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Cc: Steven Nesbit; Michael Tschiltz; Chisholm, Brandon Michael; Shirley, Barry J.; Daniel, Charles R.; Cubbage, Amy; Reckley, William; Segala, John
Subject: [External_Sender] NEI's Annotated Outline
Attachments: 2020-10-20 TICAP Guidance Document - Annotated Outline.pdf

Greetings Joe,
Please find attached the annotated outline for the to-be-proposed NEI Guidance Document. Please note that this is an InProgress product and the structure of the final guidance document could be slightly or significantly different. However, the document provides a platform for near term actions and the expected scope of the guidance document.
Best regards,
Amir

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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 A
Issued for Internal Team Review and Comment

Document Number
SC-16166-103 Rev A

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

October 20, 2020

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Office of Nuclear Energy
Under DOE Idaho Operations Office
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Disclaimer

Note to reviewers: This document provides an annotated outline (AO) for the primary TICAP product – an NEI guidance document to be submitted to NRC for review around September 2021.

The AO provides a framework for the TICAP team and others to understand how the various elements of the project will fit together in the final report. Some of the TICAP work is still under development, so the content of the AO is generally a description of what the content of the final report will be. However, there are some sections in which the words that are anticipated to be part of the final report are provided in the AO. The distinction between types of material is straightforward and described below.

- Words that are anticipated to be part of the final document are in normal type but brown font.
- Words that describe the material to be provided are italicized.

Abstract

The abstract will be prepared once the report nears completion.

Table of Contents

List of Abbreviations

AOO	Anticipated Operational Occurrence
ARRTF	Advanced Reactor Regulatory Task Force
BDBE	Beyond Design Basis Event
CP	Construction permit
COL	Combined license
DBA	Design Basis Accident
DBE	Design Basis Event
DC	Design certification
DID	Defense-in-Depth
DOE	Department of Energy
FSF	Fundamental Safety Function
LBE	Licensing Basis Event
LMP	Licensing Modernization Project
LWR	Light water reactor
NEI	Nuclear Energy Institute
NGNP	Next Generation Nuclear Plant
non-LWR	Non-light water reactor
NRC	Nuclear Regulatory Commission
NSRST	Non-Safety-Related with Special Treatment
OL	Operating license
PRA	Probabilistic Risk Assessment
PSF	PRA Safety Function
QHO	Quantitative health objective
RIPB	Risk-informed and performance-based
RSF	Required Safety Functions
SAR	Safety Analysis Report
SR	Safety-Related
SSCs	Structures, Systems, and Components
TICAP	Technology Inclusive Content of Application Project

1.0 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 solely to applicants that utilize the 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." 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."

1.1 Purpose

The guidance in this report is focused on the portions of the SAR containing material addressed in NEI 18-04, and it will 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 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, 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 reactors.
    - Provide a consistent and coherent approach for establishing the SAR scope and level of detail guidelines for various advanced technologies and designs.

  - Encourage innovation by focusing on the final results as opposed to the pathway taken to achieve the results.

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

- The NRC's longstanding focus on and commitment to continuous improvement.
- 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 how 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.

1.2 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 LWRs using 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 rather than a thermal neutron spectrum. A range of fuel types is under consideration, including fuel dissolved in molten salt and circulated throughout the primary 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 radionuclide releases. Structural materials may be different, particularly for high-temperature reactors. 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-authorized TICAP, a utility-led project, was initiated to collaborate with NRC to achieve the objective of modernizing the regulatory framework to improve the effectiveness and efficiency of its reviews. The project recognizes that significant levels of industry input and advocacy are needed in collaboration with NRC to enable the regulatory changes needed for advanced reactors.

TICAP built on the foundation that was successfully established in NEI 18-04. That document presented a modern, technology inclusive, RIPB process for selection of Licensing Basis Events (LBEs); safety classification of Structures, Systems, and Components (SSCs); specification of performance requirements for SSCs including special treatments; and evaluation of Defense-in-Depth (DID) adequacy for non-LWRs. NRC endorsed the NEI 18-04 guidance with the publication of Regulatory Guide 1.233. The TICAP guidance contained herein focuses on the portion of the application related to LMP and the documentation of the applicant's safety case. Ultimately, the information included in the application must demonstrate reasonable assurance of adequate protection of public health and safety.

This guidance document was developed as part of two-year effort by a 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. The final guidance in this document reflects the lessons learned from those tabletop exercises.

