extreme, AOOs could cause fuel damage and that would be acceptable as long as the F-C curve is met. No utility would want a design like that, so why not specify a deterministic criterion to prevent plant damage for AOOs? (Similar to GDC and ARDC 10)

- Narrative of the LBE, including the definition of the initial plant conditions and plant operating state, radionuclide source (including whether it involves multiple reactors and sources), initiating events covered in the family, characterization of the responses of SSCs that perform safety functions, identification of whether or not there is a release, and definition of the safe end state

The following information should be provided for any AOO with a release. The applicant may elect to provide some or all of the following information for other AOOs:

Commented [A130]: please clarify whether these are AOOs w/o releases.

- Plots of the responses of key plant parameters
- Tables to describe the mechanistic source term if there is a release (or a reference to the source term description in Chapter 2)
- The mean and uncertainty percentiles of the estimated frequency and dose

Commented [A131]: please clarify by providing examples of key plant parameter responses expected... typical throughout

[1] Regulatory Guide 1.203, Transient and Accident Analysis Methods, U.S. Nuclear Regulatory Commission, December 2005.

The Modular High Temperature Gas-Cooled Reactor (MHTGR) Preliminary Safety Information Document[1] Section 11.6 provides examples of more detailed versions of AOO writeups.

Commented [A132]: The guidance document should include examples of AOO write-ups rather than referring to other documents.

3.3.2, 3.3.3, et al.: The remainder of the AOOs are addressed.

3.4 Design Basis Events

This section identifies and describes the plant DBEs that are informed by the PRA event sequence families. DBEs are 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, and by definition, are less likely than AOOs. Event sequences with mean frequencies of 1×10^{-4}/plant-year to 1×10^{-2}/plant-year are classified as DBEs. DBEs take into account the expected response (including successful and unsuccessful performance of the modeled PSFs) of all SSCs within the plant regardless of safety classification. Note: If uncertainty bands for an AOO or BDBE LBE frequency fail inside the DBE frequency range, such LBEs are evaluated using the NEI 18-04 rules for both LBE categories.

3.4.1 DBE-1

For each DBE, the following information should be provided:

Commented [A133]: For consistency with the LWR safety philosophy, DBEs and DBAs are to maintain coolable core geometry (i.e., no severe accidents). There is no equivalent criterion in TICAP. Therefore, DBEs and DBAs could lead to a severe accident (and the uncertainty that comes along with those conditions) as long as the F-C curve and 50.34 dose limits are met. Why not add a criterion that DBEs and DBAs are not to result in a severe accident (e.g., maintain coolable geometry)?

- Narrative of the LBE, including the definition of the initial plant conditions and plant operating state, radionuclide source and whether it involves multiple reactors and sources, initiating events covered in the family, characterization of the responses of SSCs that perform safety functions, identification of whether there as a release, and definition of the safe end state

For the most limiting DBEs that were used to map into DBAs (see Section 3.2.2, Bullet 1), the following information should be provided. This will enable a comparison of the realistic behavior of the plant (DBE) to the conservatively analyzed behavior (corresponding DBA). The applicant may elect to provide some or all of the following information for other DBEs:

- Plots of the responses of key plant parameters
- Characterization of the response of structures, systems, and components that perform PRA safety functions
- Discussion of relevant phenomena that may impact plant response and mechanistic source terms
- Tables to describe the mechanistic source term if there is a release (This may involve a reference to Chapter 2.)
- The mean and uncertainty percentiles of the estimated frequency and dose

Commented [A134]: see previous comment on clarifying responses of key plant parameters

[1] U.S. Department of Energy, "Preliminary Safety Information Document for the Standard MHTGR," DOE-HTGR-86-024, September 1988.

The MHTGR Preliminary Safety Information Document Sections 15.2 through 15.12 provide useful guidance.

Commented [A135]: See comment above on including guidance in this document and including tables as examples.

3.4.2, 3.4.3, et al.: The remainder of the DBEs are addressed.

3.5 Beyond Design Basis Events

This section identifies and describes the plant BDBEs that are informed by the PRA event sequence families. BDBEs are 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, but are less likely than a DBE. Event sequences with mean frequencies of 5×10^{-7}/plant-year to 1×10^{-4}/plant-year are classified as BDBEs. BDBEs take into account the expected response (including successful and unsuccessful performance of the modeled PSFs) of all SSCs within the plant regardless of safety classification.

3.5.1 BDBE-1

For each BDBE, the following information should be provided:

- Narrative of the BDBE including the definition of the initial plant conditions and plant operating state, radionuclide source including whether it involves multiple reactors and sources, initiating events covered in the family, the response of plant systems, identification of whether there is a release, and definition of the safe end state

The information below should be provided for any high consequence BDBEs as well as other BDBEs to bound the risks associated with the collection of BDBEs. High consequence BDBEs are those with consequences that exceed 25 rem at the Exclusion Area Boundary for the 30-day period beginning at the onset of release. The set of BDBEs that bound the risks is that with the highest products of frequency times dose. The applicant may elect to provide some or all of the following information for other BDBEs:

- Plots of the responses of key plant parameters
- Characterization of the response of structures, systems, and components that perform PSFs
- Discussion of relevant phenomena that may impact plant response and mechanistic source terms
- Tables to describe the mechanistic source term if there is a release (This may involve a reference to Chapter 2.)
- The mean and uncertainty percentiles of the estimated frequency and dose

Commented [A136]: see previous comment on clarifying responses of key plant parameters

The MHTGR Preliminary Safety Information Document R 15-1-1 through 15-2-20 provides useful guidance.

Commented [A137]: See comment above on including guidance in this document and including tables as examples.

3.5.2, 3.5.3, et al.: The remainder of the BDBEs are addressed.

3.6 Design Basis Accidents

This section identifies and describes the plant events that will be included in the licensing basis as DBAs. As established in the NEI 18-04 methodology, DBAs are to be derived from the DBEs by prescriptively assuming that only SR SSCs are available to mitigate postulated event sequence consequences to within the 10 CFR 50.34 dose criteria. ~~limits.~~ DBAs are used to set design criteria and performance objectives for the design of Safety-Related SSCs. Conservative assumptions for the source term and dispersion characteristics are also to be used.

3.6.1 DBA-1

For each DBA, the following information should be provided:

- Narrative of the DBA including the definition of the initial plant conditions and plant operating state, radionuclide source and whether it involves multiple reactors and sources, initiating events covered in the family, the response of plant systems, identification of whether there as a release, and definition of the safe end state
- Plots of the plant response to key plant parameters
- Characterization of the response of SR SSCs
- Evaluation of relevant phenomena that may impact plant response and mechanistic source terms
- Tables to describe the mechanistic source term if there is a release (This may involve a reference to Chapter 2.)
- Description of the conservative calculation used to demonstrate that the 25 rem dose ~~limit~~ criterion in 10 CFR 50.34 is met

Deterministic calculations of DBA sequences are typically performed using one or more computer codes that constitute an analytical model of the plant response. Examples for light water reactors include the RETRAN and RELAP computer codes. If there is a release of radionuclides, the source term discussed in Chapter 2 would also be involved. The analytical methodology may be described in Section 2.3 if the methodology is applicable to multiple DBAs. If not, the applicant should address the analytical methodology in this subsection, following the same guidance provided in Section 2.3.

The MHTGR Preliminary Safety Information Document Section 15.13 provides useful guidance.

3.6.2, 3.6.3, et al.: The remainder of the DBAs are identified and described.

Two-Step Licensing

Chapter 3 should describe the methodology to be used in determining the initial set of LBEs used in the CP. These LBEs should be sufficient to support site suitability evaluation requirements, including Part 100 criteria. The structure of the chapter and sections should follow the structure for the COL guidance. The discussions should be sufficiently robust so the reader can clearly

Commented [A138]: Criteria, not limits.

Commented [A139]: see previous comments - clarify

Commented [A140]: Criterion, not limit.

Is this DBA analysis to compare to the F-C target? If so, the dose metric is 30-day TEDE at EAB.

If this is also where compliance with 10 CFR 50.34(a)(1) or 10 CFR 52.79 siting and safety analysis dose criteria are to be described, the analysis should report both the maximum 2-hour TEDE at the EAB and the TEDE at the outer boundary of the LPZ for the duration of the passage of the plume.

Commented [A141R140]: I think it is a good comment

Commented [A142]: Because this guidance is for non-LWRs, why give examples for LWRs?

Commented [A143R142]: Same earlier comments

Commented [A144]: See previous comment on including useful guidance in this document.

Commented [A145]: see previous comment about considering this to be specific to CP applicants only and adjust supplemental guidance writeup to focus on CPs

see how the methodology will lead to a final set of LBEs to be used in developing the final design, safety margins, operational program content, and FSAR content. The discussion should clearly describe the role of the PRA in determining the initial set of DBEs. The PRA methodology described in Chapter 2 should be used to determine the preliminary assessments of the Licensing Basis Events, as described in the COL guidance for Sections 3.3 through 3.6 above. (Note that ASME/ANS RA-S-1.4-2021 includes guidance on the use of that PRA methodology for CPs.)

The discussions in the various sections of this chapter should provide preliminary assessments of the AOOs, DBEs, and BDBEs, and the basis for those preliminary assessments. Any analyses performed and the methods used in those analyses should be described. The methods and analytical tools (if different from those described in Chapter 2) to be used in deriving the DBAs from the DBEs should be described in Section 3.6.

Design Certification section?

Commented [A146]: CP guidance related to LBE analysis and meeting Part 100 should be consistent with content required for an early site permit (ESP). RG 1.206, Rev 1 describes ESP content for Chapter 15 as follows:

- "Chapter 15 is analogous to a COL FSAR for the potential reactor designs but is limited to Section 15.0.3 addressing the evaluation of the radiological consequences of design basis."

Furthermore…….from SRP 15.0.3:

- "Early Site Permit Reviews: Subpart A to 10 CFR Part 52 specifies the requirements and procedures applicable to the Commission's review of an ESP application for approval of a proposed site. Information required in an ESP application includes a description of the site characteristics and design parameters of the proposed site. The scope and level of detail of review of data parallel that used for a CP review."

4 INTEGRATED EVALUATIONS

This guidance assumes that the applicant followed the NEI 18-04 methodology for the overall plant risk performance summary and the incorporation of defense-in-depth. If the applicant did not use parts of the methodology or used an alternative method, the following statement should be added:

> The following ~~departures~~deviations or exceptions from the approved methodology were taken.

The applicant should go on to list each ~~departure~~deviation or exception, describe any alternative method, and provide justification for the approach employed.

4.1 Overall Plant Risk Performance Summary

This section describes the integrated plant performance for the three cumulative plant performance metrics contained in NEI 18-04 Section 3.2.2, Task 7b for risk to the public from radiation. AOOs, DBEs, and BDBEs are included in the evaluation of overall risk. DBAs are addressed deterministically and not included in the overall risk evaluation.

4.1.1 Site Boundary Dose

This section will address the cumulative risk target that the total frequency of exceeding an exclusion area boundary dose of 100 mrem from all LBEs should not exceed one per plant-year. This section should provide the predicted total risk from the entire range of LBEs from higher frequency, lower consequences to lower frequency, higher consequences. This result should be based on mean values. The margin between the target of 100 mrem and the predicted plant performance should be described.

4.1.2 EAB Boundary Early Fatality Risk

This section addresses the cumulative average individual risk target that early fatality risk within 1 mile of the exclusion area boundary should not exceed 5×10^{-7}/plant-year. This section should provide the predicted average risk from the entire range of LBEs from higher frequency, lower consequences to lower frequency, higher consequences. This result should be based on mean values without uncertainties. The margin between the target and the predicted plant performance should be described.

4.1.3 Latent Cancer Risk

This section addresses the cumulative average individual risk target that the average individual risk of latent cancer fatality within 10 miles of the EAB should not exceed 2×10^{-6}/plant-year. This section should provide the predicted average risk from the entire range of LBEs from higher frequency, lower consequences to lower frequency, higher consequences. This result should be

Commented [A147]: please clarify whether this is all LBE event families

Commented [A148]: This guidance does not require any discussion regarding the analysis except the final result. The applicant should describe what analysis was performed. The description should include:

- The site parameters (e.g. meteorology, off-site population distribution, EAB size) used in the analysis,
- Assumptions on location of individual members of the public,
- 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 target criteria.

Commented [A149]: Clarify whether this result is with or w/o uncertainties. The similar discussion in following sections refer to mean values w/o uncertainties.

Commented [A150]: For uniformity, this section should specify the ground rules for the average individual risk calculations. Certain parameters will be site specific (e.g., meteorology, site size, population distribution around the site, source term, timing and duration of release) but other parameters used in the calculations should be uniformly applied for consistency. These include:

- Acceptable analysis tools.
- What assumptions on EP should be credited?
- Source of dose to an individual (e. g., inhalation, cloud shine, ground shine).
- Breathing rate.
- What assumptions on medical treatment should be credited?
- What latent fatality and early fatality risk factors should be used?
- Modes of operation to be considered in the analysis.
- Consideration of multiple units on a site.

Without some uniformity in the calculations, the applicant can get any conclusion he wants by adjusting input parameters.

Commented [A151]: Same comment as above under 4.1.1

Commented [A152]: Same comment as above.

based on mean values without uncertainties. The margin between the target and the predicted plant performance should be described.

Two-Step Licensing

Section 4.1 should provide a preliminary description of the integrated plant performance for the three cumulative plant performance metrics contained in NEI 18-04 Section 3.2.2, Task 7b for risk to the public from radiation. The PRA methodology described in Chapter 2 should be used in the dose and risk estimates addressed in Section 4.1. If the design is not sufficiently complete to support implementing a full-scope PRA, a performance-based approach may be used to support an NRC finding that the OL application is expected to show that the integrated plant performance targets are met by the design and site.

Design Certification section?

4.2 Defense-in-Depth

The following sections provide a summary of results of Plant Capability DID, Programmatic DID, and the Integrated Assessment of DID, respectively. They reflect the outcomes for the topics listed in NEI 18-04 Table 5-1 Risk-Informed Evaluation of DID Adequacy, including:

- *Evaluation of design attributes for DID*
- *Input to the identification of safety-significant SSCs*
- *Input to the selection of SR SSCs*
- *Evaluation of roles of SSCs in the prevention and mitigation of LBEs*
- *Evaluation of the LBEs to assure adequate functional independence of each layer of defense*
- *Evaluation of single features that have a high level of risk importance to assure no overdependence on that feature and appropriate special treatment to provide greater assurance of performance*
- *Input to SSC performance requirements for reliability and capability of risk-significant prevention and mitigation functions*
- *Input to SSC performance and special treatment requirements*
- *Integrated evaluation of the plant capability DID*
- *Integrated evaluation of programmatic measures for DID*

Note that the above information is provided for background, and there is no requirement to address each topic in the SAR material.

Because the NEI 18-04 methodology has been approved for use by the NRC in RG 1.233, the scope and content of the FSAR are focused on presenting the results and not presenting the details of the process. The summary focus is on safety-significant topics, LBEs, SSCs, and operator actions that receive special treatments as described in NEI 18-04. The summary need not address DID evaluations that did not identify further provisions for DID. The content of the DID Summary provides the foundation for the DID adequacy evaluation baseline as described in

Commented [A153]: This section should be specific to CPs since an OL has same level of detail and design finality as COL w/o reference to DC.

Commented [A154]: As discussed in Workshop #3, only presenting results is too limited. Enough information needs to be presented to provide the basis for DID adequacy. This includes the criteria used determine DID adequacy, a description of how the design meets the criteria and a summary of what are considered the DID measures.

NEI 18-04 stresses the importance of the IDP in determining DID adequacy. However, the IDP in not mentioned. The TICAP guidance should at least require applicants to address the factors listed in Section 5.9.3 of NEI 18-04 to describe how the IDP reached a conclusion on DID adequacy.

NRC staff can only make findings on docketed material. Including the basis and results in the licensing basis allows the NRC to make a determination of adequacy in meeting the regulation. (Note that this same generic issue applies in other areas – reliability & capability targets, etc.)

NEI 18-04 Section 5.9.5. Evidence of the complete DID evaluation should be retained in design records.

Commented [A155]: Not sure what is meant by "evidence" in this context. Shouldn't the complete DID evaluation be retained in the design records?

Commented [A156R155]: I like this comment for clarity

4.2.1 Plant Capability Summary

Commented [A157]: Refer to detailed DID comments provided earlier.

Plant capability DID attributes are listed in NEI 18-04 Table 5-3: initiating event and event sequence completeness, layers of defense, functional reliability, and prevention and mitigation balance. As outlined in NEI 18-04 Table 5-9, the qualitative evaluation should address the evaluation of margin adequacy, multiple protective measures, and prevention and mitigation balance across layers of defense and the physical categories of functional reliability and over-reliance on any single feature.

The application should state affirmatively that the guidelines for plant capability attributes provided in NEI 18-04 Table 5-2 have been evaluated and confirmed. Separate discussions of plant capabilities added as a result of plant capability attribute evaluations should be provided in this section.

Commented [A158]: I'm assuming this means that the underlying evaluation is maintained by the vendor as part of their design basis, correct? And it will be maintained by the applicant and can be audited by the staff if necessary?

Commented [A159R158]: I think it Is emphasizing that the use of NEI 18-04 is part of the licensing basis contained in SAR. The details of the evaluation etc do not need to be in SAR and can be audited by the staff.

Commented [A160R158]: I agree that the details don't need to be in the SAR, but it was interesting that in some sections it explicitly says that information will be in the design basis and then in a couple it simply says that the application should "state affirmatively". I do have a comment elsewhere about how the design basis is maintained post-license issuance, and this sort of ties to it. INL already has the topic listed though in a way.

During the DID adequacy evaluation process, safety-significant SSC functions may have been deemed necessary for DID adequacy. Where so, this information should be documented in tabular form in a manner that is traceable to the LBEs in Chapter 3. The information should include the rationale for the selection of LBE SSCs for NSRST classification and guide the selection of NSRST SSC performance criteria and special treatments as documented in Chapter 7.

4.2.1.1 LBE Margin Summary

This section provides the baseline margins established between the frequencies and consequences of individual risk-significant LBEs and the F-C Target. These margins are established for the risk-significant LBEs within each of the three LBE categories: AOOs, DBEs, and BDBEs. A tabular format example for mean values is shown in Table 4-1, based on Section 2.9.1 of the LMP DID report.[1] Both mean and 95th percentile values should be provided.

[1] Idaho National Laboratory, "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors: Risk-Informed and Performance-Based Evaluation of Defense-in-Depth Adequacy," Revision 0, August 2019. (https://www.osti.gov/biblio/1560534-modernization-technical-requirements-licensing-advanced-non-light-water-reactors-risk-informed-performance-based-evaluation-defense-depth-adequacy) (It should be noted that there is a subsequent revision to this report; however, that version is not readily accessible on the internet.)

Table 4-1. Example Table of LBE Risk Margins

LBE Category	Name	Mean Freq./ plant-yr	Mean Dose (rem)	F-C Target Freq. at LBE Dose/plant-yr[a]	Mean Frequency Margin[b]	F-C Target Dose at LBE Freq. (rem) [c]	Mean Dose Margin[d]
AOO	AOO-5	4.00E-02	2.50E-04	4.00E+02	1.0E+04	1.00E+00	4.0E+03
DBE	DBE-10	1.00E-02	2.00E-03	6.00E+01	6.0E+03	1.00E+00	5.0E+02
BDBE	BDBE-2	3.00E-06	4.00E-03	2.50E+01	8.3E+06	2.50E+02	6.2E+04

Notes:
[a] Frequency value measured at the LBE mean dose level from the F-C Target
[b] Ratio of the frequency in Note [a] to the LBE mean frequency (Mean Frequency Margin)
[c] Dose value measured at the LBE mean frequency from the F-C Target
[d] Ratio of the dose in Note [c] to the LBE mean dose (Mean Dose Margin)

4.2.1.2 Layers of Defense Evaluation

The DID evaluation includes confirmation that plant capabilities for DID are sufficient to prevent and mitigate each risk-significant LBE across all the layers of defense; confirmation that a balance between event prevention and mitigation is reflected across the layers of defense for risk-significant LBEs; and confirmation of sufficient independence between layers of defense for risk-significant LBEs.