1.3 Scope

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 is based on the appropriate level of design-specific information that should be provided to demonstrate that the design's safety case meets the regulatory standards for 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 scientific, technical, administrative, and managerial 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:

- Identifying design-specific safety functions that are adequately performed by design-specific SSCs during the LBEs.
- Establishing design-specific features (programmatic, e.g., inspections, or physical, e.g., redundancy) to provide reasonable assurance that credited SSC functions have adequate reliability and availability to prevent and mitigate the LBEs.

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

- 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 (LBEs, Required Safety Functions (RSFs), Safety-Related (SR) SSCs, DID, etc.).
- 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.

1.4 Organization of this Report

This section provides an overview of each section of this guidance document, including appendices, and how they relate to one another. It is essentially a roadmap for the entire report.

Section 1 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 2 provides 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 guidance assumes the license applicant is requesting a COL for an advanced reactor under 10 CFR Part 52 and is not referencing either an early site permit or a design certification.

Section 3 addresses changes required for the guidance if the applicant is following one of two different licensing approaches:

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

Section 4 summarizes the results and conclusions of the project.

The appendices are provided for information and are not part of the content of application guidance in the report. NRC is not requested to review and endorse the appendices.

Appendix A provides a description of the LMP-based affirmative safety case, its constituents, and the type of information, as well as tools used to evaluate a design against the performance objectives of the technology inclusive fundamental safety functions. It also discusses the outputs generated to define the design. This is done by labeling the LMP evaluation process and its outputs as providing answers to the following four questions:

- **What** are the performance objectives for the FSFs?
- **When** do the FSFs' performance objectives need to be demonstrated?
- **How** do plant capabilities (functional and structural) demonstrate that the performance objectives of the fundamental safety functions are met?
- **How well** do these capabilities need to be performed to provide reasonable assurance?

Appendix B provides (i) a summary of mapping current NRC nuclear power plant regulations to the FSFs and (ii) a summary of binning the General Design Criteria to the "What," "When," "How," and "How Well" questions noted above.

This information supports the reasonableness of the LMP-based affirmative safety case by providing evidence that:

- The intent of the current regulatory requirements is to provide reasonable assurance that a design meets the performance objectives of the FSFs.
- By answering the "When," "How," and "How Well" questions, the LWR General Design Criteria, when satisfied, provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

Thus, the mapping and binning activities support the conclusion that meeting the performance objectives associated with the FSFs provides reasonable assurance that the underlying safety objectives of NRC regulations have been evaluated and met.

Appendix C summarizes the insights obtained from tabletop exercises in which advanced reactor developers applied the draft guidance to their specific technologies.

2.0 DEVELOPMENT OF SAR INFORMATION

2.1 Overview

This document describes the necessary information provided in 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.

2.2 SAR Outline

Figure 1 provides a high-level outline of the SAR, and the following sections describe the content that applicants will provide. The outline is intended to present the overall safety case first and then provide the specific supporting design and operating details in subsequent chapters.