For each LBE, each qualitative guideline in NEI 18-04 Table 5-2 should be addressed and any ~~departures~~deviations from the stated criteria addressed. The applicant should provide a summary identification of the layers of defense for each risk-significant LBE and describe the extent of independence between different LBE layers of defense. The applicant should include an affirmative statement that there is no overreliance on a single layer of defense for any risk-significant LBE.

Commented [A161]: Is this defined in NEI 18-04? Or is there at least guidance provided?
Commented [A162R161]: I think it is in NEI 18-04
Commented [A163R161]: Look at table 5-2

An example presentation of the layers of defense evaluation is provided below.

Layer 1—Prevent off-normal operation and AOOs

Summarize NSR provisions for overall plant reliability and availability that minimize the frequency of plant transients.

Layer 2—Control abnormal operation, detect failures, and prevent DBEs

Table 4-2. Example Table of AOOs

AOO	Functions to Minimize Frequency of Challenges to SR SSCs
AOO-1	
AOO-2	
AOO-N	

Layer 3—Control DBEs within the analyzed design basis conditions and prevent BDBEs

Table 4-3. Example Table of DBEs

DBE	Functions to Maintain Frequency of all BDBEs < 10^{-4}/plant-year	Any single design or operational feature relied upon?
DBE-1		
DBE-2		
DBE-N		

Layers 4 and 5—Control severe plant conditions, mitigate consequences of BDBEs, deploy adequate offsite protective actions and prevent adverse impact on public health and safety

Table 4-4. Example Table of BDBEs

BDBE	Functions to Maintain Individual Risks < QHOs	Any single barrier or plant feature relied upon?
BDBE-1		
BDBE-2		
BDBE-N		

4.2.2 Programmatic DID Summary

Programmatic DID attributes are listed in NEI 18-04 Table 5-5: quality/reliability, compensation for uncertainties, and offsite response.

The application should state affirmatively that the guidelines for programmatic capability attributes provided in NEI 18-04 Table 5-6 have been evaluated and included in the design

Commented [A164]: Refer to detailed DID comments provided earlier.

Commented [A165]: Except for perhaps the RAP, the scope of DC does not include operational programs. Therefore, it raises the question of whether or the DID adequacy assessment by a DC applicant would be limited and then supplemented by the COL applicant referencing the DCD. Or could the DC applicant make assumptions for the DID adequacy assessment that would need to be verified as part of COL items?

Commented [A166]: Similar to previous comment about this… the implementation is kept in the design basis and available for staff audit if necessary, right?

Commented [A167R166]: See above

development. Separate discussions of additional programmatic additions or changes as a result of the DID programmatic attribute evaluations, including identification of the safety-significant LBEs leading to additional DID programmatic actions and resulting safety-significant compensatory actions should be provided in this section.

Summary information should be provided for the individual DID evaluation results that led to changes to the protective measures required for adequate programmatic DID.

Commented [A168]: I believe this is an iterative process as the design matures. This wording makes it sound like summary information is to be provided for earlier preliminary designs which were changed… am I reading this wrong?

Commented [A169R168]: Chris may be thinking of CP stage.

It is vague to me in terms of what it looks like and what the value of it. More guidance or example?

4.2.2.1 Programs Required for SR SSC Performance Monitoring

Section 6 should identify the plant-specific programs used to perform monitoring of SR SSCs and to assure human performance and operational controls for risk-significant functions. The requirements include consideration of DID. This section should summarize additions to or modification of the programmatic controls provided in Chapter 6 to account for and manage risk-significant uncertainties as a result of the DID evaluation. The description should be sufficient to identify the DID objectives requiring those actions.

4.2.2.2 Programs Required for NSRST SSC Performance Monitoring

Section 7 should identify the plant-specific programs used to perform monitoring of NSRST SSCs and to assure human performance and operational controls for safety-significant functions. The requirements include consideration of DID and uncertainty. This section should summarize additions to or modification of the programmatic controls provided in Chapter 7 to account for and manage safety-significant uncertainties as a result of the DID evaluation. The description should be sufficient to identify the DID objectives requiring those actions.

4.2.3 Integrated DID Summary

Integrated DID attributes, listed in NEI 18-04 Section 5.9.2 (Table 5-8), include use of the risk-triplet outside of the PRA, state of knowledge adequacy, uncertainty management, and action refinements. They are discussed in NEI 18-04 Section 5.9.4.

In order to support the DID baseline, the applicant should identify (i) additional actions taken as a result of the integrated DID evaluations, (ii) the LBEs leading to those actions, and (iii) the plant or program features addressed by the actions. The applicant should also provide a brief summary of the rationale for the actions.

Two-Step Licensing

The DID CP discussion should be plant capability-centric (Section 4.2.1). While not all of the plant capability DID attributes can be fully addressed at the CP stage, qualitative performance-based objectives for DID may be useful in establishing performance boundaries for FSAR results. It will not be practical to address programmatic DID (Section 4.2.2) and the integrated evaluation of DID adequacy (Section 4.2.3) in the PSAR, and those areas should be reserved for the FSAR unless fundamental to the affirmative safety case.

Design Certification section?

5 SAFETY FUNCTIONS, DESIGN CRITERIA, AND SSC SAFETY CLASSIFICATION

This chapter documents the Required Safety Functions and Required Functional Design Criteria, Principal Design Criteria, Safety Classification of SR and NSRST SSCs, and the Complementary Design Criteria. Because the NEI 18-04 methodology has been approved for use by the NRC in RG 1.233, the scope and content of the FSAR are focused on presenting the results and not presenting the details of the process.

This guidance assumes that the applicant followed the NEI 18-04 methodology for the development of RSFs, RFDC, PDC, and safety classification of SR and NSRST SSCs. If the applicant did not use parts of the methodology or used an alternative method, the following statement should be added:

> The following ~~departures~~deviations or exceptions from the approved methodology were taken.

The applicant should go on to list each ~~departure~~deviaton or exception, describe any alternative method, and provide justification for the approach employed.

5.1 Safety Classification of SSCs

The NEI 18-04 methodology affords some flexibility, so the specific manner in which the classification approach has been applied should be described as necessary to provide an adequate description of the LMP-Based Affirmative Safety Case. It is not necessary to repeat aspects of the methodology already covered in NEI 18-04, but rather to point out the specifics of how the methodology was applied within the range of options specified in NEI 18-04. Details of the analyses should be present in the design records.

The safety classification approach in NEI 18-04 is based on the PSFs that are identified in the definition and selection of the AOOs, DBEs, and BDBEs in Chapter 3. Tables in the following subsections list the SR SSCs and NSRST SSCs and the specific prevention and mitigation functions reflected in the LBEs and responsible for the safety classification.

5.2 Required Safety Functions

This section should present the RSFs, which are the product of applying Step 5a in Figure 3-2 of NEI 18-04. The RSFs are the PSFs that are responsible for successfully mitigating the consequences of all the DBEs inside the F-C Target and for successfully preventing any high consequence BDBEs (i.e., those with doses exceeding 25 rem) from increasing in frequency beyond the F-C Target. A summary-level justification of how the reactor-specific RSFs adequately support the FSFs should be included. Examples of RSFs from the MHTGR and

PRISM reactors are found in the LMP LBE report,[1] and other examples are found in LMP tabletop reports for the Xe-100,[2] Fluoride-Cooled High Temperature,[3] eVinci™,[4] Molten Salt Reactor Experiment,[5] and PRISM[6] reactors are found in the LMP tabletop reports.

Commented [A170]: It would be more helpful to include examples within this guidance document rather than refer to other documents.

5.3 Required Functional Design Criteria and Principal Design Criteria

Regulations (e.g., 10 CFR 50.34 and 10 CFR 52.79) require the identification of PDC. For plants that use the NEI 18-04 methodology, the PDC that flows from the LMP methodology and are needed to support the LMP-based safety case are based on the RSFs and the Required Functional Design Criteria (RFDC). The identification of RFDC is described in Task 7 under Figure 4-1 in NEI 18-04. Each RFDC constitutes a PDC.

Commented [A171]: Do RFDCs address BDBEs? If not, where are the PDCs for safety related SSCs identified in BDBEs identified?

The TICAP guidance should identify what NRC regulations and ARDCs are addressed by the LMP process. This is important information that defines the extent to which the LMP process addresses NRC requirements and will provide consistency for applicants and the NRC staff in preparing and reviewing the SAR.

This section should present the PDCs in terms of the RFDC for each of the RSFs as described in Task 7 of Figure 4-1 in NEI 18-04. These RFDC may be regarded as a decomposition of the RSFs into sub-functions that are necessary and sufficient to support the RSFs. The key elements of the RFDC that should be identified include:

- The design criteria that must be satisfied to meet each of the design-specific RSFs
- A breakdown of each RSF into reactor design-specific sub-functions that are necessary and sufficient to ensure successful completion of the RSF for all the DBAs (The RFDC are qualitatively described in a manner that translates the definition of each RSF into functional design criteria; they form a bridge between the RSFs and the Safety-Related Design Criteria (SRDC), which are assigned to specific SSCs in performing the RSFs.)
- An identification of the design-specific inherent or intrinsic reactor characteristics that must be preserved to support the LMP-based safety case and are credited in the selection of the SR SSCs (Examples of such characteristics include but are not limited to fuel and reactor material properties, geometry, power level, and power density when they enable the satisfaction of the RSFs via passive means.)

It is important to note that RFDC include intrinsic features of the reactor and plant. Table 5-1 provides selected examples of RFDCs that were developed for the MHTGR for two RSFs (see Appendix A, Table A-3 of the LMP SSC report[7] for the entire list covering additional RSFs).