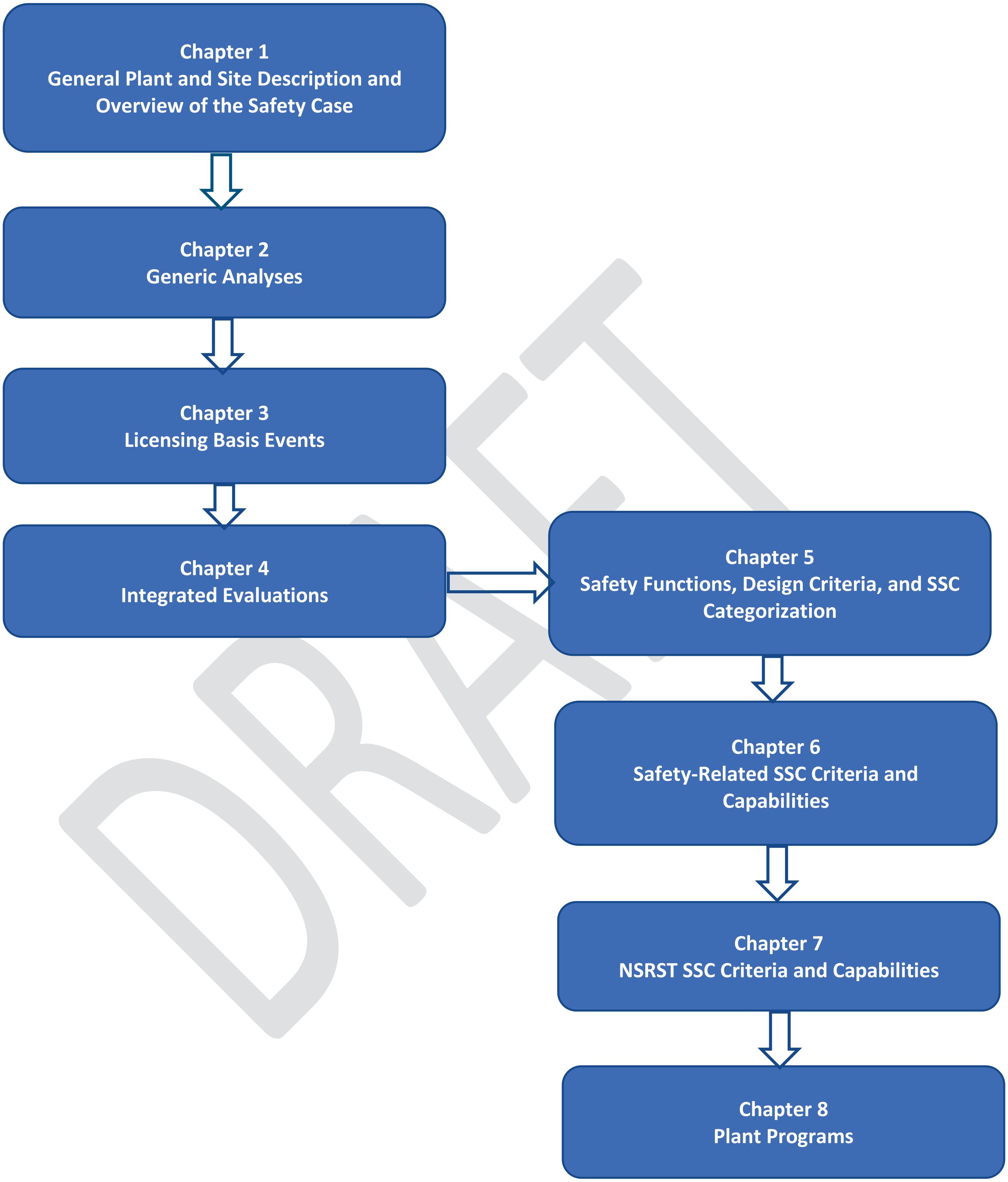

Figure 1. SAR Outline

2.3 Detailed Guidance

This section provides the detailed guidance for scope and level of detail on the formulation of the SAR content. Subsections address each chapter of SAR that relates to the LMP-based affirmative safety case. The information will be developed under WBS 10.

2.3.1 Chapter 1—General Plant and Site Description and Overview of the Safety Case

This section will provide guidance on the content and level of detail, including subsection organization, as appropriate. It will include:

- *Overview of technology (size of the reactor and planned commercial application of the design—power production, industrial application, etc.)*
- *General description of the plant systems and roles that they play in normal and off-normal conditions, including refueling*
- *General site characteristics*
- *Summary of safety case findings*
    - *Overview of affirmative LMP-based safety case methodology, including reference to NEI 18-04 and any deviations from the approved methodology*
    - *Summary of FSFs*
    - *Summary of LBEs with focus on DBAs*
    - *Summary of radiological consequence assessment*
    - *Summary of how the design provides that FSFs are met—key plant attributes and design features that provide reasonable assurance of adequate protection of public health and safety*
    - *Evaluation of DID capabilities*

Chapter 1 is "for information" and is not needed for the demonstration of reasonable assurance of adequate protection. The other SAR chapters provide the licensing basis of the design. While the licensee is expected to keep it up to date, Chapter 1 is not considered in change control evaluations (10 CFR 50.59) and other SAR chapters govern in the event of any discrepancies.

2.3.2 Chapter 2—Generic Analyses

Certain analyses are common to a number of LBE analyses. This section of the SAR provides information on those analyses.

2.3.2.1 Probabilistic Risk Assessment

The safety case is based on the NEI 18-04 methodology, and a technically sound Probabilistic Risk Assessment (PRA) is fundamental for exercising the methodology. Nonetheless, the PRA is *only a tool. Moreover, the PRA is not static; it is constantly evolving as models are added or improved, the plant configuration changes, and technology matures.*

Given its fundamental role in the NEI 18-04 methodology, the technical adequacy of the PRA will appropriately be evaluated by NRC as part of its licensing review. However, all of the information contained in the PRA is clearly not amenable or appropriate for inclusion in the SAR. It is expected that the NRC's PRA review will take place in a separate but related activity, perhaps as a topical report or a technical report with a regulatory audit.

The PRA information included in the SAR will be at a summary level only. It is included near the beginning of the SAR because of the PRA's prominent role in exercising the NEI 18-04 methodology to identify LBEs. This information is "for information" and not considered in change control evaluations.

Overview of PRA

This section summarizes the scope, methodology, and pedigree of the PRA. The pedigree is intended to be (i) a statement of compliance (with any exceptions) with the non-LWR PRA standard, ASME/ANS RA-S-1.4-2020, the manner in which the standard was used, and PRA peer review findings, or (ii) an alternative means of demonstrating PRA technical adequacy that may be proposed by the applicant.

Summary of Key PRA Findings

This section provides a summary of findings from the PRA. This includes the results for each modeled risk metric, including the frequency and consequences of all modeled event sequence families and risk significant contributors to PRA model elements. Such elements include initiating events, plant operating states, sources of radionuclides, event sequence families, and release categories.

2.3.2.2 Source Term

Source term refers to the type, quantity, and timing of the release of radioactive material from a facility during a postulated event. The source term varies with the reactor design and 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 of the events considered for the reactor, that information may be provided in this section rather than with each event.

2.3.2.3 Meteorology

This section describes the program used to measure and validate meteorological data as well as the methodology used to develop meteorological dispersion factors (X/Q values) used in the radiological consequence analyses for evaluated LBEs. The results are summarized.

2.3.2.4 Other Generic Analyses

The applicant may provide information about additional generic analyses used in subsequent sections. The efficiency of presenting additional generic analyses will be driven by the nature *of the facility and the LBE analyses. These sections are optional and up to the discretion of the applicant.*

2.3.3 Chapter 3—Licensing Basis Events

2.3.3.1 Licensing Basis Event Selection Methodology

This section will summarize the results of the process of condensing the large number of event sequences considered in the PRA into event sequence families that are used to justify the selection of the AOOs, DBEs, BDBEs, and DBAs, collectively, LBEs. The identification of the LBEs is made through an iterative process described in NEI 18-04. At the time the SAR is submitted to NRC for review as part of the advanced reactor license application, the iterative process has been completed, and the results are documented in the SAR. The SAR does not document the iterative process. The applicant may choose to document its LBE selection in a separate document, such as a technical report or topical report.

2.3.3.2 Anticipated Operational Occurrences

This section identifies and describes the plant AOOs that are informed by the PRA event sequence families.

AOO 1

This section describes the course of the event, the end state, and consequences (if any). The results should demonstrate that the performance criteria of the FSFs are met.

It also identifies the PRA Safety Functions (PSFs)—functions modeled by the PRA and responsible for preventing or mitigating a release of radioactive material as a result of the event. It goes on to identify the SSCs and operator actions (if any) required to satisfy the PSFs.

The AOO description and other information will be less detailed than the corresponding information for a DBA.

The section continues through all of the AOOs.

2.3.3.3 Design Basis Events

This section identifies and describes the plant DBEs that are informed by the PRA event sequence families.

DBE 1

This section describes the course of the event, the end state, and consequences (if any). The results should demonstrate that the performance criteria of the FSFs are met.

It also identifies the PSFs, the SSCs, and operator actions (if any) required to satisfy the PSFs.

The DBE description and other information will be less detailed than the corresponding information for a DBA.

This section continues through all of the DBEs.

2.3.3.4 Beyond Design Basis Events

This section identifies and describes the plant BDBEs that are informed by the PRA event sequence families.

BDBE 1

This section describes the course of the event, the end state, and consequences (if any). The results should demonstrate that the performance criteria of the FSFs are met.

This section identifies the PSFs—functions modeled by the PRA and responsible for preventing or mitigating a release of radioactive material as a result of the event. It goes on to identify the SSCs and operator actions (if any) required to satisfy the PRA safety functions.

The BDBE description and other information will be less detailed than the corresponding information for a DBA.

This section continues through all of the BDBEs.

2.3.3.5 Design Basis Accidents

This section identifies and describes the plant DBAs that are derived from the DBEs.

DBA 1

This section describes the course of the event, the end state, and consequences (if any). Included are the acceptance criteria, a description of the analytical methods used, analysis assumptions, and analysis results, including radiological consequences. The results should demonstrate that the performance criteria of the FSFs are met.