[1] "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors: Selection and Evaluation of Licensing Basis Events," Idaho National Laboratory, March 1, 2020. (https://www.osti.gov/biblio/1700668-modernization-technical-requirements-licensing-advanced-non-light-water-reactors-selection-evaluation-licensing-basis-events)
[2] "High Temperature Gas-Cooled Pebble Bed Reactor Licensing Modernization Project Demonstration," Southern Company, August 2018. (ML18228A779)
[3] "Fluoride-Cooled High Temperature Reactor Licensing Modernization Project Demonstration," Southern Company, December 2019. (ML19247C198)
[4] "Westinghouse eVinci™ Micro-Reactor Licensing Modernization Project Demonstration," Southern Company, August 2019. (ML19227A322)
[5] "Molten Salt Reactor Experiment (MSRE) Case Study Using Risk-Informed, Performance-Based Technical Guidance to Inform Future Licensing for Advanced Non-Light Water Reactors," Southern Company, September 2019. (ML19249B632)
[6] "PRISM Sodium Fast Reactor Licensing Modernization Project Demonstration," Southern Company, December 2018. (ML19036A584)
[7] "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors: Safety Classification and Performance Criteria for Structures, Systems, and Components," Revision 0, Idaho National Laboratory, March 2020. (https://www.osti.gov/biblio/1700535-modernization-technical-requirements-licensing-advanced-non-light-water-reactors-safety-classification-performance-criteria-structures-systems-components)

This table is not intended for direct inclusion in the SAR but is provided for informational purposes.

Table 5-1. MHTGR Required Safety Functions and Associated Required Functional Design Criteria

Required Safety Functions – Subfunctions	***Required Functional Design Criteria***
Retain Radionuclides in Fuel Particles	*I: The reactor fuel shall be designed, fabricated, and operated in such a manner that minor radionuclide releases from the fuel to the primary coolant will not exceed acceptable values.*
Control Heat Generation	*III: The reactor shall be designed, fabricated, and operated in such a manner that the inherent nuclear feedback characteristics will ensure that the reactor thermal power will not exceed acceptable values. Additionally, the reactivity control system(s) shall be designed, fabricated, and operated in such a manner that during insertion of reactivity, the reactor thermal power will not exceed acceptable values.*
Control with Movable Poisons	*IIIa: Two independent and diverse sets of movable poison equipment shall be provided in the design. Either set shall be capable of limiting the heat generation of the reactor to acceptable levels during off-normal conditions.*
Shutdown Reactor	*IIIb: The equipment needed to sense, command, and execute a trip of the control rods, along with any necessary electrical power, shall be designed, fabricated, and operated in such a manner that reactor core shutdown is assured during off-normal conditions.*
Shutdown Reactor Diversely	*IIIc: The equipment needed to sense, command, and execute a trip of the reserve shutdown control equipment, along with any necessary electrical power, shall be designed, fabricated, operated, and maintained in such a manner that the shutdown of the reactor core is assured during off-normal conditions.*
Maintain Geometry for Insertion of Movable Poisons	*IIId: The design, fabrication, operation, and maintenance of the control rod guide tubes, the graphite core and reflectors, the core support structure, the core lateral restraint assemblies, the reactor vessel, and reactor vessel support shall be conducted in such a manner that their integrity is maintained during off-normal conditions as well as provide the appropriate geometry that permits the insertion of the control rods into the outer reflector to effect reactor shutdown.*
	IIIe: The design, fabrication, and operation of the reserve shutdown control equipment guide tubes, the graphite core and reflectors, the core support structure, the core lateral restraint assemblies, the reactor vessel, and reactor vessel support shall be conducted in such a manner that their integrity is maintained during off-normal conditions, as well as provide the appropriate geometry that permits the insertion of reserve shutdown control material to effect reactor shutdown.
Note: The examples above are only a subset of the complete list of MHTGR RFDC.	

5.4 Safety-Related SSCs

Table 5-2 is an example means of displaying the combinations of SSCs that are provided in the design to fulfill each RSF and identifying whether each set of SSCs is available or not on each of the DBEs. There is one table per RSF. The provisions in the design for alternative ways to

perform each RSF constitute one element of Plant Capability DID. The tables identify which combination of SSCs is selected as SR for each RSF. Table 5-2 illustrates an important intermediate step in the LMP methodology but the SSC combinations that are not selected are not design basis information. The applicant is not required to provide Table 5-2 information in the SAR, but the information should be available in the design records.

Table 5-2. Example Table of Evaluation of SSCs for Core Heat Removal RSF

SSC Combinations Capable of Providing Core Heat Removal*	Available for DBE-1?	Available for DBE-2?	…	Available for DBE-N?	Selected as SR?
Reactor Heat Transport System Energy Conversion Area (ECA)	Yes	No	…	No	No
Reactor Shutdown Cooling System Shutdown Cooling Water System (SCWS)	No	Yes	…	No	No
Reactor Reactor Vessel (RV) Reactor Cavity Cooling System (RCCS)	Yes	Yes	…	Yes	Yes
Reactor Reactor Vessel Reactor Building (RB) passive heat sinks	Yes	Yes	…	Yes	No

**The entries in this column and the example selection as Safety-Related are examples from the MHTGR are found in Appendix A of the LMP SSC report.[1]*

The entries in Table 5-2 are an example developed for the MHTGR for a core heat removal RSF. Note that the selection of SR SSCs in this example includes SSCs needed to preserve the intrinsic characteristics of the reactor, such as power level, power density, and shape and selection of materials that enable the RSF to be fulfilled with the other identified SSCs.

A summary, as shown in Table 5-3, should be presented that lists all the SR SSCs, the AOOs, DBEs, and BDBEs, and the PSFs responsible for preventing or mitigating each of these LBEs. Given there are multiple RSFs and that each RSF may require the use of multiple SSCs, there will, in general, be multiple SR SSCs. Operator actions that may be necessary to perform any of these functions should be identified through instrumentation and equipment needed to implement those operator actions.

Commented [A172]: It is stated that alternate ways to perform each RSF constitute one element of DID. However, it appears that these alternate ways are not required to be in the SAR. If these alternate ways are part of DID, what is the basis for leaving them out of the SAR?

Commented [A173]: I would like to have a discussion with them regarding the expectations for maintenance of design records. Since less information is maintained in the FSAR with the TICAP/ARCAP approach, the design basis will be very important years down the road when a licensee wishes to submit a license amendment. I think this is already captured for discussion, but I wanted to note it here.

Commented [A174R173]: Perhaps we need internal discussion on this…

Commented [A175]: Inclusion of tables that are optional in the final guidance document (such as Table 5-2) may not provide the level of clarity desired here – perhaps either highlight or group the optional content separately.

Commented [A176R175]: I agree

Commented [A177]: Table 5-3 clearly identifies what equipment is SR. 'Should be' may not be a strong enough requirement. The identification of SR SSCs needs to be a requirement.

[1] "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors: Safety Classification and Performance Criteria for Structures, Systems, and Components," Revision 0, Idaho National Laboratory, March 2020. (https://www.osti.gov/biblio/1700535-modernization-technical-requirements-licensing-advanced-non-light-water-reactors-safety-classification-performance-criteria-structures-systems-components)

Table 5-3. Example Table of Evaluation of SSCs for PSFs

SR SSC	LBE	LBE Type (AOO, DBE, or BDBE)	PSF
SR SSC-1	LBE-11	?	PSF-11
	LBE-12	?	PSF-12
	…	…	…
	LBE-1n	?	PSF-1n
Additional SR SSCs	…	…	…

This table is organized by SR SSC in order to identify the capabilities of each SR SSC in preventing or mitigating each applicable LBE. These capabilities are used in Chapter 6 to support selection of special treatments. While SSCs are identified as SR for their role in mitigating DBEs and high consequence BDBEs, those SSCs may be used in the mitigation of other LBEs as well, and the table captures that information. The reliability and capability targets for SSCs address all LBEs, not just DBEs and high consequence BDBEs.

The LBE index numbers in the second column should be keyed to LBE indexes identified in Chapter 3 or alternatively spelled out. For each PSF identified in the last column, the spelled-out function should be listed.

5.5 Non-Safety-Related with Special Treatments SSCs

This section presents the technical basis for the selection of NSRST SSCs, presents the NSRST SSCs, and identifies the PSFs for the NSRST SSCs reflected in the LBEs in Chapter 3. Non-Safety-Related (NSR) SSCs that are classified as NSRST because they perform risk-significant safety functions are identified in Section 5.5.1. NSR SSCs that are classified as NSRST because they perform safety functions deemed necessary for adequate DID are identified in Section 5.5.2.

5.5.1 Non-Safety-Related SSCs Performing Risk-Significant Functions

This section identifies the NSR SSCs that perform risk-significant functions and meet the risk significance criteria for classification as NSRST. The risk significance classification is based on applying Steps 4B and 5B in Figure 4-1 in NEI 18-04 and the SSC risk significance criteria noted in Section 4.2.2 of NEI 18-04. Supporting documentation for details, calculations, etc., that were used to establish risk-significant SSC functions should be part of the design records.

There are two types of risk significance criteria that come into play in NEI 18-04 for NSR SSCs. The first criterion is based on identifying NSR SSCs whose prevention or mitigation function is necessary to prevent one or more LBEs from exceeding the F-C Target. Any SSC functions that are risk significant based on this criterion should be identified in a table such as the example in Table 5-4. The purpose of the table is to identify the risk-significant SSCs, the PSFs that are responsible for the classification, and the LBEs that would exceed the F-C Target if the PSFs

were not available. Operator actions that may be necessary to perform any of these functions should be identified in the description of the PSFs, as well as the instrumentation and equipment needed to implement those operator actions.

Table 5-4. Example Table of SSCs Risk-Significant Due to F-C Curve

NSRST SSC	LBE	LBE Type (AOO, DBE, or BDBE)	PSF
RS-NSRST SSC-1	LBE-RS-1	?	PSF-RS-1
RS-NSRST SSC-2	LBE-RS-2	?	PSF-RS-2
	…	…	…
RS-NSRST SSC-N	LBE-RS-N	?	PSF-RS-N

The second risk significance criterion is based on whether the cumulative contribution of the LBEs in which an SSC safety function is failed exceeds 1% of the cumulative risk metrics used for evaluating the risk significance of LBEs. In this case, each risk-significant SSC is classified this way based on an accumulation of risk from multiple LBEs. These risk significant SSCs should be identified in a table such as the following example (Table 5-5). The purpose is to identify the SSC classified as risk significant, the LBEs in which the SSC is failed, and the PSF associated with that LBE. Operator actions that may be necessary to perform any of these functions should be identified in the description of the PSFs, as well as the instrumentation and equipment needed to implement those operator actions.

Table 5-5. Example Table of SSCs Risk-Significant Due to Cumulative Risk

NSRST SSC	LBE	LBE Type (AOO, DBE, or BDBE)	PSF
RS-NSRST-SSC-1	LBE-RS-11	?	PSF-RS-11
	LBE-RS-12	?	PSF-RS-12
	…	…	…
	LBE-RS-1n	?	PSF-RS-1n
Additional RS-NSRST SSCs	…	…	…

5.5.2 NSR SSCs Performing Safety Functions Necessary for Adequate DID

This section identifies the NSR SSCs that are classified as NSRST because they perform safety functions deemed necessary for adequate DID. It should be noted that the SR SSCs identified in Section 5.4 are also key elements of the plant capability DID. Supporting documentation for details, calculations, Integrated Decision-Making Process baseline DID evaluations, etc., that were used to establish SSC functions necessary for adequate DID should be part of the design records.

As with the risk-significant SSCs, the SSC classification for DID adequacy is normally tied to specific LBEs and should be summarized in a table such as Table 5-6. There may be some NSRST SSCs that were identified via the Integrated Decision-Making Panel (IDP) that were not modeled in the PRA and not reflected explicitly in the LBEs due to limitations in the PRA or items screened out of the PRA.

Table 5-6. Example Table of SSCs Risk-Significant Due to DID

NSRST SSC	LBE	LBE Type (AOO, DBE, or BDBE)	PSF
DID-NSRST-SSC-1	LBE-DID-11	?	PSF-DID-11
	LBE-DID-12	?	PSF-DID-12
	…	…	…
	LBE-DID-1n	?	PSF-DID-1n
Additional DID-NSRST SSCs	…	…	…
Note: If an SSC is classified as NSRST but is not associated with a specific LBE, specify "N/A" under LBE and LBE type and identify the SSC function responsible for the NSRST classification under PSF.			

5.6 Complementary Design Criteria

It is important to understand that Complementary Design Criteria (CDC), as they apply to NSRST SSCs, are somewhat, but not totally, analogous to PDC and how PDC apply to SR SSCs. PDC identified through TICAP are defined at the functional level and are based on the RFDC, as described in Section 5.3 of this guidance. PDC are "top-down" in that they correspond to RFDC and are independent of the actual SR SSCs that satisfy the RFDC. NEI 18-04 defines no direct counterpart for RFDC that flow down to NSRST SSCs. Section 5.5 describes how SSCs are identified as NSRST because they perform risk-significant functions or are identified as necessary for DID. CDC are not defined in NEI 18-04 but are useful in the description of the LMP-based safety case. Unlike PDC, CDC are identified "bottom-up" in that they relate to specific NSRST SSCs. Accordingly, while PDC are defined at the functional level, CDC may also be defined at the functional level (related to the PSFs that are satisfied by the NSRST SSCs), or CDC may be expressed more at a system level, directly linked to the NSRST SSCs themselves. The designer has latitude in choosing the approach, as long as the success criteria (discussed below) are clearly conveyed.

The CDC for NSRST SSCs are defined in terms of the success criteria for the PSFs that are represented in the PRA model to prevent and mitigate the LBEs responsible for the safety classification. For example, a PSF safety function might be "Provide adequate heat removal from the reactor following initiating event X," and the success criterion might be "Provide a coolant flow rate of Y kg/sec within Z minutes and maintain maximum fuel temperature less than ZZ." SSCs are classified as NSRST either because the LMP risk significance criteria are met as identified in Section 5.5.1, or the criteria for adequate DID established by the IDP are met

Commented [A178]: I'm still not completing understanding this. Are CDC required or not (I think the answer is "no")? Also, are CDC only for NSRST SSCs? It seems like an applicant could include a CDC for a SR SSC which satisfies an RFDC in order to include a "bottom-up" criterion, but again, I doubt this is required so therefore will not be done.

Commented [A179R178]: I understand CDC is not currently discussed in NEI 18-04.

UART members has raised a question in the past whether NRC has taken an official position on CDC.

as identified in Section 5.5.2. The reliabilities and capabilities that are modeled in the PRA for the PSFs associated with the SSC trigger the meeting of the risk significance or DID adequacy criteria for NSRST classification. These, in turn, serve to prevent and/or mitigate a specific set of LBEs. Hence the CDC for the NSRST SSCs are directly tied to the success criteria established in the PRA for the PSFs responsible for the SSC classification as NSRST.

These should be presented in tabular form by listing the SSC, the PSF(s) responsible for its safety classification as NSRST, and the design criteria that are necessary and sufficient to meet the PSF (see example in Table 5-7). There may be more than one PSF that is associated with the NSRST classification and more than one design criterion for each PSF because the SSC may be represented in multiple LBEs.

Commented [A180]: It's not clear how this scope summary relates to the PDC scoping requirement in Part 50 App. A:

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

Table 5-7. Example Table of NSRST SSCs with Corresponding CDC

NSRST SSC	PRA Safety Function	Complementary Design Criteria
NSRST SSC-1	PSF-11	Design criterion for PSF-11
	PSF-12	Design criterion for PSF-12
	…	…
	PSF-1n	Design criterion for PSF-1n
Additional NSRST SSCs	…	…

Two-Step Licensing

Chapter 5 includes preliminary determination of the RSFs and RFDC, PDC, safety classification of safety-related and NSRST SSCs, and the Complementary Design Criteria leading to specific NSRST SSC design requirements. The LMP methodology for assessing safety functions, design criteria, and SSC safety classification, as described in Chapter 5 of the COL guidance, draws on results from the initial PRA. The PRA methodology described in Chapter 2 should be used in the preliminary determination of Required Safety Functions, determination of Required Functional Design Criteria, Principal Design Criteria, and SSC safety classification. The structure and preliminary content of the Chapter 5 sections should follow the structure and content of the COL guidance Chapter 5 content. It should be noted that as the design evolves and construction continues, changes to the PRA modeling and results can be expected. Thus, the results in Chapter 5 are preliminary and can be expected to change as the design is finalized and the FSAR is developed. At the CP stage, the use of performance-based bounding conditions for RFDC may be possible depending on the simplicity of the design and corresponding confidence in the PRA outputs.

Commented [A181]: And CDCs?

Thus, the PSAR content for Chapter 5 should include functional decomposition of FSFs to RSFs, a preliminary set of RFDC /PDC with performance-based criteria, and preliminary SSC classifications based on preliminary PRA results.

Design certification discussion?

6 SAFETY-RELATED SSC CRITERIA AND CAPABILITIES

Commented [A182]: This is a very good approach to documenting the SR SSCs, the RSFs, the SRDC and the ST applied to each SSC. It also covers those SSCs that credited to protect SSCs for DBEHLs

In Section 5.4, the SR SSCs were identified, and the bases for their classification were provided. Chapter 6 provides further detail on the criteria and capabilities of all SR SSCs in the LMP-based affirmative safety case, consistent with the NEI 18-04 methodology. This further detail includes SRDC, reliability and capability performance-based targets, and special treatment requirements to provide sufficient confidence that the performance-based targets intended in the design will be achieved in the construction of the plant and maintained throughout the licensed plant life. Chapter 6 also summarizes design requirements for NSR SSCs that provide confidence that the NSR SSCs will not adversely impact the ability of SR SSCs to support RSFs in the event that a hazard occurs at the DBEHL. Note that these NSR SSCs may be classified as NSRST or NST in the SSC safety classification process of LMP, but that classification process is based on their normal PSFs, not on the passive function of protecting (or at least not impairing) SR SSCs as discussed in Section 6.1.3.

As discussed in NEI 18-04, an LMP-based safety case is both risk-informed and performance-based. The formulation of the SRDC should be quantitatively framed so that the successful performance of the SSC may be confirmed via calculation or monitoring to facilitate the performance-based attribute of the safety case, as discussed in Section 6.1. Additional performance-based elements are incorporated in the selection of SSC reliability targets that are used to establish special treatment requirements covered in Section 6.2.

This guidance assumes that the applicant followed the NEI 18-04 methodology for the establishment of SR SSC criteria and capabilities. If the applicant did not use parts of the methodology or used an alternative method, the following statement should be added:

> The following ~~departure~~deviations or exceptions from the approved methodology were taken.

The applicant should go on to list each ~~departure~~deviation or exception, describe any alternative method, and provide justification for the approach employed.

6.1 Design Requirements for Safety-Related SSCs

This section describes the outputs of NEI 18-04 Section 4.1, Task 7. Details of the analyses and justifications for the development of SRDCs should be in the design records.

6.1.1 Design Basis External Hazard Levels

One general category of design requirements flows from the need to protect the SR SSCs in the performance of their RSF from design basis external hazards. Each external hazard is characterized by a Design Basis External Hazard Level (DBEHL) (e.g., wind speed). This is discussed in NEI 18-04 Section 3.2.2, Task 6, and the following text from the first page of Section 4 in NEI-18-04.

> *It is noted that there will be design requirements to protect all SR SSCs from any adverse impacts of any DBEHLs. This may lead to design requirements to prevent any adverse impacts from the failure of an SSC classified as NST or NSRST that could otherwise prevent an SR SSC from performing its RSFs.*

It is important to note that the DBEHLs go beyond environmental hazards originating external to the plant. As defined in NEI 18-04, the scope of the DBEHLs includes external hazards such as seismic events, wind including tornados and wind-generated missiles, external flooding, hazards from external facilities, and internal plant hazards such as internal fires, internal floods, high energy line breaks, and internally generated missiles. These internal plant hazards are frequently described as "area events." Guidance on the scope of hazards may be found in Chapter 3 of the Standard Review Plan (NUREG-0800). The concept is to ensure that hazards with a frequency down to 10^{-4}/plant-year are identified so that design requirements identified in Section 5.4 for the SR SSCs to protect them against any DBEHL can be specified. Note that the DBEHLs are one of the inputs to the analysis of hazards in the PRA.

The DBEHLs should be summarized in this section. A tabular form such as Table 6-1 is recommended. The determination of the DBEHLs is documented elsewhere in the SAR.

Table 6-1. Example Table of Design Basis External Hazard Levels

Hazard	Design External Hazard Level
Seismic Events	Specify design basis earthquake parameters
Tornado Wind Speed	Specify design basis wind speed
External Flood	Specify design basis flood levels
Internal Fires	Identify fire areas where SR SSCs are located and fires may occur
Internal Flood	Identify flood areas where SR SSCs are located and flood may occur
High energy line breaks (HELB)	Identify areas where SR SSCs are located and a HELB may occur
Other hazards that SR SSCs are protected against	Identify areas and specify appropriate parameters.