This section identifies the PSFs—functions modeled by the PRA and responsible for preventing or mitigating a release of radioactive material as a result of the event. It goes on to identify the SSCs and operator actions (if any) required to satisfy the PRA safety functions.

The DBA description and other information will be more detailed than the corresponding information for an AOO, DBE, or BDBE. The DBA information will generally be commensurate with the treatment of a DBA in Chapter 15 of a light water reactor SAR.

This section continues through all of the BDBEs.

2.3.4 Chapter 4—Integrated Evaluations

This chapter will present the results of the integrated evaluations that are required by NEI 18-04.

2.3.4.1 Evaluation of Integrated Plant Risk

This section provides the integrated risk of all the LBEs relative to the three cumulative risk targets.

- *The total mean frequency of exceeding a site boundary dose of 100 mrem from all LBEs should not exceed 1/plant-year. This metric is introduced to ensure that the consequences from the entire range of LBEs from higher frequency, lower consequences to lower frequency, higher consequences are considered. The value of 100 mrem is selected from the annual cumulative exposure limits in 10 CFR 20.*
- *The average individual risk of early fatality within 1 mile of the exclusion area boundary from all LBEs based on mean estimates of frequencies and consequences shall not exceed* 5×10^{-7}*/plant-year to ensure that the NRC safety goal quantitative health objective (QHO) for early fatality risk is met.*
- *The average individual risk of latent cancer fatalities within 10 miles of the exclusion area boundary from all LBEs based on mean estimates of frequencies and consequences shall not exceed* 2×10^{-6}*/plant-year to ensure that the NRC safety goal QHO for latent cancer fatality risk is met.*

2.3.4.2 Defense-in-Depth

This section will describe the results of the evaluation of DID and the DID baseline.

2.3.5 Chapter 5—Safety Functions, Design Criteria, and SSC Categorization

2.3.5.1 Principal Design Criteria and Safety-Related SSCs

Based on the results summarized in Chapters 3 and 4 and the criteria of NEI 18-04, this section identifies the RSFs and Required Functional Design Criteria from the population of PSFs. It goes on to identify the Principal Design Criteria established for the design and the SR SSCs and operator actions (if any).

For each SR SSC, the basis for such classification will be indicated in a traceable manner.

2.3.5.2 Complementary Design Criteria and Non-Safety-Related with Special Treatment SSCs

Based on the results summarized in Chapters 3 and 4 and the criteria of NEI 18-04, this section identifies the other risk significant safety functions and other safety functions for adequate DID that together comprise the Complementary Design Criteria. It goes on to identify the Non-Safety-Related with Special Treatment (NSRST) SSCs and operator actions (if any).

For each NSRST SSC, the basis for such classification will be indicated in a traceable manner.

2.3.6 Chapter 6—Safety-Related SSC Criteria and Capabilities

The Safety-Related Design Criteria and special treatment requirements are specified for each SR SSC. Information will be provided for each Safety-Related SSC to support a determination that the SSC will meet its reliability and performance targets as credited in the PRA.

There are a number of options for presenting the detailed information in Chapter 6. The optimal approach may vary between reactor designs and technologies. TICAP will develop *recommendations on the scope, level of detail, and format for information in Chapter 6. These recommendations will be informed by the planned tabletop exercises.*

2.3.7 Chapter 7—NSRST SSC Criteria and Capabilities

The special treatment requirements are specified for each NSRST. Information will be provided for each SR SSC to support a determination that the SSC will meet its reliability and performance targets as credited in the PRA.

There are a number of options for presenting the detailed information in Chapter 7. The optimal approach may vary between reactor designs and technologies. TICAP will develop recommendations on the scope, level of detail, and format for information in Chapter 7. These recommendations will be informed by the planned tabletop exercises.

2.3.8 Chapter 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 plant programs relied upon to meet reliability and performance targets. This chapter would include a discussion of such programs, which could include such areas as human factors, training, and reliability assurance.

3.0 ALTERNATIVE LICENSING PATHS

NRC regulations provide applicants with a number of options for obtaining an OL for a nuclear power reactor. The guidance in Section 2 assumes an applicant is applying for a 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 DC or an existing early site permit. In this scenario, the applicant would need to provide the maximum amount of information compared to other approaches.

Advanced reactor applicants may choose one of the alternative licensing pathways. Section 3 addresses modifications to the guidance provided in Section 2 for several alternative pathways deemed to be reasonably likely. Those pathways are:

- Two-step licensing—The applicant first applies for and obtains a CP under 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities, and subsequently applies for and obtains an OL 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. A future applicant would marry the DC with a site as part of a COL.