Note: The development of the DBEHLs is addressed by ARCAP and summarized in SAR Chapter 2.

Two-Step Licensing

The content for Section 6.1.1 addressing DBEHLs should be as complete as possible based on the site characterization required in 10 CFR 50.34(a). The external and internal hazard levels are inputs to the PRA and to SSC design. Thus, including details on this information in the PSAR will support demonstrating the viability of the design and developing the FSAR consistent with Chapter 6 of the COL guidance.

Commented [A183]: Beyond listing a hazard level value in a table, the guidance should specify that the applicant describe how each DBEHL is used as an input parameter to the design analysis of safety-related SSCs. For example:
- Regarding the seismic design input, where would the application describe the design ground motion response spectra, floor response spectra, damping values, and time histories? Refer to SRP Section 3.7 for relevant topics to be addressed.
- For design basis wind hazards, where would an application describe the design-basis wind loadings including design wind velocity and its recurrence interval and the methods used to transform the wind velocity into an effective pressure applied to surfaces of structures? Refer to SRP Section 3.3 for relevant topics to be addressed.

Design Certification

Section 6.1.1 should describe the bounding site characterizations considered in the definition of the DBEHLs. To the extent that the hazard characterizations impact internal hazard levels, those impacts should be identified and included in the PRA.

6.1.2 Summary of SRDC

In the text for Task 7 of Figure 4-1 in NEI-18-04, it is stated:

> *"The RFDC, SRDC, the reliability and capability targets for SR and NSRST SSCs, and special treatment requirements for SR and NSRST SSCs define safety-significant aspects of the descriptions of SSCs that should be included in safety analysis reports."*

The RFDC are identified in Section 5.3, and the RSFs that they support are identified in Section 5.2. For each of the RFDC, this section should identify a set of SRDC appropriate to the SR SSCs selected to perform the RSFs. These SRDC exclude Special Treatment Requirements, which are separately covered in Section 6.2. The RFDC, which are expressed in the form of functions and involve collections of SSCs and intrinsic capabilities of the plant, may be viewed as a bridge between the RSFs and the SRDC. The SRDC are more detailed requirements for specific SR SSCs in the performance of the RSF functions in specific DBAs. Examples of SRDCs that were developed for the MHTGR are found in Appendix A of the LMP SSC report.[1]

Commented [A184]: It would be good to include some of these examples in this guidance document

For the SRDC, the following information is presented in tabular form, as shown in Table 6-2:

- The first column contains the SSC names.
- The second column provides brief SSC functional descriptions.
- The third column lists the RFDC that the SR SSCs support. Most likely, there is only one RFDC associated with each SR SSC, but if there is more than one, all should be listed. Note that the links from the SR SSCs back to the LBEs that define the RSFs are provided in Chapter 5.
- The fourth column lists the SRDCs. There may be more than one SRDC for each SR SSC.

Commented [A185]: This is a good format for documenting what design criteria are applicable to a specific SR SSC.

[1] "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors: Safety Classification and Performance Criteria for Structures, Systems, and Components," Revision 0, Idaho National Laboratory, March 2020. (https://www.osti.gov/biblio/1700535-modernization-technical-requirements-licensing-advanced-non-light-water-reactors-safety-classification-performance-criteria-structures-systems-components)

Table 6-2. Example Table of Safety-Related Design Criteria for SSCs

SR SSC	Functional Description	RFDC	SRDC
SR SSC1	Functional Description of SR SSC1	RFDCx	SRDC11
			SRDC12
			…
			SRDC1n
Additional SR SSCs	…	…	…

6.1.3 Summary of DBEHL-related Requirements for NSR SSCs

Chapter 6 also identifies DBEHL-related design requirements for non-SR SSCs. These design requirements are to support the special safety functions that are applied to the non-SR SSCs to prevent adverse impacts on the ability of the SR SSCs to perform the RSFs. An example is the requirement for anchorage to prevent a non-safety-related SSC from failing in such a manner that it would impact an SR SSC and cause it to fail to perform its RSF.

It is important to note that the non-SR SSCs covered in these requirements are not for the SSC functions that they normally perform but for the special function of preventing any adverse impact on the capability of any SR SSC in the performance of the RSF. The DBEHL includes external hazards such as seismic events as well as internal plant hazards such as internal fires and floods, turbine missiles, and high energy line breaks. When an NSR SSC is required to protect the SR SSCs in their ability to perform their RSFs, such NSR SSCs are not necessarily NSRST. The NSRST classifications are based on the PSFs these SSCs perform to prevent or mitigate event sequences and not functions that are focused on protecting the SR SSCs.

For the NSR SSCs that have design requirements to protect the SR SSCs in the performance of the RSFs in response to a DBEHL, the following information in tabular form should be provided, as illustrated in the example below (Table 6-3).

Commented [A186]: This is excellent that these are identified here.

- The first column identifies the NSR SSCs.
- The second column lists the RSFs and SR SSCs that are protected.
- The third column identifies the DBEHLs that are associated with these requirements.
- The fourth column identifies the specific design requirement for the function to protect the SR SSCs for each of the DBEHLs. Note that this function is different from the PSFs for the same NSR SSCs.

Table 6-3. Example Table of NSR SSCs Protecting SR SSCs from DBEHLs

NSR SSC	Protected RSF and SR SSC	DBEHL	NSR SSC Design Requirement
NSR SSC-1	RSF/SR SSCx	DBEHL-1	NSR DC-11
		DBEHL-2	NSR DC-12
		…	…
		DBEHL-n	NSR DC-n
Additional NSR SSCs	…	…	…

6.2 Special Treatment Requirements for SR SSCs

NEI 18-04 adopted the definition of special treatment that is provided in RG 1.201, which was developed for implementing 10 CFR 50.69.

> *"…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."*

Anything that is done beyond procuring commercial-grade equipment to provide increased assurance in the capability and reliability of the SSC falls into the category of special treatment. Hence, all the design requirements provided in Section 6.1 are part of the special treatment. This section identifies the additional special treatments that are applied to SR SSCs. Candidate special treatments (STs) for consideration are identified in Table 4-1 of NEI 18-04.

As noted in Section 4.4.5 of NEI 18-04, the selection of STs for all safety-significant SSCs (SR and NSRST) are informed by a set of targets for the reliability and availability of the SSCs in their prevention functions as well as targets for the capability of the SSCs in the performance of their mitigation functions. These specific targets should not be stated in the SAR but should be available in the plant records. The focus of this section in the application is to produce the resulting special treatment requirements.

Commented [A187]: The role of reliability and availability targets is significant in formulating special treatments for SSCs and human actions. These targets may need to be in SAR as part of the licensing/design bases.

The introduction to Chapter 6 says " This further detail [Chapter 6] includes SRDC, reliability and capability performance-based targets, and special treatment requirements to provide sufficient confidence that the performance-based targets intended in the design will be achieved in the construction of the plant and maintained throughout the licensed plant life. This statement appears to support that these targets should be in SAR.

Commented [A188R187]: I think INL covers it.

For the selected STs, the license application should identify the treatments in the license application with details available for NRC audits in the license application.

The STs should be summarized in tabular form (see Table 6-4) by listing each SR SSC, providing a brief performance-based functional description and identifying which ST has been selected for each SR SSC.

Table 6-4. Example Table of SR SSC Special Treatments

SR SSC	Functional Description	SR SSC Special Treatments
SR SSC$_1$	Short SSC functional description for SR SSC$_1$	SR SSC$_1$ Special Treatment No. 1
		SR SSC$_1$ Special Treatment No. 2
		…
		SR SSC$_1$ Special Treatment No. n
Additional SR SSCs	…	…

6.3 System Descriptions for SR SSCs

This section provides system descriptions for SR SSCs. These descriptions include the specific design features for SR SSCs that are responsible for meeting the SRDC and fulfilling their RSFs to mitigate the DBAs. This description should include features that demonstrate system capability and reliability for both prevention and mitigation of LBEs, as applicable.

6.3.1 Description for SR SSC-1

This description should include:

- Simplified schematic diagram
- Narrative design descriptions that address the design aspects relevant to the performance of the RSFs systems, including:
    - The system purpose in the context of supporting the RSFs
    - The specific SSC function in the context of supporting the RSFs
    - System location and environmental conditions
    - Key design features relevant to the performance of RSFs
    - Seismic and industry (e.g., ASME and IEEE) code classifications and the design codes applicable to the SR SSC
    - A description of system operation, including an account of the performance modes of system operation relevant to the RSFs
    - Identification of operator actions needed to implement the RSFs
    - Controls and displays needed to accomplish RSFs
    - Logic circuits and interlocks needed to support RSFs
    - Electric power, support systems, and interface requirements needed to support the RSFs
    - Equipment to be qualified for harsh environments as needed to meet SR SSC special treatment requirements defined in Section 6.2

Commented [A189]: Add “Materials of construction” to the list.

6.3.2, 6.3.3, et al.: Descriptions of the remainder of the SR SSCs are provided.

Two-Step Licensing

While the classification of SSCs from Chapter 5 is preliminary, the approach and methodologies to be used in developing the FSAR Chapter 6 content should be clearly described. The description should include any consensus codes and standards used or expected to be used in the design of SR SSCs. The descriptions for safety-related systems should be provided at a functional level and should identify the performance-based requirements needed for individual major components. Any safety-related first-of-a-kind components should be identified, as should plans for component performance validation and acceptance criteria. The guidance for other system description content in Chapter 6 should be tailored to the information available at the CP stage. The preliminary results from Chapter 5 should be used to frame the development of SRDC and special treatment requirements for SR SSCs. The content of this Chapter 6 should use the tabular format provided in Chapter 6 of the COL guidance.

Design Certification discussion?

Commented [A190]: may want to clarify that this is PSAR since OL and COL have same level of detail and design finality

7 NSRST SSC CRITERIA AND CAPABILITIES

In Section 5.5, the NSRST SSCs were identified. Chapter 7 provides further detail on the role of each NSRST SSC in the LMP-based affirmative safety case, consistent with the NEI 18-04 methodology. Complementary design criteria for NSRST SSCs are covered in Section 5.5.4. The remaining criteria and capabilities for NSRST SSCs include reliability and capability performance targets and special treatment requirements to provide sufficient confidence that the performance targets will be maintained throughout the life of the licensed plant. Anything that is done beyond procuring commercial-grade equipment to provide increased assurance in the capability and reliability of the SSC falls into the category of special treatment

As noted in Section 4.4.5 of NEI 18-04, the selection of STs for all safety-significant SSCs (SR and NSRST) are informed by a set of targets for the reliability and availability of the SSCs in their prevention functions as well as targets for the capability of the SSCs in the performance of their mitigation functions. These specific targets should not be stated in the SAR but should be available in the plant records. Hence, this chapter should be limited to defining STs for NSRST SSCs that are necessary and sufficient for meeting the CDCs and the supporting reliability and capability targets.

This guidance assumes that the applicant followed the NEI 18-04 methodology for the establishment of SR SSC criteria and capabilities. If the applicant did not use parts of the methodology or used an alternative method, the following statement should be added:

> The following ~~departure~~deviations and exceptions from the approved methodology were taken.

The applicant should go on to list each ~~departure~~deviation and exception, describe any alternative method, and provide justification for the approach employed.

7.1 Special Treatment Requirements for NSRST SSCs

This section documents the special treatment requirements for NSRST SSCs.

NEI 18-04 adopted the definition of special treatment provided in RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance,"[1] which was developed for implementing 10 CFR 50.69:

[1] Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," U.S. Nuclear Regulatory Commission, May 2006. (ML061090627)

Commented [A191]: Good method for documenting special treatment requirements.

Commented [A192]: This is very helpful to continue to reiterate and define the language being used because for those who have been in the industry for some time equate the increased level of quality standards, reliability, and equipment qualification for nuclear safety related equipment to be normal and that special treatment would be something beyond that.

Formatted: Body Text

Commented [A193]: There may a nexus here to risk-informed TS LCO completion times that are currently afforded to operating plants who have amended their TS's to allow for this program. However, would the NRC expect a description of the program on reliability targets in Chapter 8?

Commented [A194]: More discussion is needed regarding the need to include reliability and performance targets in the SAR (i.e., licensing basis).

> "…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."

Anything that is done beyond procuring commercial-grade equipment to provide increased assurance in the capability and reliability of SSCs falls into the category of special treatment. Section 6.1 identified NSR SSC design requirements associated with protecting SR SSCs from design basis external hazards. Hence, if an NSR SSC identified in Section 6.1.3 is also an NSRST SSC, then all the design requirements provided in Section 6.1 are part of the NSRST's special treatment. This section identifies the additional special treatments that are applied to NSRST SSCs.

For the selected STs, the license application should identify the treatments in the SAR with details available in the design records.

The STs should be summarized in tabular form (see Table 7-1) by listing each NSRST SSC, providing a brief functional description, and identifying which ST has been selected for each NSRST SSC.

Table 7-1. Example Table of NSRST SSC Special Treatments

NSRST SSC	Functional Description	NSRST SSC Special Treatments
NSRST SSC1	Short SSC functional description for NSRST SSC1	NSRST SSC1 Special Treatment No. 1
		NSRST SSC1 Special Treatment No. 2
		…
		NSRST SSC1 Special Treatment No. n
Additional NSRST SSCs	…	…

7.2 System Descriptions for NSRST SSCs

This section provides system descriptions for NSRST SSCs. These descriptions include the specific design features for NSRST SSCs that are responsible for meeting their safety-significant functions identified in the LBEs responsible for the classification as NSRST. This description should include features that demonstrate system capability and reliability for both prevention and mitigation of LBEs, as applicable. It is expected that these system descriptions are generally less detailed than those provided for SR SSCs in Section 6.3, even though the list of covered items is similar.

The bulleted list in 7.2.1 is very similar to the list in 6.3.1 for SR SSCs. The NSRST SSC information does not need to be as comprehensive and detailed as the SR SSC information.

7.2.1 Description for NSRST SSC-1

This description should include:

Commented [A195]: Add "Materials of construction" to the list.

- A simplified schematic diagram
- Narrative design descriptions that address the design aspects relevant to the performance of the safety-significant functions systems including, as applicable:
  - The system purpose in the context of supporting the safety-significant functions
  - Significant functional performance-based characteristics in performing safety-significant functions
  - System location
  - Key design features relevant to the performance of safety-significant functions
  - Seismic and industry (ASME, IEEE, etc.) code classifications and the design codes applicable to the NSRST SSC
  - Description of system operation, including a description of the performance modes of system operation relevant to the safety-significant functions
  - Identification of any operator actions needed to implement safety-significant functions
  - Controls and displays needed to support safety-significant functions
  - Logic circuits and interlocks needed to support safety-significant functions
  - Electric power, support systems, and interface requirements needed to support the safety-significant functions
  - Equipment to be qualified for harsh environments as needed to meet SR SSC special treatment requirements defined in Section 7.2

Commented [A196]: Why does this refer to SR SSC special treatment requirements? Section 7.2 only refers to SR SSCs for comparison of level of detail. Should this refer to NSR SSCs or refer to a different section than 7.2, such as special treatment requirements for NSRST SSCs in Section 7.1?

7.3.2, 7.3.3, et al.: Descriptions of the remainder of the NSRST SSCs are provided.

Two-Step Licensing

Chapter 7 content addressing NSRST SSCs should follow the approach used in Chapter 6 for SR SSCs. A preliminary list of NSRST SSCs should be populated to the extent the DID evaluation has been performed and described in Chapter 4 and should use the tabular format provided in Chapter 7 of the COL guidance. Descriptions for NSRST systems as described in Chapter 7 should be developed to identify safety-significant functions to be provided by those systems. This section should be finalized in the FSAR once the Integrated DID Evaluation task has been completed for the final design.

Commented [A197]: See comment above re: distinction between CP and OL applicants and PSAR v. FSAR

8 PLANT PROGRAMS

Depending on the nature of the design and the LMP-based affirmative safety case, special treatments for SR SSCs and NSRST SSCs may involve programs relied upon to provide reasonable assurance that (i) reliability and performance targets are met throughout the plant lifetime and (ii) safety-significant uncertainties are effectively addressed as part of DID. This chapter should include a discussion of such programs as applied across different plant lifetime phases, i.e., design, construction, testing, and operations. Program areas could include but are not limited to quality assurance, equipment qualification, human factors engineering, in-service inspection, and maintenance and reliability assurance activities. The selection of specific SSC special treatments is described in Chapters 6 and 7 of the SAR.

The guidance in this chapter is not intended to address all plant programs. It focuses on those that play a substantive role in the LMP-based affirmative safety case. For example, nuclear power reactors require a radiological protection program for plant workers in order to ensure compliance with occupational dose regulations in 10 CFR Part 20. However, the radiological protection program for plant workers is unlikely to be a component of the affirmative safety case, which is based on protection of the public from radiological hazards.

The chapter should provide an overview of the special treatment programs, addressing the purpose, scope, and performance objectives as well as applicability to SR SSCs, NSRST SSCs, or operations activities. The intent is not to provide a detailed description of how each program works. Such details will inevitably evolve over time and are fully described in owner-controlled records. However, the performance objectives of the program, both quantitative and qualitative, should be provided to enable an understanding of the adequacy of the program relative to the special treatments identified for SR SSCs and NSRST SSCs. This information should be included in the SAR or in documents that are incorporated by reference (IBR).

This chapter should describe the set of programs that are used for special treatments for SR or NSRST SSCs as described in Chapters 6 and 7, respectively, that provide reasonable assurance (i) reliability and performance targets are met throughout the plant lifetime and (ii) safety-significant uncertainties are effectively addressed as part of DID.

Examples of possible program topics are provided below. The actual program topics are determined by the special treatments documented in Chapters 6 and 7. Not all of the following examples may be required for a given application, and program topics not listed in the examples below may be required.

- Quality assurance programs (design, construction, operations)
- Startup testing
- In-service testing
- In-service inspection
- Equipment qualification
- Performance monitoring

Commented [A198]: This guidance is insufficient to properly document important aspects of the program. The following describes two examples where additional guidance is appropriate:

First example – 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. NRC RG 1.160 provides additional guidance on one way to meet the requirements of 10 CFR 50.65. Current evolutionary LWR licensees have maintenance program descriptions that incorporate by reference NEI 07-02A, "Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed under 10 CFR Part 52." The staff, in its SER, determined that incorporation of NEI 07-02 by reference in a COL application will provide an acceptable method for (1) complying with the requirement in 10 CFR 52.79(a)(15) that FSARs contain a description of the program, and its implementation, for monitoring the effectiveness of maintenance to meet the requirements of Section 50.65 and (2) satisfying the acceptance criteria of SRP 17.6. Topics addressed in the NEI template include:

- Maintenance rule scoping per 10 CFR 50.65(b)
- Monitoring and corrective action per 10 CFR 50.65(a)(1)
- Preventive maintenance per 10 CFR 50.65(a)(2)
- Periodic evaluation of monitoring and preventive maintenance per 10 CFR 50.65(a)(3)
- Risk assessment and risk management per 10 CFR 50.65(a)(4)

... [20]

Commented [A199]: Second example - Example) Reliability Assurance Program (RAP()
NUREG-0800, Section 17.4 provides a discussion on the RAP required by 10 CFR 52. 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 RAP is coordinated/integrated with the maintenance, surveillance and ISI/IST programs
- 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).

- Reliability assurance

The applicant should provide the following information in this chapter of the SAR:

- Program topic and summary description
- SSC applicability (SR, NSRST, or NST)
- Controlling program document including title, document number, revision number, and effective date. The expectation is that each program supporting the LMP-based affirmative safety case will have an associated controlling document maintained by the design authority or licensee. This document could be a topical report that has been reviewed and approved by the NRC or an internal document.
- Standard industry program references, if any, used as a basis for a program, e.g., standards or Nuclear Energy Institute guidance documents (For each reference, it should be noted whether it is incorporated by reference [IBR] or provided for information [Ref] as described in Section 1.4.)
- In the event of any significant ~~departures~~deviations from an IBR standard industry program, identification of the departure with appropriate justification

The applicant has the option of providing the information in this chapter in a tabular format or in text. For the former, an example table with entries for programs is provided in Table 8-1~~Table 8-1~~. (Note that the example entries are meant to be illustrative, not approved content for a specific program, and not comprehensive.) For the latter, the applicant would be expected to use a section for each program topic (e.g., 8.1 Quality Assurance Program and 8.2 Reliability Assurance Program).