3.1 Two-Step License

This section addresses differences between (i) the baseline approach described in Section 2 for a COL applicant and (ii) an applicant that obtains a construction permit, constructs the plant, and obtains an operating license. Issuing a CP does not constitute approval to operate a facility. Accordingly, NRC expects the information in a CP application to be supplemented and updated in the OL application. CP applicants typically provide less information than OL applicants. In addition, the PRA and the LBE analyses would not have attained the maturity expected for an OL application.

The application content for all licensing paths may 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 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 OL SAR under Part 50 is expected to be commensurate with the combined CP and OL SAR submitted under Part 52 (the baseline process). Therefore, this section addresses only the differences at the CP stage of a two-step license.

The remainder of Section 3.1 will address differences between the CP SAR and the COL SAR as described in Section 2.

3.2 Design Certification

This section addresses differences between (i) a SAR for a COL application as described in Section 2 and (ii) a SAR submitted as part of a 10 CFR Part 52 Subpart B design certification application. The design certification SAR lacks site-specific information, so external hazards are generally addressed through parameter envelopes that must eventually be shown to bound actual site conditions.

The remainder of Section 3.2 will address differences between the DC SAR and the COL SAR as described in Section 2.

This information in Chapter 3 will be developed under WBS 7.

4.0 SUMMARY AND CONCLUSIONS

This summarizes the work and concludes that the guidance provided in Sections 2 and 3 should satisfy the goals set forth in Section 1.

5.0 REFERENCES

To be added.

Appendix A LMP-Based Affirmative Safety Case

This guidance document assumes that an applicant is using the methodology defined in NEI 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development" and endorsed by NRC in 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 NEI 18-04 methodology uses a modern, technology inclusive, RIPB process for selection and evaluation of LBEs; safety classification of SSCs and associated risk-informed special treatments; and determination of DID adequacy for non-LWRs. The process for developing these three elements requires a design-specific safety case to be used in the evaluation process. The TICAP application guidance will focus on the portion of the application related to NEI 18-04 and the applicant's safety case.

This appendix provides a description of the LMP-based affirmative safety case, its constituents, and the type of information as well as tools used to evaluate a design against the performance objectives of the technology inclusive fundamental safety functions. It also discusses the outputs generated to define the design. This is done by labeling the LMP evaluation process and its outputs as providing answers to the following four questions:

- ***What** are the performance objectives for the FSFs?*
- ***When** do the FSFs' performance objectives need to be demonstrated?*
- ***How** do plant capabilities (functional and structural) demonstrate that the performance objectives of the fundamental safety functions are met?*
- ***How well** do these capabilities need to be performed to provide reasonable assurance?*

This appendix is not part of the content of application guidance in the report. It is provided for information, and NRC will not be requested to review and endorse the appendix.

The content is based on the work of WBS 6.

Appendix B Fundamental Safety Function Mapping and General Design Criteria Binning

This appendix provides (i) a summary of mapping current NRC nuclear power plant regulations to the FSFs and (ii) a summary of binning the General Design Criteria to the "What," "When," "How," and "How Well" questions discussed in Appendix A.

This information supports the reasonableness of the LMP-based affirmative safety case by:

- *Providing evidence that the intent of the current regulatory requirements is to provide reasonable assurance that a design meets the performance objectives of the fundamental safety functions.*
- *Providing evidence that by answering the "When," "How," and "How Well" questions, the light water reactor General Design Criteria, when satisfied, provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.*

Thus, the mapping and binning activities support the conclusion that meeting the performance objectives associated with the FSFs provides reasonable assurance that the underlying safety objectives of NRC regulations have been evaluated and met.

This appendix is not part of the content of application guidance in the report. It is provided for information, and NRC will not be requested to review and endorse the appendix.

This appendix is based on the work of WBS 2 and WBS 3, for which full reports were developed. These appendices will summarize the work but not replicate the full reports.

Appendix C Tabletop Exercises

This section provides a summary of the tabletop exercises in which the guidance in Section 2 was applied to advanced reactor designs.

The significant lessons learned from the tabletop exercises are also summarized.

This appendix is not part of the content of application guidance in the report. It is provided for information, and NRC will not be requested to review and endorse the appendix.

This appendix is based on the work in WBS 9. It will not include the level of detail provided in the reports written up for each tabletop exercise.