Commented [A200]: See above comments.

Commented [A201]: The application should have a program description (e.g., NEI 07-02) that is part of the licensing basis and submit to "50.59" like change control. The staff cannot reply solely on an "internal" document.

Table 8-1. Example Table of Special Treatment Programs

Program Topics / Objectives	SSC Applicability			Program References and Departures	IBR	Ref
	SR	NSRST	NST			
Quality Assurance (QA) Program The QA Program comprises the planned and systematic activities implemented to demonstrate that SSCs and activities will fulfill their requirements. It provides assurance that the actual plant configuration is consistent with the capabilities modeled in the licensing basis event analyses. Depending on the requirements associated with the specific SSC, the full range of QA controls may be applied, or a subset thereof.	√	√	√	Insert controlling program document information (e.g., XYZ123, Quality Assurance Topical Report, Rev. 0, July 1, 2025).	√	
				ASME NQA-1, Quality Assurance Requirements for Nuclear Facility Applications, 2019.	√	
				Departures: The Quality Assurance Program is based on NQA-1, but a graded approach is applied as described in the Quality Assurance Topical Report.		
Reliability Assurance Program The plant reliability assurance program controls the reliability and availability targets of SR SSCs and NSRST SSCs consistent with the PRA.	√	√		Insert controlling program document information (e.g., ABC789, Reliability Assurance Program).	√	
				NUREG-0800 Section 17.4, Reliability Assurance Program Rev. 0, 2007.	√	
				ASME Section XI-Rules for Inservice Inspection of Nuclear Power Plant Components, Division 2, Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants, 2019.	√	
				Departures: NUREG-0800 Section 17.4 applies to SR SSCs only as described in the Reliability Assurance Program document.		
Maintenance Program The plant maintenance controls the performance of maintenance activities in the plant. It provides assurance that the actual plant configuration is consistent with the capabilities modeled in the licensing basis event analyses.	√	√	√	Insert controlling program document information (e.g., LMN456, Plant Maintenance Program).	√	
				Regulatory Guide 1.160, Revision 4, Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, August 2018.		√
				NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 4F, Nuclear Energy Institute, April 2018.		√

Commented [A202]: Deviation?

Commented [A203]: While it is OK to IBR a TR, the guidance should provide information regarding what needs to go into the TR.

Commented [A204]: An application doesn't really IBR an SRP Section.

Commented [A205]: Deviation?

Commented [A207]: What about NEI 07-02 for SAR content regarding the maintenance rule program?

Commented [A206]: Not something that would be expected to be submitted by a DC applicant.

Two-Step Licensing

The plant program descriptions for the PSAR should be largely the same as for the FSAR content in Chapter 8 of the COL guidance. The PSAR should contain complete identification of design, manufacturing, and construction programs to be used and any references to standardized industry programs that are used as frameworks or templates for these programs. Similarly, the CP should contain general descriptions of the operational programs that may be incorporated in the FSAR to support the affirmative safety case. The PSAR content for Chapter 8 should be reasonably similar to the FSAR content and should use the tabular format provided in Chapter 8 of the COL guidance, where any preliminary information is clearly identified.

Commented [A208]: could be just commitments

Commented [A209]: not required for a CP applicant

Commented [A210]: Disagree. The expectation on level of detail of detail, if any, is much different, except for perhaps QA. Cannot make such a broad statement in guidance

Design Cetification discussion?

D. SUMMARY

The guidance contained in Section C of this report is one acceptable means of providing content for the portion of an advanced reactor SAR related to the application of the RIPB LMP-based affirmative safety case methodology described in NEI 18-04.

The Section C guidance applies to a COL applicant under 10 CFR Part 52, Licenses, Certifications, and Approvals for Nuclear Power Plants and to an OL applicant under 10 CFR Part 50.... The guidance further assumes that the applicant is not referencing an existing Design Certification. Section D provides modifications to the Section C guidance for two alternative licensing approaches:

- Two-step licensing—The applicant first applies for and obtains a Construction Permit under 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities, and subsequently applies for and obtains an Operating License also under 10 CFR Part 50.
- Design certification—The applicant is a reactor vendor that applies for a standard DC under 10 CFR Part 52, Subpart B. It does not contain site-specific information. A future applicant would reference the DC along with site-specific and owner-specific information as part of a COL application.

Commented [A211]: See previous comment on limiting this to CP applicant only since OL = COL.

Page 27: [1] Commented [A86] Author

Because this guidance is for non-LWRs, why give examples for LWRs? The NRC recently completed SCALE demonstration calculations for heat pipe and high-temperature gas cooled reactors. The NRC has SCALE demonstration calculations underway or planned for other non-LWR designs.

Page 27: [2] Commented [A88R86] Author

LWR-SMRs may use LMP so including these in the TICAP guidance is helpful, in my opinion.

Page 27: [3] Commented [A89] Author

The attenuation mechanisms credited in the source term evaluation should be described along with their basis.

Page 27: [4] Commented [A90] Author

Because this guidance is for non-LWRs, why give examples for LWRs? The NRC is adding modeling to MELCOR for non-LWRs. The NRC recently completed MELCOR demonstration calculations for heat pipe and high-temperature gas cooled reactors. The NRC has MELCOR demonstration calculations underway or planned for other non-LWR designs.

Page 27: [5] Commented [A93] Author

It is not clear if transport also includes modeling of radionuclide release barriers and radionuclide retention and removal.

The applicant should describe the barrier assessment. In addition, the applicant should describe retention and removal of radionuclides by engineered systems or by natural processes, and justify the applicability of the analytical methodologies to the characteristics of the plant, including a discussion of the underlying experimental or analytical basis. Credit for engineered systems in DBAs should be justified.

Page 27: [6] Commented [A94R93] Author

Seems important to add what Michelle suggested.

Page 27: [7] Commented [A95] Author

Should also include "time-dependent release rates."

Page 27: [8] Commented [A98] Author

Specifically the MS element. Staff has not endorsed yet. Still in process.

Page 27: [9] Commented [A100] Author

I suggest referencing INL/EXT-20-68717 which describes NRC, DOE, and industry approaches to source term development.

Page 27: [10] Commented [A101] Author

For CP application only. OL applications should include complete information

Page 28: [11] Commented [A107] Author

I generally agree with the contents of this section and have no significant comments.

Items that I noticed absent from this document, and that are generally significant inputs into a transient/accident analysis, are safety limits and limiting safety system setpoints.

Page 28: [12] Commented [A109] **Author**

I generally agree. I took the "The applicant should justify the applicability of the analytical methodology to the characteristics of the plant, including a discussion of the underlying experimental or analytical basis" to very generically and broadly cover what Chris pointed out. Granted, one sentence does NOT provide detail. I noted that Section 3.4 points to RG 1.203, which covers those topics and may be a good pointer to add in this section as well.

Page 28: [13] Commented [A112] **Author**

This statement seems a little vague as currently written. I would recommend something to the effect of "If the DBA calculation predicts a release of radionuclides, that release should be considered as part of the source term discussed above."

Page 28: [14] Commented [A114] **Author**

I think a high-level discussion of the methodology should include any major assumptions or treatments that apply to all analyses using the methodology as well as any significant new models that may have been developed to simulate plant response.

Page 28: [15] Commented [A116] **Author**

I've typically seen technical reports submitted on the docket and incorporated by reference, not just available for audit.

Page 28: [16] Commented [A118] **Author**

If this section is intended to include actual methodologies as opposed to referencing a separately approved one, then the guidance here needs to be much more detailed. Discussion should include uncertainty, bias, data, V&V, etc.

Page 28: [17] Commented [A120] **Author**

The applicant should provide a description of the assumptions used in the consequence assessment for the LMP implementation, including description of the atmospheric transport modeling, exposure pathways, dose modeling, meteorological data needs, etc. The discussion should include computer codes used and justification of applicability of the analytical models. Describe considerations included in the consequence uncertainty quantification. With regard to radiological consequence assessment see discussion in the RC element technical requirements in the non-LWR PRA Standard ASME/ANS RA-S-1.4-2021 for example.

Methodology may be a reference instead of described in the FSAR.

Page 28: [18] Commented [A122] **Author**

There should be more structure to this than making this optional. For example, if there are safety related structures, piping/vessels, electrical equipment, then sections to evaluate the seismic design of structures, mechanical piping, and electrical systems should not be optional.
Is this the section that would discuss seismic and dynamic qualification of mechanical and electrical equipment?

Page 28: [19] Commented [A123] **Author**

If these analysis descriptions are not provided here then they need to be provided somewhere else in the SAR.

Page 62: [20] Commented [A198] Author

This guidance is insufficient to properly document important aspects of the program. The following describes two examples where additional guidance is appropriate:

<u>First example – Maintenance Program</u>
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. NRC RG 1.160 provides additional guidance on one way to meet the requirements of 10 CFR 50.65. Current evolutionary LWR licensees have maintenance program descriptions that incorporate by reference NEI 07-02A, "Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed under 10 CFR Part 52." The staff, in its SER, determined that incorporation of NEI 07-02 by reference in a COL application will provide an acceptable method for (1) complying with the requirement in 10 CFR 52.79(a)(15) that FSARs contain a description of the program, and its implementation, for monitoring the effectiveness of maintenance to meet the requirements of Section 50.65 and (2) satisfying the acceptance criteria of SRP 17.6. Topics addressed in the NEI template include:

- Maintenance rule scoping per 10 CFR 50.65(b)
- Monitoring and corrective action per 10 CFR 50.65(a)(1)
- Preventive maintenance per 10 CFR 50.65(a)(2)
- Periodic evaluation of monitoring and preventive maintenance per 10 CFR 50.65(a)(3)
- Risk assessment and risk management per 10 CFR 50.65(a)(4)
- Maintenance Rule Training And Qualification
- Maintenance Rule Program Relationship With Reliability Assurance Activities
- Maintenance Rule Program Relationship With Industry Operating Experience Activities
- Maintenance Rule Program Implementation